SHARE

H2020 NFRP-2018 CSA: Coordination and Support Action

Grant Agreement n° 847626

D3.1: Report detailing applicable technologies/ methodologies

Author: Federica Pancotti, [SOGIN]

With contributions from: Fanny Fert [CEA], Ludovic Vaillant [CEA/CEPN], Jorge Borque Liñán, Emilio Garcia Neri [ENRESA], Istvan Szőke, Lucas Stephane [IFE], Angelika Bohnstedt, Simone Müller [KIT], Gintautas Poškas, Povilas Poskas, Egidijus Babilas [LEI], Samantha Ree, James Dewar, Ed Butcher [NNL], Kurt Van Den Dungen, Luc Noynaert [SCK-CEN], Alessandro Mattioli, Domenico Lisi, Rossella Sciacqua, Valerio Maturo, Gianpaolo Di Bartolomeo, Carlo Rusconi [SOGIN], Markus Airila, Raimo Launonen, Rafael Popper, Antti Räty, Anumaija Leskinen, Liro Auterinen, Jaakko Leppänen, Petri Kotiluoto [VTT]

Reviewers: Christine Georges [CEA], Réka Szőke [IFE], Laura Aldave de las Heras [JRC], Muhammad Junaid Ejaz Chaudhry [KIT], Anthony Banford (NNL]

SHARE

A roadmap for research in Decommissioning

Deliverable nature	Report
Dissemination level	Public
Contractual delivery date	31.10.2020
Actual delivery	24.03.2021
Version	V2

Version history

Version	Date	Editors	Description
V1	13.08.2020	All	Writing of the deliverable
V2	29.01.2021	All	New version after Review from WP3 participants

Deliverable content

Abstract

One of the objectives of SHARE Project is to identify existing and emerging innovative techniques and solutions for decommissioning, employed across the nuclear industry, to meet the current and future needs.

This deliverable consists of a general review of international best practices and advanced technologies in relation to the thematic and sub-thematic areas defined in the questionnaire (WP2).

The research was conducted through existing databases, workshops and literature (for example, journals, conference papers, industry reports, etc). The Consortium and Expert Review Panel workshop was used to review and verify the consolidated knowledge.

The report is structured in 8 Chapters which cover all the thematic and sub-thematic areas identified in the questionnaire (WP2). At the beginning of each Chapter a description of historic and on-going international initiatives is reported.

The different sub-thematic themes are organised into specific Sections where a general description of the state of the art of technologies, methodologies and best practices is reported together with experiences and case studies.

When possible, a short evaluation of the current technology/knowledge and the possible areas for further development/implementation is described. This report, mapping the existing state of the art technologies and organisational best practice currently employed across the nuclear industry will be used to compare the current technologies, with the results of the needs identified during the GAP analysis/benchmarking (D3.2).

List of figures

List of tables

1. Safety and Radiological Protection aspects

Safety and risk management is one of the most important aspects in nuclear decommissioning projects. Compared to construction and operation, conditions encountered by workers in decommissioning projects are challenging in both a technical and a human centred perspective. The National Research Council (NRC): “Many current technologies are labour intensive and time consuming. Most current Dismantling and Decommissioning (D&D) technologies require hands-on contact by workers who must operate powerful equipment (e.g. plasma torches, saws, and lifting devices) while wearing bulky protective clothing. The facilities present hazards to workers that include penetrating irradiative areas, airborne contamination, toxic chemicals, and other industrial hazards.” [NRC 2001].

Since decommissioning workers must penetrate areas with the risk of exposure to radiation, pre-evaluation of planned work using an ALARA (as low as reasonably achievable) assessment is important. Monitoring the worker’s active or passive risk/dose ensures that unacceptable exposure is detected, and measures for avoiding further exposure can be implemented. The safety/radiological evaluation of work prior to the work being completed, will ensure that safety breaches are detected before work, and an optimal work procedure is implemented from an ALARA perspective. In addition, pre-job dose estimation can also be used for the preparation of the workers for plausible and extreme emergency situations to minimise excess exposure that could be received in a variety of scenarios. However, risk/radiological conditions in decommissioning work can be complex, exhibiting strong variations in both location and time. Hence, methods applied for risk/dose evaluations must be quick enough to allow for changing situations, but also accurate enough to yield dependable results.

In tandem with more traditional safety and radiological protection techniques, 3D real-time work and radiological simulation based ALARA evaluation of planned normal and/or intervention work are increasingly being applied. The on-going research and rapid advancement of underlying technologies, such as high-performance computing, have paved the way for the adoption of more modern techniques being used, as a general tool for ensuring safety in decommissioning.

International initiatives

A summary of the international initiatives related to Safety and Radiological Protection aspects are listed in the following.

A detailed overview of the international initiatives related to Safety and Radiological Protection) to support the chapter *International harmonisation of safety standards and safety approach for decommissioning* is reported in the specific sub-chapter 0.

IAEA Initiatives

- A set of International Atomic Energy Agency (IAEA) safety standards, safety guides and recommendations [1], [2] provide information on different decommissioning topics that can be applied when developing national regulatory guidance. They cover a wide range of sites and facilities, from small nuclear facilities to large complex reprocessing and reactor sites. These references [3], [4] may assist in assessing the adequacy of the existing regulations and regulatory guides or serve as a road map for countries that are developing their own regulations for the first time.

NEA Initiatives

- The Nuclear Energy Agency's (NEA) Expert Group on Legacy Management (EGLM) [5] is an expert panel on Legacy Management. Their report [6] is based on several case studies and outlines that a holistic approach to management and regulation of the hazards and risks is required, to achieve proportionate risk management and overall optimisation. This implies the need to consider chemical and other hazards, alongside the radiological risks. Proportionate health and safety and risk management strategies should be adopted and applied corresponding to the regulatory requirements.

  While European and national regulations and standards cover the management of radiological risk and the management of industrial risk, there is a growing need from utilities and users for regulatory and practical guidance for the simultaneous management of a large spectrum of risks, for both workers and the public.

  Several initiatives coming from NEA and dealing with decommissioning, starting with the creation of the NEA Co-operative Programme for the Exchange of Scientific and Technical Information on Nuclear Installation Decommissioning Projects (CPD[7]). CPD addressed several topics related to decommissioning (cost, waste management, etc.) which do not include so far occupational radiological protection.

---

[1] Decommissioning of facilities. IAEA safety standards series, General safety requirements Part 6, GSR Part 6. Vienna, IAEA, 2014.

[2] Advancing implementation of decommissioning and environmental remediation programmes, CIDER project: baseline report. IAEA nuclear energy series, NW-T-1.10. Vienna, IAEA, 2016

[3] Decommissioning of facilities. IAEA safety standards series, General safety requirements Part 6, GSR Part 6. Vienna, IAEA, 2014.

[4] Advancing implementation of decommissioning and environmental remediation programmes, CIDER project: baseline report. IAEA nuclear energy series, NW-T-1.10. Vienna, IAEA, 2016

[5] Challenges in Nuclear and radiological Legacy Management, NEA Report 7419, 2019. https://www.oecd-nea.org/rp/pubs/2019/7419-eglm.pdf

[6] Challenges in Nuclear and radiological Legacy Management, NEA Report 7419, 2019. https://www.oecd-nea.org/rp/pubs/2019/7419-eglm.pdf

[7] https://www.oecd-nea.org/jointproj/decom.html

European Commission (EC) Initiatives

- The H2020 PLEIADES "PLatform based on Emerging and Interoperable Applications for enhanced Decommissioning processes" project (see Chapter 2) introduces innovative technologies to support decommissioning safety.

Other Initiatives

- International Organization for Standardization (ISO) is a network of 163 national standardisation bodies, ISO develops International standards for products, services, processes, materials and systems and for conformity assessment, managerial and organisational practice.
  Within ISO's Technical Committee (TC) 85 "Nuclear energy, nuclear technologies and radiological protection" (ISO/TC85), are 28 participating countries, 4 observing countries and representation from international organisations, including the European Commission (EC) and IAEA. The radiological protection sub-committee, ISO/TC 85/ SC 2, contributes to radiological protection in the fields where the standard may have a role to play and produces ISO standards[8] of specific interest in decommissioning such as:
  - Measurement of radioactivity — Measurement and evaluation of surface contamination — Part 1: General principles[9];
  - Evaluation of surface contamination — Part 2: Tritium surface contamination[10];
  - Measurement of radioactivity - Measurement and evaluation of surface contamination — Part 2: Test method using wipe-test samples[11];
  - Evaluation of surface contamination — Part 3: Isomeric transition and electron capture emitters, low energy beta-emitters (maximum beta energy (E bêta$_{max}$) less than 0.15 MeV)[12] ;
  - Measurement of radioactivity — Measurement and evaluation of surface contamination — Part 3: Apparatus calibration[13] ;
  - Radiation protection — Clothing for protection against radioactive contamination — Design, selection, testing and use[14].

- The Information System on Occupational Exposure (ISOE) [15] was created in 1992 to provide an international forum for radiation protection professionals from nuclear electricity licensees and national regulatory authorities to share dose reduction information, operational

8 https://www.iso.org/committee/50280/x/catalogue/
9 ISO 7503-1:2016
10 ISO 7503-2:1988
11 ISO 7503-2:2016
12 ISO 7503-3:1996
13 ISO 7503-3:2016
14 ISO 8194:1987
15 http://www.isoe-network.net

experience and information to improve the optimisation of radiological protection at nuclear power plants (NPPs). ISOE is jointly sponsored by the Organisation for Economic Co-operation and Development (OECD) NEA and the IAEA.

Within ISOE, a Working Group on Radiological Aspects of Decommissioning Activities in Nuclear Power Plants (WG DECOM) is aimed at providing a forum for experts to develop a process within the ISOE programme to better share operational Radiological Protection (RP) data and experience for NPPs in some stage of decommissioning or in preparation for decommissioning.

According to ISOE WG DECOM, Collective Radiation Exposure (CRE) associated with NPP decommissioning usually reach several person-Sv while '*RP staff usually face huge challenges with increase potential for internal contamination, asbestos, lead, etc. which must be carefully dealt with (holistic approach)*'.

The creation of the WG DECOM within the ISOE community outlines the need for a practical experience (success and drawbacks) information exchange network. Experiences from past nuclear facilities decommissioning should feed preparation of future decommissioning activities to better manage occupational exposures.

1.1 International harmonisation of safety standards and safety approaches for Decommissioning

Nuclear safety and the safe management of radioactive waste and spent fuel are national responsibilities. The EU Member States have national regulatory bodies and national legislation to regulate their nuclear activities, including the management of spent fuel and radioactive waste. The three main influences on the development of national nuclear safety legislation and practices are:

- The international frameworks, in particular, international safety conventions;
- EU legislation in the field of nuclear and radioactive waste safety, in particular the Directive on Nuclear Safety from 2009 with the amendment in 2014 and the Directive on Management of Spent Fuel and Radioactive Waste from 2011. These two framework directives supplement the Basic Safety Standards (BSS) Directive on radiation protection;
- The international guidance and regulatory methodologies, such as those published by international bodies like the International Atomic Energy Agency (IAEA), the Nuclear Energy Agency (NEA, part of the OECD), and other groups.

Harmonisation of safety standards constitutes an essential contribution to maintain a high level of nuclear safety. A series of initiatives have been undertaken in this field, coming from industry, regulatory and public authorities. Dialogue is necessary between the relevant stakeholders. To achieve a harmonised level of nuclear safety various organisations and initiatives have developed sets of requirements and instruments for supervision over the last 25 years.

1.1.1 Regulatory

1.1.1.1 EU Legislation

The European Atomic Energy Community (Euratom) Treaty[16] forms the basis of many EU actions related to radiation protection, nuclear safety and the safe management of radioactive waste and spent fuel, as well as other activities which use radioactive sources for research, industrial and medical purposes. These activities include research, the drawing-up of safety standards, and the peaceful uses of nuclear energy. The EU Member States can interact with Euratom research activities through the EU Framework Programmes for research and technological development.

The Euratom Treaty does not explicitly address particular aspects of nuclear installation safety. While the European legislation provides a binding framework as far as authorisation, inspection and enforcement, the responsibility for implementing and enforcing such European legislation lies with the Member States.

The European Commission (EC) ensures that the community establishes uniform safety standards to protect the health of workers and of the general public and ensures that the standards are applied. In

[16] https://europa.eu/european-union/sites/europaeu/files/docs/body/consolidated_version_of_the_treaty_establishing_the_european_atomic_energy_community_en.pdf

2003, the EU institutions underlined the importance of the safe decommissioning of nuclear installations, including long-term management of radioactive waste and spent fuel. In the following years the EC acted accordingly, issuing several acts, including the 2006 'Recommendation on the management of financial resources for the decommissioning of nuclear installations, spent fuel and radioactive waste' and later the directives which refined the regulatory framework in the nuclear energy sector, with an enhanced focus on safety and waste management, and including decommissioning.

The Council Directive 2009/71/Euratom[17] established a 'Community Framework for Nuclear Safety', which was adopted in June 2009 and revised in July 2014.[18] The purpose of which was to take into account the results of the stress tests which were performed within Europe following the Fukushima accident. This Directive reinforces a national legislative, regulatory and organisational framework for nuclear safety in Europe. It strengthens, in particular, the role and independence of Europe's national regulators and endorses agreed safety objectives for NPPs, aligning with the recommendations from WENRA. At least every six years, peer reviews of national assessments for the safety of relevant nuclear installations are performed. Each assessment is based on a specific topic, the first being based on ageing management. Whereas, complementary peer reviews of national safety frameworks are performed at least every ten years. The reviews aim to highlight the responsibility of operators and regulators to ensure safety at all nuclear facilities.

The EC decided to revise its Basic Safety Standards (BSS) on radiation protection to reflect the new recommendation of the International Commission on Radiological Protection[19] (ICRP) and to strengthen the community legislation. The EC's proposal for a revised Council Directive laying down BSS for protection against the dangers arising from exposure to ionising radiation was formally adopted in May 2012[20]. The most relevant change within the updated EU BSS, from a decommissioning perspective, is related to clearance levels (for unrestricted release) for bulk material.

In July 2011, the European Council adopted its Directive establishing an EU framework for the safe management of radioactive waste and spent fuel from nuclear power plants, research, medicine and industry[21]. The EU's Radioactive Waste and Spent Fuel Management Directive require the EU Member States to have a national policy for the management of the materials concerned. To achieve this, all EU countries must draw up national programmes for the disposal of nuclear waste, including plans for the construction of nuclear waste disposal facilities.

1.1.1.2 WENRA

The Western European Nuclear Regulators Association (WENRA) aims to harmonise nuclear regulatory systems in the EU countries and associated members outside the EU. WENRA acts as an informal association to develop a common approach to nuclear safety and to provide an independent capability

[17] https://eur-lex.europa.eu/legal-content/EN/TXT/?uri=CELEX%3A32009L0071
[18] https://eur-lex.europa.eu/legal-content/EN/TXT/?uri=uriserv%3AOJ.L_.2014.219.01.0042.01.ENG
[19] https://www.icrp.org/
[20] http://www.ensreg.eu/nuclear-safety-regulation/eu-instruments/Basic-Safety-Standards-Directive
[21] https://eur-lex.europa.eu/legal-content/EN/ALL/?uri=CELEX:32011L0070

to examine current nuclear safety in its member's countries. WENRA's objectives are to become "a network of chief nuclear safety officers in Europe, exchanging information and discussing significant safety issues". The network comprises of senior regulators from different countries where NPPs are in operation. In fulfilment of its goals, WENRA established two working groups to consider harmonisation of safety approaches in Europe, one on reactor safety (RHWG) and another on decommissioning and radioactive waste/spent fuel storage (WDWG).

The decommissioning related requirements of WENRA serve to stipulate a harmonised high level of nuclear safety among the European countries. Their implementation in the national regulatory systems takes place within the full responsibility of the individual WENRA member countries. WENRA has published safety reference levels on the operation of nuclear power plants, on the decommissioning of facilities, on the storage of radioactive waste and spent fuel, and on the disposal of radioactive waste (see Table 1).

The WENRA Decommissioning Safety References Levels report[22] covers 62 safety reference levels addressing: safety management, decommissioning strategy and planning, the conduct of decommissioning, and safety verification. The report is based on IAEA's safety standards WS-R-5 and on WENRA's experiences in decommissioning. Aspects of radiation protection are mainly not addressed because they are already subject to the binding European regulation BSS[23].

1.1.1.3 ENSREG

The European Nuclear Safety Regulators Group[24] (ENSREG) is an independent, expert advisory group composed of senior officials from the national nuclear safety, radioactive waste safety and radiation protection regulatory authorities and senior civil servants with competence in these fields from the EU Member States and with representatives of the EC. ENSREG's role is to help to establish the conditions for continuous improvement and to reach a common understanding in the areas of nuclear safety and radioactive waste management in the EU. ENSREG's work also covers the financing of the decommissioning of nuclear installations in the EU. A Working Group on Waste Management and Decommissioning (WGRWMD) assists ENSREG.

1.1.1.4 IAEA

The Convention of Nuclear Safety (CNS) and the IAEA Safety Standards (provided as fundamentals, requirements and guides) are highly recognised in the nuclear community as the leading sets of requirements, due to the holistic approach, the waste management hierarchy applied and the detailed information provided. The IAEA Safety Standards are developed with the help of expert committees and often used as the basis of Member States national regulations. IAEA also develops a safety glossary of the terms used in its standard.

---

22 http://www.wenra.org/media/filer_public/2015/10/14/wgwd_report_decommissioning_srls_v2_2.pdf
23 https://eur-lex.europa.eu/eli/dir/2013/59/oj
24 http://www.ensreg.eu/

IAEA reviewed its former Safety Standards Series No. WS-R-5, "Decommissioning of Facilities Using Radioactive Material"[25], and published more recently their IAEA Safety Standards Series No. GSR Part 6, "Decommissioning of Facilities"[26]. The concepts of exclusion, exemption and clearance are addressed in IAEA Safety Standards Series No. RS-G-1.7 "Application of the Concepts of Exclusion, Exemption and Clearance"[27].

IAEA Safety Standards Series No. SSG-47 "Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities"[28], the Safety Reports Series No. 97 "Management of Project Risks in Decommissioning"[29] and IAEA Nuclear Energy Series NP-T-3.28 "Technical Support to Nuclear Power Plants and Programmes"[30] are documentation available for guidance. Also, a variety of publications in the IAEA series like TECDOC Series and Nuclear Energy Series are available providing experiences and lessons learned from the IAEA Member States on the decommissioning of facilities[31].

1.1.1.5 The OECD's Nuclear Energy Agency (NEA)

There are 28 countries (including 18 from EU) which are members of the OECD/NEA[32]. The NEA is the international focus for the developed nations on nuclear issues. It brings together several countries from North America, Europe and the Asia-Pacific region, and this membership represents much of the world's nuclear expertise. The role of the OECD NEA is complementary to that of the IAEA. It shares "best practices" amongst its members and focuses more on processes and procedures rather than standards. The NEA has several standing committees that feed into guidance on nuclear safety and radioactive waste management, including, a report regarding recycling and reuse of materials arising from the decommissioning of nuclear facilities[33], which is further described in Chapter 8.9.

1.1.1.6 MDEP

The Multinational Design Evaluation Programme (MDEP)[34] is an international initiative to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities who will be tasked with the review of new reactor power plant designs. MDEP comprises the safety authorities of 15 countries, with involvement from the IAEA and OECD Nuclear Energy Agency (NEA).

[25] https://www.iaea.org/publications/7536/decommissioning-of-facilities-using-radioactive-material
[26] https://www.iaea.org/publications/10676/decommissioning-of-facilities
[27] https://www.iaea.org/publications/7118/application-of-the-concepts-of-exclusion-exemption-and-clearance
[28] https://www.iaea.org/publications/12210/decommissioning-of-nuclear-power-plants-research-reactors-and-other-nuclear-fuel-cycle-facilities
[29] https://www.iaea.org/publications/12328/management-of-project-risks-in-decommissioning
[30] https://www.iaea.org/publications/12242/technical-support-to-nuclear-power-plants-and-programmes
[31] https://www.iaea.org/publications/search?keywords=nuclear+decommissioning
[32] http://www.oecd-nea.org/workareas/
[33] https://www.oecd-nea.org/rwm/pubs/2017/7310-recycle-decom.pdf
[34] https://www.oecd-nea.org/mdep/

1.1.1.7 ENISS

ENISS (European Nuclear Installations Safety Standards Initiative)[35] represents the nuclear utilities and operating companies from 16 European countries with nuclear power programme and enables a constructive debate with WENRA, ENSREG, the EU and the IAEA to strengthen the harmonisation of safety requirements in Europe. Additional information is reported in Table 1.

1.1.1.8 Other Initiatives

Other groups that contribute to EU nuclear safety initiatives include:

- Eurosafe[36] is an international initiative aimed at the convergence of nuclear safety practices in Europe. It collects the ideas of various European safety organisations and disseminates the information to a wide audience.
- The European Nuclear Energy Forum (ENEF)[37] is a platform to facilitate the discussion on transparency, opportunities and risks of nuclear energy. ENEF gathers relevant stakeholders in the nuclear field: EU Member State governments, European institutions, including the European Parliament and the European Economic and Social Committee, the nuclear industry, electricity consumers and civil society.

1.1.2 Industry

1.1.2.1 WANO

The World Association of Nuclear Operators (WANO)[38] is a platform created after the Chernobyl accident for the exchange of operating experience, professional and technical development amongst the NPPs, as well as technical support and exchange of information. After the Fukushima accident, WANO decided to extend its role and approved a series of recommendations. These include expanding the scope of WANO's activities, developing a world-wide integrated event response strategy, improving WANO's credibility and visibility and improving the quality of all WANO products and services.

1.1.2.2 WNA

The World Nuclear Association (WNA)[39] is an international organisation that supports the global nuclear industry. WNA established its working group on "Cooperation in Reactor Design Evaluation and Licensing" (CORDEL) with the aim of facilitating dialogue with nuclear regulators on the benefits of globally standardised designs for new reactors. Achieving reactor design standardisation will require the combined efforts of industry, regulators, policy makers, governments and international institutions.

---

35 https://www.eniss.eu/
36 https://www.eurosafe-forum.org/
37 https://ec.europa.eu/energy/topics/nuclear-energy/european-nuclear-energy-forum-enef_fr
38 https://www.wano.info/
39 https://www.world-nuclear.org/

1.1.2.3 CEN / CENELEC

European Committee for Standardisation (CEN), and the European Committee for Electrotechnical Standardisation (CENELEC) collaborate with industry partners, the EC and other stakeholders (through 65 Technical Committees) to develop and adopt European standards that support the successful implementation of European legislation.

CEN and CENELEC collaborate with the international standardisation organisations, ISO and IEC, to develop and publish standards that ensure the safety, environmental and technical requirements of the European nuclear industry[40]. The main technical bodies and activities are the following:

- CEN/TC 430 Nuclear energy, nuclear technologies, and radiological protection;
- CEN/WS 064 Phase 1 Design and Construction Code for mechanical equipment of innovative nuclear installations (European Sustainable Nuclear Industrial Initiative);
- CEN/WS 064 Phase 2 Design and Construction Codes for Gen II to IV nuclear facilities (pilot case for process for evolution of AFCEN codes);
- CLC/SR 45 Nuclear instrumentation;
- CLC/TC 45AX Instrumentation, control and electrical power systems of nuclear facilities;
- CLC/TC 45B Radiation protection instrumentation.

1.1.3 Looking Ahead on International Harmonisation

There have been substantial efforts in exchanging information to facilitate international harmonisation of approaches for decommissioning. However, there are limitations to this practise and there is a continued need to strengthen international co-operation in developing harmonised safety requirements and practices.

The 'International Conference on Advancing the Global Implementation of Decommissioning and Environmental Remediation Programmes'[41] organised by the IAEA in May 2016, concluded that the effective implementation of decommissioning and remediation programmes is strongly dependent on the establishment of appropriate regulatory regimes and associated standards to protect the safety of the workforce, the public and the long term safety of the environment, and ability of the regulatory bodies to enforce regulation. The key elements of this framework are generally well understood and appropriate regulation is in place for many countries undergoing decommissioning projects. There is a significant degree of harmonisation of standards for unconditional clearance of materials from decommissioning, although this does not generally cover conditional clearance, where a range of national approaches exists. The conference recommended that international standards and associated guidance should be developed for conditional clearance of materials from decommissioning[42]. This can be accomplished by internationally adopting an accepted clearance criteria for materials. Ambiguity over the interpretation of clearance levels (e.g. averaging of

[40] https://www.cen.eu/work/Sectors/Energy/Pages/NuclearEnergy.aspx

[41] https://www.iaea.org/publications/11155/advancing-the-global-implementation-of-decommissioning-and-environmental-remediation-programmes

[42] Report by the Conference President, Mr J.J. Zaballa, on Friday, 27 May 2016

measurements, use of statistics) continue to pose difficulties and misunderstandings in accomplishing harmonisation. Operators and regulators are required to agree on the details of clearance methods as early as possible during the planning stages of decommissioning[43].

The legislation divergence in requiring a periodic review of the decommissioning plans during the facility's operation, as recommended by the IAEA, poses as another challenge. The legal obligation to periodically update decommissioning plans is still limited to only a few countries (Finland, Sweden etc.), but is increasing.

The need for the regulatory process during decommissioning to adjust in real-time to the constantly changing plant configuration and associated hazards is a regulatory challenge. As an example, it would be desirable that regulatory consent is achieved to changes to the decommissioning plan and to work procedures, and the regulatory review of unexpected events be as timely and flexible as possible.

Finally, it is increasingly being recognised that the licensing approach to decommissioning projects should be accelerated, as a means to reduce the costs of administrative delays.

In the context of the standards applicable to the release of sites from regulatory control and making them available for alternative uses (whether as the final step of a decommissioning project or as a result of environmental remediation), the situation is significantly more complex, and a wide range of different approaches are being applied in different countries. Development of a consensus on the requirements between national regulatory and international bodies will increase acceptance and build confidence in the process.

In the context of waste management, guidance is needed on how to better integrate decommissioning and environmental remediation with waste management, especially when very large quantities of low-level waste are involved.

[43] Nuclear Decommissioning: Its History, Development, and Current Status. [Michele Laraia, 2018] https://www.springer.com/gp/book/9783319759159

Table 1.1-1 International Harmonisation in Safety Standards - Regulatory

Organisation	Description	Main Documents	Link
WENRA Working Group on Waste and Decommissioning (WGWD)	The working group on waste and decommissioning (WGWD) is mandated to analyse the current situation and the different safety approaches, compare individual national regulatory approaches with the IAEA Safety Standards, identify any differences and propose a way forward to possibly eliminate the differences. The proposals are expected to be based on the best practices among the most advanced requirements for nuclear waste facilities.	• WENRA Main tasks • Decommissioning Safety Reference Levels 2015 • Umbrella Document Interfaces and Interdependencies 2017	• http://www.wenra.org/harmonisation/working-group-waste-and-decommissioning/ • http://www.wenra.org/media/filer_public/2015/10/14/wgwd_report_decommissioning_srls_v2_2.pdf • http://www.wenra.org/media/filer_public/2017/09/28/umbrelladocument_sep_2017.pdf
European Nuclear Safety Regulators Group (ENSREG)	ENSREG is an authoritative expert body which helps the EC to 'establish the conditions for continuous improvement and to reach a common understanding in the areas of nuclear safety and radioactive waste management'.		• http://www.ensreg.eu/nuclear-safety
European Commission (EC)	The EU promotes the highest standards of nuclear safety across Europe and beyond, establishes requirements for safe long-term management of		• https://ec.europa.eu/energy/topics/nuclear-energy_en?redir=1

Organisation	Description	Main Documents	Link
	radioactive waste and plays an important role in the decommissioning of nuclear facilities.		
IAEA	IAEA Safety Standards are developed with the help of expert committees and often used as the basis of Member States national regulations. Joint Convention on the safety of spent fuel management and on the safety of radioactive waste management.	• General Safety Requirements (GSR) Part 1, Governmental, Legal and Regulatory Framework for Safety, IAEA (2010), • General Safety Requirements (GSR) Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA (2014) • General Safety Requirements (GSR) Part 6, Decommissioning of Facilities, IAEA (2014)	• http://www-pub.iaea.org/MTCD/publications/PDF/Pub1465_web.pdf • http://www-pub.iaea.org/MTCD/publications/PDF/Pub1578_web-57265295.pdf • http://www-pub.iaea.org/MTCD/publications/PDF/Pub1652web-83896570.pdf • https://www.iaea.org/topics/nuclear-safety-conventions/joint-convention-safety-spent-fuel-management-and-safety-radioactive-waste

Organisation	Description	Main Documents	Link
(Organisation for Economic Co-operation and Development) OECD's Nuclear Energy Agency (NEA)	International guidance and regulatory methodologies.	• Nuclear safety technology and regulation • Radioactive waste management and decommissioning	• http://www.oecd-nea.org/nsd/ • http://www.oecd-nea.org/rwm/
Multinational Design Evaluation Programme (MDEP)	The MDEP was set up to share resources, knowledge and information accumulated by national nuclear regulatory authorities during their assessment of new reactor designs, to improve the efficiency and effectiveness of the process. OECD/ NEA is the secretariat for the MDEP.		• https://www.oecd-nea.org/mdep/
European Nuclear Installations Safety Standards Initiative (ENISS)	ENISS provides the nuclear industry with a platform to exchange information on national and European regulatory activities, to express views and provide expert input on all aspects related to international safety standards.	• https://www.eniss.eu/publications/	• https://www.eniss.eu/

Table 1.1-2: International Harmonisation in Safety Standards - Industry

Organisation	Description	Main Documents	Link
European Committee for Standardisation (CEN) / European Committee for Electrotechnical Standardisation (CENELEC)	CEN and CENELEC collaborate with the international standardisation organisations, ISO and IEC, on the development and publication of standards that ensure the safety, environmental and technical requirements of the European nuclear industry.	• https://www.cen.eu/work/Sectors/Energy/Pages/NuclearEnergy.aspx	• https://www.cen.eu/work/Sectors/Energy/Pages/default.aspx
World Association of Nuclear Operators (WANO)	WANO provides a platform for the exchange of operating experience, professional and technical development (including technical support, information and recommendations) amongst the NPPs.		• https://www.wano.info/
World Nuclear Association (WNA)	is the international organisation that represents the global nuclear industry. Its mission is to promote a wider understanding of nuclear energy among key international influencers by producing authoritative information, developing common industry positions, and contributing to the energy debate.	Working Groups are exclusive forums that convene regularly in order to allow members to share best practice, conduct analysis, and develop consolidated positions on economic, safety and environmental issues.	• https://www.world-nuclear.org/our-association/what-we-do/working-groups.aspx • https://www.world-nuclear.org/our-association/what-we-do/working-groups.aspx/#cordel • https://www.world-nuclear.org/our-association/what-we-do/working-groups.aspx/#decommission

1.2 Development for National regulatory guidance for Decommissioning

Recommendations and regulations for the decommissioning of nuclear facilities in the EU have been developed since the early ‘80s. At that time decommissioning issues formed the subject of a series of R&D programs of the European Community that mainly focused on the improvement and practical demonstration of techniques such as decontamination and dismantling [44], [45], [46]. Later the review of policies, decommissioning-specific regulatory standards and criteria were published in [47]. The following topics were identified as principles in 1991 [48]:

- Radiation protection and safety applicable to decommissioning;
- Decommissioning requirements to be considered during the design and operation of nuclear installations;
- Choice of a decommissioning strategy;
- Management of a project, organisation required, necessary research and other preparations;
- Maintenance of a nuclear installation in a safe condition over a long period;
- Selection of decommissioning techniques;
- Reducing worker exposure during decommissioning operations at nuclear installations;
- Exemption rules, recycling materials, free use of buildings and land.

The growing experience in decommissioning is aiding the development of specific standards, principles and regulations for decommissioning that can facilitate regulatory decisions being made more easily and faster. Some countries have a very comprehensive legal and regulatory decommissioning framework; however, this may still be incomplete for others. An essential requirement is that the national government develops a policy that specifies national roles and responsibilities and provides the basis for introducing the legal/regulatory decommissioning framework.

[44] Decommissioning of nuclear power plants. Proceedings of a conference held in Luxembourg, 22-24 May 1984. EUR 9474

[45] Decommissioning of nuclear installations. Proceedings of a conference held in Brussels, 24-27 October 1989. EUR 12690

[46] The Community's research and development programme on decommissioning of nuclear installations (1989-93). Annual progress report 1991. EUR 14498

[47] Policies, regulations and recommendations for the decommissioning of nuclear installations in the European Community. European Commission. Nuclear science and technology series. EUR 15355 - Luxembourg, 1994, ISBN 92-826-7733-8.

[48] Crégut A., Roger J. Inventory of information for the identification of guiding principles in decommissioning of nuclear installations. EUR 13642. Luxembourg: Euratom, 1991.

1.2.1 Description of National regulatory guidance

1.2.1.1 Preparatory activities

Chapter 5 covers preparatory activities in further detail.

The type of preparatory activities carried out depends on the stage of the decommissioning. Initially, the decommissioning plan is prepared to implement a selected decommissioning strategy. This decommissioning plan typically consists of the initial and final plans and changes following decommissioning activities.

The initial decommissioning plan should:

- consider major safety issues;
- provide evidence that decommissioning can be carried out safely using proven/developing techniques;
- include a generic study showing the feasibility of decommissioning;
- consider the environmental aspects of decommissioning, such as management of waste and radioactive effluents;
- provide a basis to assess the costs of the decommissioning work and the financing mechanisms.

The final decommissioning plan should:

- be consistent with the decommissioning strategy proposed for the facility;
- be consistent with the safety case for decommissioning;
- describe the decommissioning activities, including the timeframe, the end-state of the project, and the phases of work;
- describe the facilities, systems and equipment required;
- describe the organisational structure, skills and qualifications required;
- describe the management of residual material and waste in accordance with the national waste strategy;
- describe the program of the final radiation survey at the end-state.

The decommissioning plan is supported by an appropriate safety assessment covering the planned activities and abnormal events that may occur during decommissioning. The assessment considers occupational exposures and the potential releases of radioactive substances that could result in exposure to the public.

Preparatory activities also considers technical activities which are implemented during the permanent shutdown and decommissioning of the nuclear facility. This can include [49]:

- Surveying of radiological situation for confirmation of the data used in the planning of dismantling activities;

[49] Safety assessment for decommissioning. Safety Reports Series No. 77. Vienna, IAEA, 2013.

- Covering of the floor with protective foils to inhibit the floor contamination;
- Installation of local ventilation to suppress the aerosols from dismantling;
- Installation of scaffolding for dismantling activities;
- Installation of temporary connections for electricity and other media needed;
- Delineation of cuts on equipment;
- Transport of dismantling tools to the dismantling sector;
- Isolation/check of equipment from electrical connection or operating media;
- Preparation of dismantling tools for the work;
- Installation of protective tenting for suppress the spreading of aerosols;
- Preparation of the working group (WG) for the decommissioning work; and
- Preparation of containers for waste from dismantling.

Moreover, the regulatory requirements for funding to be allocated for completing the preparatory activities is also required.

1.2.1.2 <u>Dismantling</u>

Chapter 6 covers dismantling in further detail.

There are two main types of dismantling:

- immediate dismantling;
- deferred dismantling.

Immediate dismantling begins shortly after the permanent shutdown of the facility. Equipment, structures and parts of the facility which may contain radioactive contaminants are removed or decontaminated, to a level that permits the facility to be released for unrestricted use, or with restrictions imposed by the regulatory body. Immediate dismantling involves prompt and complete decommissioning, allowing for the removal or processing of all radioactive material from the facility to another new or existing licensed facility for either long-term storage or disposal.

Deferred dismantling takes place after the permanent shutdown of a facility. For nuclear installation, nuclear fuel is first removed. Part or all of a facility containing radioactive material is either processed or conditioned to permit storage. The facility is then maintained until it can subsequently be decontaminated and/or dismantled. Deferred dismantling can involve the early dismantling /processing and removal of some of the radioactive material, to permit preparatory steps for storage for the remaining sections of the facility.

Dismantling also covers the disassembly and demolition of the structures, systems and components (SSCs) of a facility during decommissioning. Dismantling may involve the deliberate destruction and removal of engineered SSCs that had fulfilled specified safety functions during operation of the facility (e.g. confinement, shielding, ventilation or cooling). If these safety functions are still required, they should be provided by suitable alternative means or SSCs (e.g. tents, temporary systems or structures, fire systems, electrical systems and/or administrative procedures) for the duration specified on the safety assessment. The procedures for changing the equipment providing a safety function during decommissioning should be justified and demonstrated in advance of its implementation.

There are many available dismantling techniques which can be utilised in decommissioning, each with their respective advantages and disadvantages. For example, where remote dismantling is necessary, owing to high radiation fields, thermal cutting methods allow the use of relatively simple holding mechanisms. However, these methods generate large quantities of radioactive aerosols requiring local ventilation with filtration systems; this results in the generation of secondary wastes. In contrast, mechanical cutting methods need robust and elaborate holding mechanisms, but these methods usually result in smaller quantities of secondary wastes. Basic cutting, dismantling and remote operating capabilities have been developed and used. Special tools and devices may be needed during dismantling. In such cases, these tools and devices should be tested in mock-up trials before use. The applicability of these techniques to the particular decommissioning project should be considered before application.

The selection of dismantling methods and techniques should consider:

- the types and characteristics (e.g. size, shape and accessibility) of materials, equipment and systems to be dismantled;
- the availability of proven equipment;
- the radiation hazards to the worker and the general public, e.g. level of activation and surface contamination, production of aerosols and dose rates;
- the environmental conditions of the workplace, e.g. temperature, humidity and atmosphere;
- the radioactive waste produced.

Each dismantling task should be analysed to determine the most effective and safe method for its performance. Some considerations are as follows:

- equipment should be simple to operate, decontaminate and maintain;
- effective methods for controlling airborne radionuclides should be implemented;
- there should be effective control of discharges to the environment;
- when underwater dismantling and cutting is used, provision should be made for water processing to ensure good visibility and assist in effluent treatment;
- the effect of each task on adjacent systems and structures and on other work in progress should be evaluated; and
- waste containers, handling systems and routes should be defined before the start of dismantling work.

The dismantling of nuclear facilities is usually associated with the generation of large volumes of radioactive waste which requires a comprehensive radioactive waste treatment and management program. Dismantling operations can include waste treatment and conditioning activities, to account for their need to process radioactive material for further storage, transport and disposal.

1.2.1.3 Clearance of structures and materials

Section 7.1 covers the clearance of materials in further detail.

The International Basic Safety Standards (BSS)[50] defines mechanisms for exclusion, exemption and clearance of materials. IAEA Safety Guide[51] advises national authorities and operating organisations on the application of the concepts of exclusion, exemption and clearance. Clearance is defined as the removal of radioactive materials or radioactive objects within authorised practices from any further regulatory control by the regulatory body. The general criteria for clearance are:

- Radiation risks arising from the cleared material are sufficiently low as not to warrant regulatory control, and there is no appreciable likelihood of occurrence for scenarios that could lead to a failure to meet the general criterion for clearance; or
- Continued regulatory control of the material would yield no net benefit, in that no reasonable control measures would achieve a worthwhile return in terms of reduction of individual doses or reduction of health risks.

The radioactivity of a decommissioning material may have one or both of the following origins:

- Neutron activation. The material's origin is usually a reactor and comprises the bulk of the total radioactivity inventory of the decommissioning materials.
- Surface contamination. Radionuclides deposited onto the surface of components, which can occur on all types of nuclear installations. Components that are only contaminated can often be decontaminated to a level making the release of the component possible. During decontamination the radioactivity is transferred to secondary waste. Nevertheless, the total volume of radioactive material may be substantially reduced.

In general, the radioactivity of the bulk of the solid materials arising from decommissioning is low and/or can be reduced to such levels by decontamination, that in releasing the material there is minimal radiation exposure to the public. Currently, such releases are authorised by the regulatory body, applying the general regulations for radiological protection. Criteria for clearance of decommissioning materials (specific activity, surface contamination) are established in various countries that apply for unconditional and restricted release, i.e. for reuse, recycling and disposal. Concrete structures and ferrous scrap are typical materials that meet the criteria for clearance. In addition to the specific activity of surface contamination, the measurement procedures (for instance, mass or surface area over which measurements are averaged, methods for taking account of difficult-to-measure radionuclides) and a mass limitation can be specified.

The clearance of material is key to the generation and pre-treatment of radioactive material, i.e. collecting and segregating material that qualifies as radioactive waste or decontaminating large volumes of waste in the decommissioning of nuclear facilities. For example, radioactive material can be stored over an extended period (decay storage) to reduce the radioactivity of material, and hence reducing the volume of waste that will later require treatment. Furthermore, processing of waste, in particular, in the extraction and concentration of radionuclides, certain waste streams may be produced that do not qualify as radioactive waste and which can be disposed of conventionally.

[50] Radiation protection and safety of radiation sources: International basic safety standards. Safety Standards Series, GSR Part 3. Vienna, IAEA, 2014.
[51] Application of the concepts of exclusion, exemption and clearance. Safety guide, RS-G-1.7. Vienna, IAEA, 2004.

1.2.1.4 Final site release

Final site release is also discussed in Sections 7.6 and 7.7.

The release of the site for another use after the removal of all the significant amounts of radioactive materials is the ultimate objective of the decommissioning activities. This could be the site's unrestricted release from regulatory control or its restricted release, administered through a form of institutional control. For restricted use, the type, extent and duration of the restrictions and controls can range from monitoring and surveillance to restricted access of the site. The restrictions are proposed based on a graded approach and with consideration given to other factors (for example, the type and level of residual contamination after the completion of the remediation, the relevant dose constraints and release criteria, and the human and financial resources needed to implement the restrictions and controls).

Guidance to the regulatory body and operators on the release of sites or areas from regulatory control is presented in [52]. The regulatory body should establish safety requirements and guidelines for the release of land, buildings and structures from regulatory control. For the evaluation of potential radiological consequences associated with the site after its release, all relevant exposure pathways should be considered. It is necessary to use dose assessment involving direct radiation, inhalation and ingestion pathways to derive the release criteria (Bq/g or Bq/cm^2). Two main approaches can be considered: either the regulatory body may develop generic release criteria for use by the operator, or the operator can derive site-specific release criteria based on the radiation protection aspects, for the approval of the regulatory body. The flow chart of the release of sites from regulatory control is presented below (extracted from [53]).

[52] Release of sites from regulatory control on termination of practices. Safety guide, WS-G-5.1. Vienna, IAEA, 2006.

[53] Release of sites from regulatory control on termination of practices. Safety guide, WS-G-5.1. Vienna, IAEA, 2006.

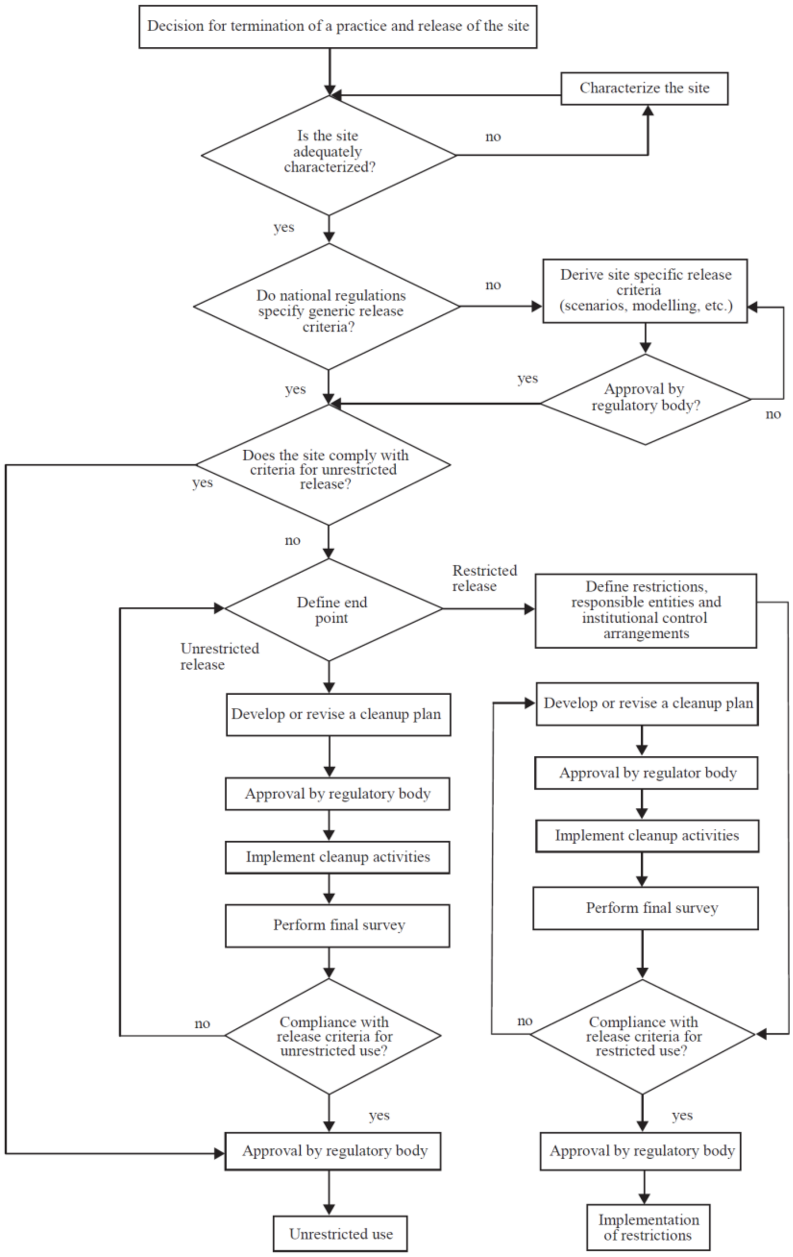


Figure 1.2-1 The flow chart of the release of sites from regulatory control

1.2.2 Experiences/Case Studies

1.2.2.1 Guidance in France

In France, once a basic nuclear installation (BNI) is shut down, it must go through the decommissioning and remediation process, with the end goal of achieving a predetermined final state in which all the hazardous substances have been removed. Accomplishing these decommissioning operations can often present a technical and project management challenge to licensees.

ASN issued 3 guides[54] to provide recommendations for operators of BNIs for the decommissioning and site remediation practices that it considers satisfactory from a regulation perspective. These guides may apply to sites that are in current operation, that are undergoing partial decommissioning, or performing remediation activities at a specific location (e.g. soil remediation):

- Guide 6: Final shutdown, decommissioning and delicensing of BNIs in France. At the end of their operation cycle, BNIs are shut down and undergo a decommissioning process, with the end objective being to delicense the site and release the land for other activities. The term decommissioning generally covers all of the activities carried out after the shutdown that are performed to achieve a pre-defined end state. These activities include cleaning and dismantling of equipment and structures, remediation of the grounds and disposal of all waste. This guide outlines the decommissioning process.
- Guide 14: Post-Operational Clean-Out (POCO) methodologies acceptable in BNIs in France. All BNIs evolve throughout their operation. Areas and buildings change or are demolished, the licensee may be required to perform POCO activities to eliminate contamination. This guide lays out ASN's recommendations regarding the remediation methodology to be used by licensees.
- Guide 24: Management of soils contaminated by the activities of BNIs in France. This guide is intended for BNI operators at sites where soil contamination has been detected, which has led to the undertaking of a remediation or soil management procedure. The guide outlines the procedure for managing and cleaning contaminated soils, including classification, excavation, and disposal of the soil. The guide was developed in conjunction with IRSN and ASND to clarify and harmonise the guidance relating to soil remediation in documents issued by several organisations.

On 28th June 2006, a French law was passed on the sustainable management of radioactive materials and waste, which requires the Government to draw up a national plan on the management of radioactive materials and waste (PNGMDR) every three years[55]. The first set of plans were assigned to the parliament in March 2007. The Parliamentary Office for the Evaluation of Scientific and Technological Choices (OPECST) was instructed to assess this first PNGMDR, available at [56].

The purpose of the PNGMDR is to consider the existing management of radioactive materials and waste, to make an inventory of foreseeable needs for storage and waste disposal, to indicate the necessary capacities for such installations and the duration of storage and, for the radioactive wastes for which a definitive management way does not still exist, to determine the objectives to reach.

[54] http://www.french-nuclear-safety.fr/References/ASN-Guides-non-binding

[55] http://www.french-nuclear-safety.fr/Information/Publications/Others-ASN-reports/French-National-Plan-for-the-Management-of-Radioactive-Materials-and-Waste-for-2016-2018

[56] www.senat.fr/opecst/

In 2003, the Nuclear Safety Authority (ASN) prepared the first draft of PNGMDR, using the framework of a multi-party working group involving operators, administrations and NGOs. The PNGMDR was published during the second semester of 2005 for consultation.

Since the 28th June 2006, the draft of PNGMDR has been updated, in particular, to take into account articles 3 and 4 of the law prescribing objectives required to determine some management solutions for different categories of radioactive waste, including high and intermediate level wastes in the long term. The new draft of PNGMDR has been prepared under the care of ASN and the Ministry of Industry (Office of Energy and Raw materials - DGEMP) and presented during a meeting of the working group in October 2006.

The decree which prescribes actions connected to the plan was published on the 18th April 2008. In connection with DGEMP, ASN lead the multi-party working group which will be in charge in future to follow the progress report in the PNGMDR.

1.3 Methods and tools for nuclear safety

This chapter describe the methods and tools already implemented according to the reference documents [57], [58], [59], [60], [61].

The safety assessment process for decommissioning provides a basis on which the safety of workers and the public can be ensured through the evaluation of the consequences of potential hazards and the identification of the ways that they can be mitigated so that the associated residual risks are as low as reasonably achievable (ALARA). A general requirement in decommissioning is the development of a decommissioning plan that includes, or has associated with it, an evaluation of the potential radiological consequences to the public and workers during planned decommissioning activities and as a result of any credible accidents that might occur during such activities.

Safety assessments are required to support the decommissioning plan and, therefore, need to be incorporated into the decommissioning plan or be contained in supporting documents. For larger projects consisting of several phases, it is usual practice for the detailed safety assessments to be separated from, but complementary to, the decommissioning plan. The decommissioning plan for such projects may, however, contain an overall or preliminary safety assessment. Regardless of the way the safety assessment is documented, it should be performed in a systematic, logical, and transparent manner with clear start and endpoints for each phase, and a clear end state for the decommissioning project as a whole. There is rarely a single safety assessment for a decommissioning project, other than for the less complex projects (e.g. facilities using only sealed radioactive sources). For projects with several distinct phases, it is normal to produce a safety assessment for each phase as the project proceeds; this allows flexibility as experience is gained.

The primary purpose is to identify hazards during normal and potential accident conditions and then to identify engineered and administrative control measures to prevent, eliminate or mitigate the hazards and their consequences. As part of this process, it should be demonstrated that residual risks have been reduced to ALARA and to within nationally prescribed safety criteria. It is important to demonstrate to the regulatory body and other interested parties that the safety of the planned decommissioning activities follows regulatory criteria. Industrial and chemical hazards are generally more significant in decommissioning activities than radiological hazards.

---

[57] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Guide No. WS-G-5.2 - Safety Assessment for the Decommissioning of Facilities Using Radioactive Material

[58] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. WS-G-2.1, IAEA, Vienna (1999).

[59] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Medical, Industrial and Research Facilities, IAEA Safety Standards Series No. WS-G-2.2, IAEA, Vienna (1999).

[60] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. WS-G-2.4, IAEA, Vienna (2001)

[61] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Reports Series No. 77 Safety Assessment for Decommissioning

Owing to the complexity and variety of the activities during the decommissioning process, a graded approach is applied to the evaluation of safety during decommissioning, with technical resources being allocated in proportion to the risks presented by the planned decommissioning activities.

At each type of facility, there will be hazards related to the decommissioning activities and the potential for incidents and accidents. Weighting should be given to activities and events with a higher potential risk in the safety assessment. Risks without significant consequences should be identified so that less analytical effort is expended on them.

1.3.1 Level of detail for safety assessments and documentation

The level of effort to be expended is based on a consideration of consequence and likelihood. This is shown in Figure 1.3-1.

The risk classification system can be consequence-based, i.e. determined by an assessment of unmitigated dose. The safety assessment can then consider the mitigating effects of engineered and procedural safety measures.

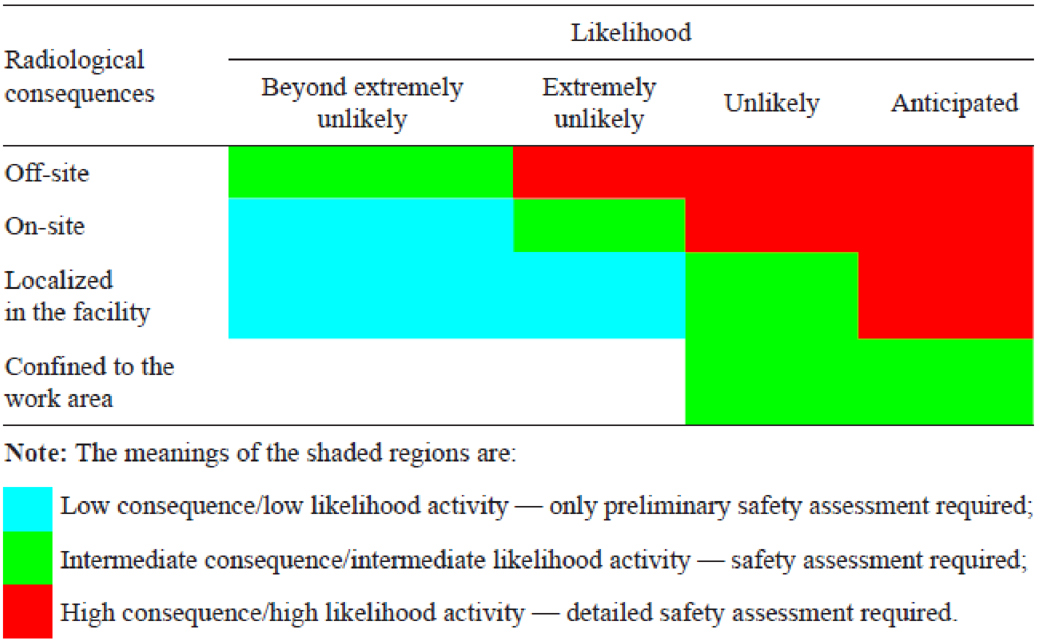


Figure 1.3-1 Level of detail for safety assessments based on consequences and likelihood

It is good practice to adopt a graded approach in assigning a safety category to decommissioning activities on facilities based on the highest risk class identified.

A preliminary safety assessment is used to assess unmitigated radiological consequences and a frequency band determined on a conservative basis. From the table, the appropriate risk class can be determined for each scenario. Facility classification can be allocated based on the highest risk class determined.

Accident scenarios can become complicated (due to there being several possible outcomes), hence it may be necessary to use detailed analysis to track and describe frequencies and illustrate dominant scenarios, though the need for this will be quite rare in decommissioning safety assessment. The frequency of the event should be taken into consideration. For example, the frequency of a vehicle accident may be anticipated, an accident that strikes and breaches material containers, and catastrophically ruptures the fuel tank and ignites would not be considered as a frequent accident.

The definitions and requirements in the safety assessment for each of the four risk classes are as follows:

(a) Risk class I events are those that could have a significant off-site consequence; therefore, the public must be protected with higher integrity engineered safety measures (Structures, Systems and Components - SSCs) and administrative safety measures (with engineered measures being preferred). Events resulting in high off-site radiological consequences must be subject to detailed safety assessment, irrespective of the assessed frequency of occurrence.

(b) Risk class II events are those that have lesser off-site consequences than risk class I, but significant on-site effects. Both classes I and II must also be considered for protection with high-level SSCs and administrative safety measures. The consideration of control(s) should be based on the effectiveness and feasibility of the considered measures. Further controls for class I and II accident sequences should be considered over and above the requirements of the accident safety criteria, if it is justified on ALARA grounds. This is sometimes described as defence in depth.

(c) Risk class III events are those with localised consequences. They are generally considered to be adequately protected by the operator's safety management programme. Class III accidents may be considered for defence in depth safety measures, if justified on ALARA grounds. A formal safety assessment would not normally be required unless requested by the regulatory body.

(d) Risk class IV events are those with low consequences and do not require additional safety measures but are considered to be adequately protected by the operator's safety management programme, and consequently, a documented safety assessment is not usually required.

It is common practice to classify a facility based on the highest risk class arising from the unmitigated accident safety assessment.

Risk class	Fundamental definition	Interpretation
I	Off-site hazard	≥5 mSv public off-site
II	On-site hazard	≥5 mSv on-site or ≥20 mSv in a building
III	Hazard in a building	≥0.02 mSv public off-site or ≥0.5 mSv on-site or ≥5 mSv in a building
IV	Hazard confined to local work area	<0.02 mSv public off-site or <0.5 mSv on-site or <5 mSv in a building

Figure 1.3-2 Risk classes

1.3.2 Safety Assessment Approach

Different approaches can be applied in safety assessment to estimate the potential radiological and non-radiological impacts of decommissioning activities on workers, the public and the environment:

—A **deterministic** approach, identifies the lines of defence against accidental activity release or exposure. The approach has been most commonly been applied in the safety assessment of a facility undergoing decommissioning. This approach focuses on the integrity and robustness of the lines of defence and provides a clear demonstration of the failure tolerance of the safety assessment.

—A **probabilistic** approach can be used to complement the deterministic assessment, but should not replace it, except for the application for accident sequences with a low consequence/occurrence frequency where the risk criteria are met without additional control measures being required. A probabilistic approach can be used as a tool to screen or eliminate accident scenarios for which the overall risk is shown to be acceptably low so that no further safety assessment is required. The unmitigated consequences (without control measures in place), together with an estimated frequency of occurrence, are compared to accident risk criteria to determine whether further analysis is required or not.

1.3.2.1 <u>Hazard Analysis: Identification and Screening</u>

One of the first steps in developing a safety assessment for decommissioning activities is the identification of existing and future hazards (both radiological and non-radiological) that can affect workers, members of the public, and the environment during decommissioning activities under normal and accident conditions.

Accident identification

Accident Identification is the determination of the possible initiating events during a decommissioning process. Accident data from the construction industry can be utilised in the absence of decommissioning data. The process analyses exposure paths and possible accidents during NPP decommissioning can be derived.

To identify the possible accidents, initiating events are analysed in the decommissioning of NPPs. In general, internal (for example, mechanical failures and human errors) and external events (for example, plane crashes and extreme weather) are considered to identify potential initiating events.

A **preliminary safety assessment** of hazards is useful to predict the bounds of potential consequences and to identify whether further detailed analysis is required. Once the hazards have been identified, a detailed evaluation of the hazards should be carried out. Low-risk accident scenarios do not usually require the additional evaluation, as at low-risk levels, the safety control measures introduced as part of the operator's safety management programme are generally sufficient to minimise risk.

A risk classification system can be used to determine the requirements for when further safety assessment is needed, circumstances where no further assessment is required, the level of control measures required, and the level of regulatory approval of the safety assessment.

Hazard evaluation

When the potential hazards and initiating events have been identified, and the bounding preliminary assessment of consequences and the frequency of occurrence has been carried out, the accident scenarios that present a higher risk category need to be evaluated.

Accident scenarios have to be developed for all initiators, however it is desirable to group them to reduce the number of individual scenarios analysed.

The radiological exposure of workers and the highest exposed group in the local population, using the 'critical group' concept, should be evaluated.

As a first step, accident initiators are sorted into several categories, such as:

(a) Operational accidents (e.g. initiated by plant failure, fire, operational error) within the facility;

(b) Human-made external events (initiated by activities outside the facility that may or may not be related to facility operations);

(c) Events initiated by natural phenomena.

These categories of events could be further subdivided, for example, operational accidents could be further broken-down into fires, spills, and explosions, and possibly subdivided into accidents inside containment and accidents in facilities without containment.

Once an accident scenario is selected for accident analysis, it is characterised and analysed to evaluate its consequences. This characterisation covers the amount of material, the physical form and the composition of the material, the physical surroundings that affect the material behaviour and release characteristics, and the initial set of assumptions used to perform the modelling.

The description of the accident scenario should include the following information:

- Accident type;
- Accident duration;
- Causes and activities;
- Preventive control measures;
- Termination of the accident;
- Mitigating control measures;
- Frequency;
- Consequences;
- Assumptions necessary to support the calculation of consequences.

The accident analysis should be as broad and as bounding as necessary to capture the applicable features and hazards of similar accidents. It is important to present the accident analysis in a realistic manner.

Hazards associated with individual decommissioning activities can be identified by using appropriate approaches and methods. Some of these are listed below:

- **Checklists**: The use of checklists can be a useful approach for identifying hazards and initiating events. For small facilities with few radioactive sources, a checklist can be a sufficient means for hazard identification. Checklists can also be useful in assessing individual decommissioning activities in a larger facility. This list can be used as a starting point for hazard identification, but care needs to be taken to add hazards that may be relevant for the particular facility. Since hazards may vary during decommissioning, the checklists have to be reviewed for each phase/stage of decommissioning.
- **Hazard and operability study (HAZOP)[62], [63]:** The HAZOP method is regularly used in the nuclear industry – it is a formal, systematic, and critical approach to identifying the qualitative potential of hazards and operating problems associated with an existing or new system, or piece of equipment caused by deviations from the design intent and their resulting consequential effects.

  HAZOPs can be used at varying times during the life cycle of the process.

  The procedure identifies:
  - Possible initiating events;

[62] CENTER FOR CHEMICAL PROCESS SAFETY, Guidelines for Hazard Evaluation Procedures, American Institute of Chemical Engineers, New York (1992).
[63] IEC 61882:2001 Hazard and operability studies (HAZOP studies) — Application guide

- Nature of accident consequences;
- Existing Engineered Safety Systems;
- Existing Operational Safety Systems;
- Requirement for additional safety systems: engineered or managerial;
- Operability or functionality issue

The HAZOP technique is usually a team effort to identify hazards, contributory causes and operability problems in plant and procedures. It is usually carried out by a team of 4–6 people, including a trained leader (with safety and reliability experience) and individuals involved in the design and the operation of the process to be studied. For example, during decommissioning, HAZOPs would review the cutting, lifting, cleaning and transport operations.

To carry out a systematic study, it is necessary to divide the plant into individual items, modification design, or procedure into operational steps (segmented items are referred to as nodes). Each node is given a unique identification number during the study.

Each node will be defined using keywords that are hazard-based or fault initiating events that should stimulate the identification of hazards. Examples of hazard-based standard keywords in the nuclear industry are:

- Fire / Explosion
- Radiation / Loss of Shielding
- Airborne / Surface Contamination
- Loss of Containment
- Wounding
- Impact / Dropped Loads
- Loss of Services – power, air, ventilation

Common hazards identified during decommissioning activities are:

- Spill of decontamination fluid
- Fire, spread of steam and aerosols from radioactive materials, solutions and chemicals
- Failure of ventilation system
- Flood of radioactive solutions
- Fall of radioactive piece or equipment
- Leak of liquid radioactive waste reservoir
- Collected activated or contaminated dust particles on dust extractor filters
- Airborne aerosols and gases released at the workplace

The HAZOP team identifies and records safeguards that are currently built into the design or form normal practice on the NPP. The safeguards will be separated into engineered (e.g. structural, containment, shielding, cladding, control and instrumentation etc.) and operational/managerial safeguards (e.g. procedures, training, supervision etc).

The HAZOP team's role also includes the identification of engineered safety features, safety-related equipment and safety management provisions for the plant or modification and perform the evaluation of results and identification of safety control measures.

After completion of the HAZOP study, a list of the identified potential initiating events will be generated, and their outcome will be summarised in the Fault Schedule.

The Fault Schedule provides an input to the accident analysis of the selected design option during the preparation of the Safety Assessment Report.

- **Failure Mode and Effect Analysis (FMEA)**[64]: FMEA is the process of reviewing components, assemblies, and subsystems to identify potential failure modes in a system and their causes and effects. For each component, the failure modes and their effects on the rest of the system are recorded in a specific FMEA worksheet. A FMEA can be a qualitative analysis that can rely on a quantitative result from mathematical failure rate models when they are combined with a statistical failure mode ratio database. It was one of the first highly structured, systematic techniques for failure analysis.

  FMEA is a "bottom-up" approach to the analysis that begins at the lowest level of design and works upward to consider system-level faults, assessing the cause and effect of each failure mode of each component of a system. The FMEA produces a detailed description of how failures of individual components influence system behaviour. The results obtained from the analysis are qualitative. As a minimum, the output from the analysis is a table listing individual components, the failure causes, failure modes, and the consequences of the individual failure modes.

  The procedure for performing an FMEA is outlined:

  1. Each component in your system is identified. Overlooking a component or one of its failure modes in a critical safety area may render the entire exercise useless, so it is necessary to be systematic and thorough.
  2. The functions of each part of the system performs are identified.
  3. List separately the failure modes for each of the functions. A failure mode is best described as a simple two-word statement of how the function may fail. Failure modes are described by stating what the product "does" or "does not do" when it fails.
  4. Identify the possible causes for the failure mode. The causes can be internal (e.g. mechanical defect) or external (e.g. failure of power supply) to the component.
  5. Describe the effects that each failure mode of that component would have. The effects should be described in enough detail that the severity of the effects can be judged.

[64] EN 60812: 2006 Analysis Techniques For System Reliability - Procedure For Failure Mode And Effects Analysis (FMEA)

The FMEA is a formalised, systematic practice of common engineering sense. It categorises and documents the considerations every good engineer always considers:

- What happens if this breaks?
- How can it break?
- What can I do to prevent it from breaking?
- What risk is involved?
- How shall I defend my design in light of established practices and the state of the art?

In the decommissioning process FMEA could be used to perform detailed analysis in order to demonstrate the reliability of a specific system, such as remote operating vehicles in a high radiation area, handling high level waste, or during retrieval operation.

- **Fault Tree Analysis (FTA)[65], [66]**: FTA is a logical analysis that organises reasoning from a general perspective to become more specific. The system is postulated to fail in a certain way, and branches of basic faults contributing to the undesired event are developed systematically. In summary, inductive methods are used to determine the possible system states, and deductive methods are used to determine how a system state can occur, and where the system states of interest are usually failed states. Deductive reasoning is used to identify the primary or top event(s) to be evaluated as well as the contributory events that could cause the top event. For safety analysis, top events usually have a significant undesirable consequence. An undesired state of a system is specified as the top event, and the system is then analysed in the context of its environment and operation to find all credible ways in which the undesired event can occur.

  A fault tree is a graphical logic diagram that shows the cause and effect relationship between contributory factors and the top event of interest. A set of event symbols is used to depict the cause-effect relationships. Causative events are referred to as input events and are located before a logic gate. The resulting event is referred to as the output event and is located after the logic gate. This cause and effect relationship is carried from the top event down to the level of component failures or external events. The top event, which serves as the starting point for the whole analysis, corresponds to an unwanted result such as a deterioration of the performance characteristics of the system or a change in function causing a hazard to the environment. Each different top event requires a specific tree. A FTA that does not consider the right top events is useless.

  FTA was initially developed to determine quantitative probabilities of top events, but it is also useful for qualitative analysis because of the systematic reasoning that various contributing factors are developed in a qualitative approach. The efforts to determine, assign, and calculate probabilities are not always warranted. The lack of statistical data in some decommissioning

[65] IEC 61025:2006 Fault Tree Analysis
[66] NUREG-0492 Fault Tree Handbook

systems can mean only a qualitative analysis can be performed. The goal of the qualitative analysis is to uncover all the root causes (contributory events) of the top events under consideration.

Selection of the top event(s) is the first step in the construction of a fault tree. For example, for the analysis of a remote handling system of radioactive waste one top events could be considered: an event potentially causing the system being irretrievable.

The contributory events that could cause the top event are then drawn from the top event as branches. They are separated from the top event by logic gates. Contributory events are then subjected in turn to the same process as the top event. The bottom level of each completed branch, a primary event, is a component failure, an error or other initiating event.

The symbols used in fault trees can be grouped in the categories below.

Primary event Symbol

Primary events are events that have not been developed further. Probabilities of primary events must be provided if the fault tree is to be used to compute the probability of the top event. There are four types of primary events:

1. The Basic Event: The circle describes a basic initiating event. No further development is required. The appropriate limit of resolution has been reached.
2. The Undeveloped Event: The diamond describes a specific fault event that is not developed further either because the event is inconsequential or because relevant information regarding the event is not available.
3. The Conditioning Event: The ellipse is used to record any conditions or restrictions that apply to a logic gate. It is used primarily with the INHIBIT and PRIORITY AND gates.
4. The External Event: The house symbol is used to signify an external event that is normally expected to occur and thus is not of itself a fault

Intermediate Event Symbol

An intermediate event is a fault event occurring because of one or more antecedents' causes acting through logic gates. All intermediate events are symbolised by a rectangle.

There are two basic types of fault tree gates: the OR gate and the AND gate. Other gates are special cases of these two basic types.

The <u>OR Gate</u>: The OR gate is symbolised by a shield with a curved base; it is used to show that the output event occurs only if one or more of the input events occur. Any number of input events may lead into an OR gate.

The AND Gate: The AND Gate is symbolised by a shield with a flat base; it is used to show that the output fault occurs only if all the input faults occur. Any number of input events may lead into an AND gate.

The INHIBIT Gate: The INHIBIT gate, represented by a hexagon, is a special case of the AND gate. The output is caused by a single input, but some qualifying condition must be satisfied before the input can produce the output. This qualifying condition is a conditional input, which is identified within an ellipse connected to the INHIBIT gate. The output condition OCCUTS only if the input occurs under the condition specified by conditional input.

The EXCLUSIVE-OR Gate: The EXCLUSIVE-OR gate, represented in one of the two ways indicated above, is a special case of the OR gate in which the output event occurs only if exactly one of the input events occurs. The quantitative difference between the inclusive and exclusive OR gates is generally insignificant, so the distinction is not usually needed.

The PRIORITY-AND Gate: The PRIORITY-AND gate is a special case of the AND gate in which the output event occurs only if all input events occur in a specified ordered sequence.

Once a complete fault tree is constructed, it should indicate all the factors, events and inter-relationships leading to the top event. The fault tree can be used either quantitatively or qualitatively to indicate where corrective actions could be taken. Quantitatively, the fault tree can be broken down into cut sets to determine the probability of the top event occurring. A cut set is the minimum sequence leading to the top event. Boolean logic and algebra are used to calculate this probability. The probabilities of each cut set sequence must be known to determine the probability of the top event. Quantitative fault trees to determine probabilities of occurrence of top events are generally costly. Statistical data may not be available for all significant events.

Most of the benefits of fault trees derive from qualitative analysis. They can be used to ensure that single-point failures are not possible for critical systems and can provide an indication of the relative safety of a product or system.

Qualitatively, each bottom event can be evaluated to determine where corrective measures could be taken. Checking the number of AND gates present in the fault tree can provide a quick evaluation of relative reliability. The AND gates are indicative of the need for all input conditions to be present to cause the output condition. The probability of the output condition is the product of the probabilities of the input condition. Conversely, the presence of many OR gates may be indicative of a relatively unsafe design because only one input condition is needed to cause the output condition. The probability of the output condition is the sum of the probabilities of the input conditions for an OR gate.

In the Table 1.4-1 (Section 1.4.1) there is summary of the main features of the above mentioned approaches.

1.3.3 Experiences/Case Studies

1.3.3.1 Italian Case Study - HAZOP methodology

As part of the design of a liquid waste treatment system, SOGIN spa developed a detailed nuclear safety analysis using the HAZOP methodology. The analysis was related to the treatment process of radioactive liquids waste by in drum cementing process. Utilising the HAZOP methodology, the plant under study was divided into "nodes". This sub-division does not respond to any well-defined "rule", with the aim of restricting the analysis to a set of equipment and process lines small enough to prevent confusion and forgetfulness. For the present study, a single system/subsystem has been defined as a node that has a single "Unit Operation" within it according to the meaning applied in the process engineering. The limit or boundary of the node normally arises at a block valve on a line or a nozzle on a tank.

The HAZOP study focused on 35 nodes, where the "INTENTION" is also presented for each node, i.e. the description of the operations performed on the node itself.

An example of a node description is below.

1 From NPS to: 51A-FX124X, 51A-FX134, 51A-BL009. Included: Vessel BL001A/B, Valve system FX120A/B and FX123A/B, Transfer pumps CA-101A/B.
Notes: Transfer the liquid wastes from NPS or evaporation unit to the discharge of 51A-CA101A/B.

Drawings:
SL-CX-01328-Rev03_140-GD-A-08061
SL-CX-02288-Rev03_140-GD-A-08064
SL-CX-02290-Rev03_140-GD-A-08066

2 Alcalinization of liquid waste. From valve 51A-FX134X to valves: 51A-FX-132X, 54A-FX-123X. Including 51A-BL003X, 51A-BL004X, pumps 51A-CA102A/B, pumps 54A-CA103A/B for NaOH dosing.
Notes: Transfer the liquid waste from discharge of pumps 51A-CA101A/B to pumps discharge 51A-CA102A/B through the Alcalinization reactor (51A-BL-003X) and Dosing tank (51A-BL-004X)

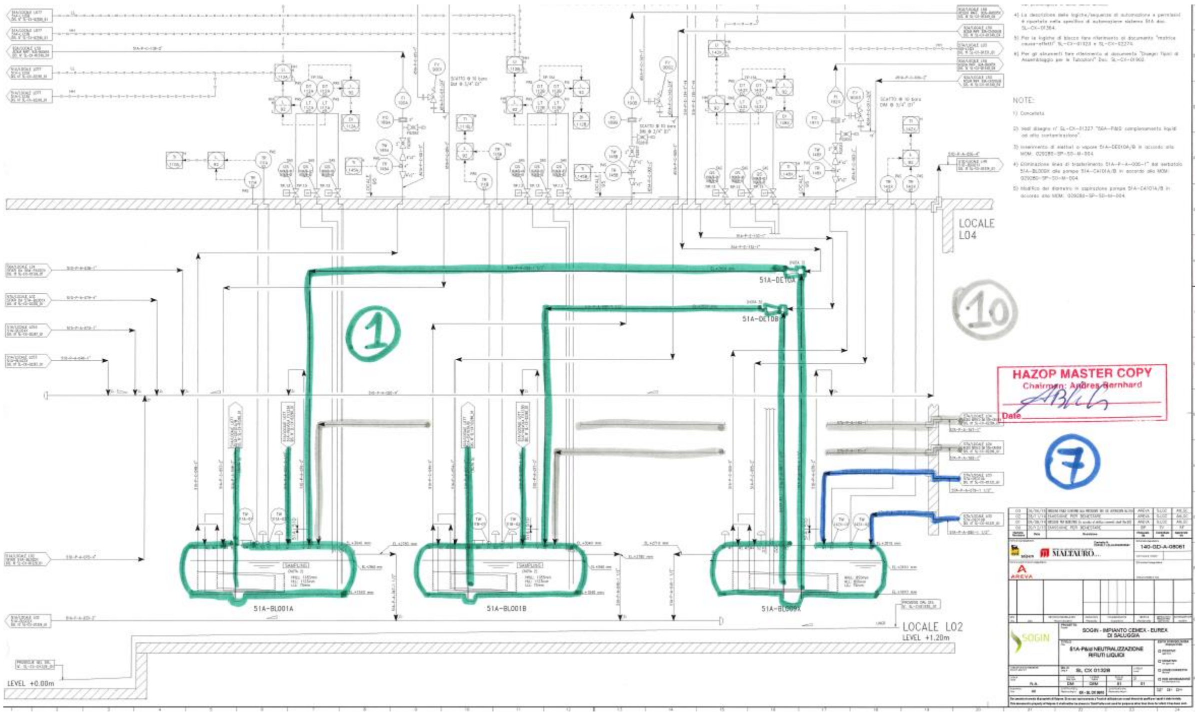
Drawings:
SL-CX-02287-Rev03_140-GD-A-08063
SL-CX-02288-Rev03_140-GD-A-08064
SL-CX-02289-Rev03_140-GD-A-08065

3 Cementation in 51C-DW005X. From valves: 51A-FX132X, 51A-FX106X, 51A-FX150X, 52A-FX133X to 51C-DW005X. Included: 51A-BL011X.
Notes: To form the cement drum by mixing waste, cement, water.

Drawings:
SL-CX-02289-Rev03_140-GD-A-08065
SL-CX-01333-Rev03_140-GD-A-08067 sheet1/2

…

Nodes are graphically defined by marking the P&ID used for the study, reproduced in the attached example.



The HAZOP study was carried out following the traditional multidisciplinary analysis methodology, internationally recognised, described in the publication "HAZOP - Guide to Best Practice" of EPSC (European Process Safety Center), IChemE and Chemical Industries Association, 2000 edition. Below an example of the analysis worksheets.

Worksheet

Company: Cliente (Sogin); Progettista (RTI Saipem-Maltauro) Page: 142 of 150
Facility: Impianto CEMEX

Session: (9) 30/07/2015
Node: (21) Process off-gas washing in scrubber 51D-BD001X/51D-BL002X. Wash of gas is by closed loop circulation of water maintained by pump 51D-CA101A/B. Make-up of demi water is made on scrubber bottom level controller.
Node Intention: Part of water can be withdrawn on the radiation dose indication in Room 45 from 51D-XI114X and sent by the pump 51D-CA102A/B to filtration through 51D-CA105A/B and ion exchange in 51D-FG010X and sent to drain collection system 67A-PT002X to prevent solid accumulation in circulating water. Threated off-gas is heated by hot air injection to prevent damage to HEPA filters downstream.
Drawings: SL-CX-01332-Rev03_140-GD-A-08007_sheet1/2
SL-CX-01334-Rev03_140-GD-A-08009_sheet1/4
SL-CX-01334-Rev03_140-GD-A-08009_sheet2/4
SL-CX-01334-Rev03_140-GD-A-08009_sheet3/4
Parameter: Level

GW	DEVIATION	CAUSES	CONSEQUENCES	SAFEGUARDS	RECOMMENDATIONS	BY
More	Higher Level in 51D-BL002X	21.22. 51D-LI112X malfunction not closing valve 51D-FX110X on demi water make-up line when reaching high level	21.22.1. Potential flooding of 51D-BD001X resulting in loss of scrubbing efficiency and potential carry-over of liquid to HEPA filters downstream (Node 20)	51D-LI112X signal from 51D-LT112X-01/02 in 1oo2 voting logic		
		21.23. 51D-FX110X failure in open position	21.23.1. See above	Pump 51D-CA102A started by 51D-LI112X H signal	40. FX110X to be specified Fail Closed	Saipem
Less	Lower Level in 51D-BL002X	21.24. 51D-LI112X malfunction not stopping pump 51C-CA102A on exhausted water withdrawal line when reaching low level	21.24.1. Loss of suction head to pump 51D-CA102A possibly resulting in pump 51D-CA102A damage	51D-LI112X signal from 51D-LT112X-01/02 in 1oo2 voting logic 51D-PAL113X/51D-PAL127X (red marked on P&ID master copy)		
			21.24.2. Loss of suction head to pump 51D-CA101A possibly resulting in loss of scrubbling water circulation and pump 51D-CA101A damage	51D-LI112X signal from 51D-LT112X-01/02 in 1oo2 voting logic 51D-FAL104X (red marked on P&ID master copy)		

After the issue of the HAZOP report, the recommendations resulting from the study will be included in a web-based system that will represent the basis for the follow-up. The implementation of the requested actions will be monitored by updating their status until all the recommendations have been implemented and fulfilled by Sogin.

1.3.3.2 **Italian Case Study – FMEA approach**

Another example of a nuclear safety analysis undertaken by Sogin includes the study to demonstrate the recoverability of a remote-controlled machine (called MRF) whose purpose was the recovery of radioactive drums stored in vaults (see picture below). Based on the level of MRF design development, a Functional FMEA was chosen in this type of analysis approach, each analysed component(s) represents a function. The analysis consists of examining the possible failure modes for each function and the effects that these failures have on the system.

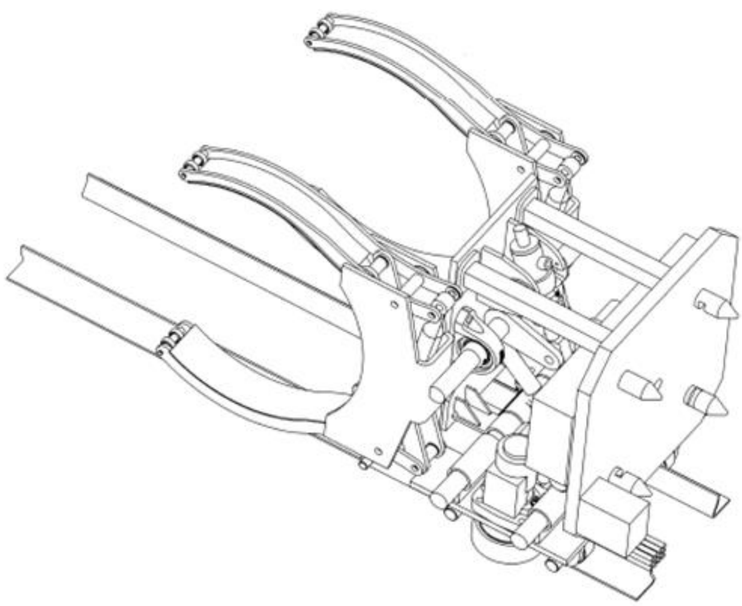


Figure 1.3-3 Remote operating Recovery drum machine (MRF)

The FMEA format used in the MRF fault analysis tables consists of the following columns:

- PART / ITEM: components that make up the MRF assembly, at any level (system, subsystem, elementary component). Where available, the numerical reference (tag) indicated in the elaborate CA FR 00050 is also indicated.
- FUCTION / DESCRIPTION: describes the function performed by the component.
- FAILURE MODE: describes how the component can fail
- FAILURE CAUSE: potential internal and / or external cause that can cause the component to fail
- EFFECT ON SYSTEM PERFORMANCE: direct consequence of the failure mode
- COMPENSATING PROVISION: Technical and / or organisational measures that can prevent and / or mitigate the consequences related to the failure mode
- EFFECT RETRIEVABILITY: final effect of the failure mode in relation to the recoverability of the MRF, of the possibility of terminating operations, of emergency or maintenance interventions connected with exposure to dose fields.

- REMARKS: further comments, notes or specific explanations that complete the reading of the fault mode.

SYSTEM : MRF						**Prepared By:** Sogin - ADS	
***SUB SYSTEM*:** Grab per Fusto						**Date:** 18/05/2016	
Operating Mode: Esercizio						**Revision:** 00	
PART/ITEM (TAG)	**FUNCTION/ DESCRIPTION**	**FAILURE MODE**	**POSSIBLE FAILURE CAUSES**	**EFFECT ON SYSTEM PERFORMANCE**	**COMPENSATING PROVISION**	**EFFECT ON RETRIEVABILITY**	**REMARKS**
16. Open center valve with four ways and three positions	Traction engine DX/SX	It does not open / does not close	Electrical failure on the control Mechanical valve failure	Grab motion stopped Impossibility of grab recovery	Redundant hydraulic circuit Overcenter valve that keeps the engine braked	No effect	
17. Fluid supply line from hydraulic power unit (pipes)	Hydraulic connection between the control unit and the traction motors of the garb	Low fluid pressure Fluid leak	Leakage	Impossibility of grab recovery	Redundant fluid supply line	No effect	

Figure 1.3-4 FMEA worksheet for radwaste remote handling machine

The analysis has shown that the individual equipment failures do not compromise the recoverability of the MRF inside the storage vaults.

Faults that require operator intervention, and that involve doses, are related to the failure of the piston due to locking. In the event of operator intervention, however, it was considered that the piston release operation is quite simple and can be carried out by personnel who are not highly qualified. The accessibility of the operator, with the MRF near the storage vault, will take place with suitable systems for working at height.

The dose fields allow the operator to access and perform the intervention at an ALARA value.

This failure mode in consideration of the environment in which it operates and the low work cycles is to be considered extremely remote, the literature data attribute a value of approximately 3 faults per million operating hours (λ = 3 FPMH), also considered that the expected operation is about 100 hours.

The UNI EN ISO 13849-1 standard relating to the safety principles of machinery provides the mechanical and/or hydraulic systems with the MTTF (Mean Time To Failure) value of a single component, for example a valve or a cylinder, estimated at 150 years (1 failure every 150 years) provided that the component has been designed using "basic" and "well-tested" safety principles.

In the design of the MRF, the safety principles indicated above and listed in EN ISO 13849-2, Tables C.1 and C.2, are respected, such as:

- use of the energy deactivation principle: the safe state is reached in the event of an energy interruption;
- safe position: the moving part of the component is kept in one of the possible positions by mechanical means (for example springs), to change the position it is necessary to apply force;
- sufficient positive overlap in the piston valves: the positive overlap guarantees the stop function and prevents inadmissible movements.

1.4 Methods and tools for conventional industrial safety

This chapter describes the methods and tools already implemented according to the reference documents [67], [68], [69], [70].

Industrial safety is being protected from physical danger as a result of workplace conditions. Industrial safety programmes in a nuclear context are the policies and protections put in place to ensure nuclear facility workers are protected from hazards that could cause injury or illness.

Work at nuclear facilities such as nuclear power plants, fuel fabrication facilities or waste processing and storage sites can subject workers to several industrial health and safety risks. These risks also characterise the decommissioning of nuclear facilities. Such facilities can contain hazardous processes and materials such as hot steam, harsh chemicals, electricity, pressurised fluids and mechanical hazards. Workers can be exposed to these and other hazards during normal duties (including slips, trips and falls, driving accidents and drowning).

Workers need to be protected by eliminating or reducing the radiological and non-radiological hazards that may arise during routine decommissioning and waste management activities and as well as during accidents. The non-radiological or conventional industrial hazards to which workers are subjected during the decommissioning and dismantling process may be greater than those experienced during the operational lifetime of the facility. These hazards could include:

- **Fire** - Fire is the conventional hazard that most frequently occurs in facility dismantling projects. The methods used for certain equipment dismantling operations (e.g. thermal cutting techniques) or for decontamination of surfaces (e.g. aggressive decontaminating solutions, etc.) are often the cause of localised fires. Moreover, while dismantling activities are in progress, the temporary accumulation of combustible materials and waste (plastic, cotton, etc.) is common, thus increasing the potential for fires in the area. Fortunately, such fires can be promptly detected and extinguished by appropriate fire protection measures and are generally of little importance. Although it has been assumed in this study that spent fuel is completely removed from the facility being decommissioned it is worth noting here, because of its possible consequences, that, where spent fuel elements remain in pools, rapid oxidation of zirconium in fuel cladding may be started if it is exposed to high temperature in water steam and/or oxidising atmosphere. Fire hazards during decommissioning activities must therefore be examined thoroughly, specifically the techniques and reagents to be used, the conditions under which the activities will be carried out, and the arrangements for storage of materials that will be generated in the operation. Fire protection measures should then be determined based on this analysis.

[67] Industrial Safety Guidelines for Nuclear Facilities, IAEA Nuclear Energy Series No. NP-T-3.3, Vienna (2018)
[68] Safety Assessment for Decommissioning, IAEA Safety Reports Series No. 77, Vienna (2013)
[69] Achieving the Goals of the Decommissioning Safety Case, A Status Report Prepared on Behalf of the WPDD by its Task Group on the Decommissioning Safety Case, OECD-NEA, Paris (2005)
[70] A 5 Step Guide for Employers, Workers and their Representatives on Conducting Workplace Risk Assessments, ILO, Geneva (2014)

- **Explosion -** In addition to normal fires, explosions may occur during decontamination and dismantling as a result of the chemical reagents and equipment used, (e.g. decontaminating solutions, thermal cutting devices such as blowpipes fuelled by highly inflammable materials, etc.) Such explosions may even be caused by the reaction of such reagents with radioactive materials remaining in tanks or associated with equipment due for decontamination, thus creating both radiological and non-radiological hazards.

  Some materials generated in the process of dismantling a facility, such as inflammable dusts, may in certain circumstances acquire explosive characteristics. Also, at facilities where a considerable time has elapsed since shutdown and chemical reagents or liquid waste have been awaiting conditioning for lengthy time periods, there is a possibility of auto-concentration phenomena that may cause explosive conditions, and special care must be exercised in such circumstances.
- **Toxic and hazardous materials -** The dismantling of nuclear facilities sometimes reveals that they were built using materials that are now banned and whose removal requires special measures because of their toxic or hazardous properties of the building materials. It is common, for example, to find asbestos used in thermal insulation or fire barriers, lead in paint, counterweights and shielding, and polychlorobiphenyls (PCBs) in oils and electrical insulation. Furthermore, some of the materials used in the decommissioning process, such as decontaminating solutions may, in and of themselves, be toxic and hazardous. All require appropriate protective measures to be taken.

  Particular care should be taken when these non-radioactive hazardous/toxic materials are either chemically combined or contaminated with radioactive material. In these instances, operators may need to devise safety and disposal strategies that address both the radiological and non-radiological hazard. In some instances, albeit rarely, implementing normal safety procedures for one hazard may increase the potential for the other. Thus, careful analysis of the safety (and disposal) requirements for this mixed material should be performed by specialists familiar with the inherent hazards. Safety and disposal practices should be implemented only after this analysis has been performed and practices developed that address the hazards from both materials.
- **Electrical hazards -** The dismantling of electrical installations in an environment where live wiring may be present, and inadvertently cut, is a hazard that must be recognised and addressed effectively for decommissioning activities. For this reason, it may be prudent to use new, completely separate electrical systems and to disconnect the original ones.
- **Physical hazards -** The physical hazards typically associated with demolition activities, or with the construction and use of temporary facilities, are also important, (e.g. collapse of structures, falling of heavy objects, working at heights, etc.) and need to be addressed.
- **Liquid and gaseous effluents -** Some of the wastes generated during decommissioning of a nuclear facility may be different from that generated during operations. This is because some of the materials used in decommissioning and some of the activities involved (e.g. in cutting and decontamination) are different from those during the operating stage. The quantity of liquid effluents generated from decontamination operations may be larger during

decommissioning whereas the quantity of gaseous effluent generated from ventilation of work areas usually decreases. The waste management arrangements for decommissioning activities should be designed to minimise the volume and radioactive content of such discharges into the environment, with appropriate abatement systems being provided according to the chemical and radioactive characteristics of the particular waste stream.

1.4.1 Description of the methodologies

The methodologies described in this chapter are referred to [71], [72], [73], [74], [75].

Safety assessment is directed primarily at analysing those pathways and event sequences that have the potential for causing significant radiological and non-radiological harms to the public or workers on-site. The assessment of these event sequences and the engineering and procedural controls that may be put into place to mitigate their impacts are then documented in the safety assessment as part of the overall set of safety arguments.

Thus, in the cycle of life of a nuclear facility, an integrated assessment of radiological and industrial hazards is needed. In particular, many legacy sites contain old chemical processing plants, and these can represent a significant source of risk during POCO and decommissioning. Dangerous chemicals may also be used for decontamination purposes. The most significant risk to workers on decommissioning sites will normally arise from the industrial hazards that exist on sites where building and demolition work is taking place. These hazards must also be considered in the safety assessment.

A key requirement of the safety measures or safety management programme is for the hazards associated with planned tasks to be assessed during the development of procedures and task-specific instructions, both for routine tasks within a decommissioning project and for tasks performed once only, to identify any necessary controls. This may be achieved by describing the scope of the planned tasks. This description is then used to perform a hazard assessment to identify potential hazards. Finally, the control measures necessary to reduce the risk from the identified hazards to an acceptable level are determined.

It is important to recognise that the safety control measures arising from a facility's safety assessment and those that arise from the assessment of the industrial hazards present during the execution of decommissioning tasks are complementary.

The controls arising from the task level safety assessment are designed to ensure that decommissioning activities can be conducted safely. Controls such as respiratory protection, the use

[71] HEALTH AND SAFETY LABORATORY, Review of Hazard Identification Techniques, HSL/2005/58, Crown, London (2000)
[72] HARMS-RINGDAHL, L., Guide to Safety Analysis for Accident Prevention, IRS Riskhantering AB, Stockholm (2013)
[73] HAMMER, W., Handbook of System and Product Safety, Prentice Hall, Upper Saddle River, NJ (1972)
[74] TAYLOR, J.R., Risk Analysis for Process Plant, Pipelines and Transport, E&FN Spon, London (1994)
[75] AVEN, T., Risk Analysis: Assessing Uncertainties Beyond Expected Values and Probabilities, John Wiley & Sons, Chichester (2008)

of safety harnesses, the isolation of live systems and PPE are typically specified. Many parts of an operator's safety management programme are designed to implement health and safety legislation on matters such as lifting integrity, working with hazardous chemicals and working at heights.

Where chemical or other hazardous substances may represent a significant hazard to workers or the public, there may be national legal requirements for their control. An example is the United Kingdom's regulations on the Control of Substances Hazardous to Health (COSHH) Regulations.

Industrial safety hazards are best addressed using a hierarchy of prevention and protection, which is often referred to as the five safety basics:

(1) Identify the hazards and monitor them for change;

(2) Eliminate the hazards whenever practical;

(3) Control the hazards when they cannot be eliminated;

(4) Protect the workers by providing and using PPE;

(5) Minimise the severity of an injury if an accident occurs.

A risk assessment is a thorough examination of a workplace to identify objects, situations and processes that may cause harm. After identification is complete, organisations evaluate how likely and severe the risks are, and then decide on what measures should be in place to prevent or control effectively the harm from happening.

Hazard identification techniques for systems and processes

See also Section 1.3.2.1.

In the following table, an overview of Hazard and Operability study (HAZOP), Fault Tree Analysis (FTA), Failure Modes and Effects Analysis (FMEA) and Task Analysis main features is reported.

Table 1.4-1 Summary of the methodologies for nuclear and conventional industrial safety

Methodology	What is working	What is missing	Assessment and possibility for improvement
HAZOP (Hazard and Operability) analysis	HAZOP analysis is extensively used in the chemical process industry. It performs a systematic search for deviations that may have harmful consequences. Each HAZOP element is defined in terms of its intention (what it is supposed to do), and potential deviations (ways of functioning that may lead to hazardous situations).	Application of HAZOP analysis to decommissioning requires a description of activities in terms of process	It could be useful the introduction of specific guide words for decommissioning process

Methodology	What is working	What is missing	Assessment and possibility for improvement
FMEA (Failure Modes and Effects Analysis)	FMEA has been in use since the 1950s and looks at the ways in which a component might fail and the effects and consequences that might arise. It is widely used in aerospace sector and manufacturing industry. FMEA is typically performed via the following main steps: 1) Aim, scope and assumptions are defined. 2) The system is divided up into different units, often components, but sometimes functions modelled in a block diagram. 3) Failure modes are identified for the various units, one by one. 4) Conceivable causes, consequences and frequencies of failure are estimated for each failure mode. 5) An investigation is made into how the failure can be detected. 6) An estimation of severity is made. 7) Recommendations for suitable control measures are made."	Application of FMEA to decommissioning requires a description of decommissioning activities in blocks and a systemic vision of interactions among each block to identify potential pathways of initiating events through safety barriers	A detailed analysis using FMEA can take extensive effort, and the amount of documentation can be large. A disadvantage of FMEA is that all components are analysed and documented, including failures with small consequences. It could be useful a preliminary screening of incidents related to decommissioning activities before applying the methodology
Fault Tree Analysis	A Fault Tree is a diagram showing logical combinations of causes of an accident or an undesired event, the top event. It can also be used to estimate the probability of the top event. It is extensively used in nuclear safety analysis, but can also be applied to industrial safety. A binary approach is adopted in that either an event occurs or it does not. Systems and events are modelled using Boolean logic (e.g. using 'and' and 'or' gates). Modelling is typically done by selecting the top event of interest, summing up known causes, constructing the fault tree, confirming its logic and assessing results.	Some disadvantages are that the method can be time consuming, requires expertise and training and may give the illusion of high accuracy when significant errors may be present	It could be useful to implement databases like, for example, the IAEA PRIS WEDAS module for NPP in decommissioning and to include industrial risks with failure probabilities related to decommissioning activities

Methodology	What is working	What is missing	Assessment and possibility for improvement
Task Analysis	Task Analysis covers a variety of human factor techniques. There are a large number of methods documented, roughly divided into action oriented and cognitive task analysis approaches. It is widely used in decommissioning project management and provides useful information about potential incidents due to technical failures or human errors	Task Analysis should be integrated with a Preliminary Hazard Analysis	Task Analysis could be integrated in more specific Safety Assessment Methodologies as HAZOP and FMEA

1.4.2 Experiences/Case Studies

1.4.2.1 Italian Experiences[76], [77]

Safety culture training programmes

In 2008, Sogin developed a training course at its Italian School for Radiation Protection, Safety and the Environment, focused on Safety Culture among workers and managers to improve knowledge about human and organisational factors related to conventional and nuclear safety. The course has continued annually, and the feedback from students has highlighted some interesting features that have changed the initial "classic" concept, where most of the time teachers explained the main topics and verified learning through questions and exercises, into a dynamic one, based on an interactive approach where teachers become coordinators of brainstorming sessions during which students, divided into workgroups, participate in simulations aimed at making players aware of their roles in improving the organisation's Safety Culture. The training course lasts two and a half days, with around 15 participants.

The teacher, becoming a group leader (co-ordinator), provides the students (players) with "cards", i.e. the concepts and the definitions that they will need in order to carry out simulations and role plays. Typical cards are:

- Hazards: in terms of physical, chemical, biological and organisational agents that represent the risk sources,
- Probability: expressed as confidence level according to subjective approach,

[76] RUSCONI C. "Knowledge management methodologies for improving safety culture" in Proceedings of the International Conference on Human and Organisational Aspects of Assuring Nuclear Safety – Exploring 30 Years of Safety Culture, IAEA, Vienna 22-26 February 2016
[77] RUSCONI C. "Training labs: a way for improving Safety Culture" in Transactions of the American Nuclear Society, Vol. 109, Washington, D.C. 10–14 November 2013

- Risk: expressed as a decisional variable, function of probability and damage related to adverse events,
- Risk scenario: expressed as a combination of hazards, targets and exposure paths,
- Safety: expressed as risk control, according to operational approach,
- Other cards as human and organisational factors, context etc.

After discussing former definitions (in particular, each worker's subjective perception of safety and risks), the more complex concept of Safety Culture is introduced from INSAG 4.

At this point, students observe a series of images representing usual and unusual workplaces. Each individual writes the sources of risk (hazards) they think or imagine could be found in the represented workplace. At the same time, they assign a risk index to each hazard to make a ranking list in terms of perceived likelihood and severity.

Subsequently, they are grouped in homogeneous groups (according to task(s) and/or worksite), where they perform the same evaluation together, merging individual results within each group to reach a common conclusion.

By comparing individual results with group conclusions, the influence of group pressure on individual observation emerges and, in a more general way, the influence of background and experience on risk perception becomes apparent.

Typically, technicians show a strong awareness of physical or chemical hazards (e.g. electrical devices or toxic substances) while office workers and managers are more sensitive to general and context hazards (e.g. natural hazards, falls, fire etc.).

This game allows players to gain awareness of the influence of group pressure on individual perception and of the importance of sharing knowledge and of developing communication skills to make colleagues aware of recognised hazards .

Sometimes groups may give less importance to valid individual perception of hazards than to more standard group observations, for example in the case of chemical hazards in car repairs and kitchens or natural hazards in outdoor workplaces.

Digitalisation of Industrial Safety Management

In previous years, the Sogin Department of Health and Safety carried out a global project of digitalisation of Industrial Safety Management through the exploitation of a suitable platform for safety compliance, risk analysis, PPE management, safety and maintenance of machinery and equipment, workplace health surveillance programmes. Specific software allows managers and workers to have a deep and updated knowledge on the main issues related to health and safety and to quickly identify the best technical and organisational measures to cope with potential risks deriving from decommissioning and waste management activities.

Managers and workers are strongly engaged in this process because the software modules are directly run by field operators with the supervision of Health and Safety experts.

Eu OSHA campaign

In 2018, Sogin joined the Healthy Workplaces' Campaign 2018-2019 'Healthy Workplaces Manage Dangerous Substances' organised by the European Agency for Safety and Health at Work.

The Sogin Department of Health and Safety organised a workshop on the management of dangerous substances in NPP decommissioning to share with public and private stakeholders (the National Institute for Prevention and Insurance against Workplace Injuries (INAIL), Professional Association of Engineers, Manufacturers' Association, Public Health Department, Italian Fire Department) knowledge and best practices- about topics as "Human factor and chemical risk: from reliability models to safety culture" and "Chemical and radiochemical aspects of Latina NPP decommissioning".

Operational Experience Feedback System

With the aim to strengthen interaction between nuclear and industrial safety assessment methodologies and to carry on the continuous improvement of Safety Culture, Sogin Department of Health and Safety is implementing an Operating Experience Feedback system capable of identifying and analysing warnings provided by workers during decommissioning and waste management activities. This system is constituted by an input module where workers can insert their observations with the aid of Windows menu containing an exhaustive check-list of near-miss and workplace anomalies and by the application of Root Cause Analysis techniques (e.g. 5 Whys, Ishikawa Diagram etc.) on behalf of a team of experts.

The outcomes of the analysis can be found in a report which is shared with managers and workers to carry out identified actions to improve safety standards.

1.4.2.2 UK Experiences

The exchange of learning in terms of safety with other industries (e.g. decom in oil & gas, chem. plants) is strengthening. A good example is a partnership between the Oil and Gas Tech Centre (Aberdeen) and the Nuclear Decommissioning Authority (NDA) in the UK. [78]

[78] https://www.totaldecom.com/cross-industry-learnings-report/

1.5 Development of radiological protection approaches and guidance for Decommissioning

As stated by the IAEA, '*Adequate planning for decommissioning and implementation of decommissioning actions are required to ensure the protection of workers, the public and the environment. […] Exposure during decommissioning shall be considered to be planned exposure situation and the relevant requirements of the Basic Safety Standards (BSS) shall be applied accordingly during decommissioning*'[79]. Occupational radiation protection is a key aspect of a decommissioning project and radiological protection standards and regulations apply during decommissioning as for operation.

Workers involved in the decommissioning of nuclear facilities are usually occupationally exposed workers. Their external, as well as internal exposures, are managed according to national regulations which are based, in the European Union, on BSS provided in Directive 2013/59/Euratom[80]. These standards reinforced the 3 principles of the radiological protection system which are defined in ICRP Publication 103[81] as follow:

- '*The principle of justification: any decision that alters the radiation exposure situation should do more good than harm.*
- *The principle of optimisation of protection: the likelihood of incurring exposures, the number of people exposed, and the magnitude of their individual doses should all be kept as low as reasonably achievable, taking into account economic and societal factors.*
- *The principle of application of dose limits: the total dose to any individual from regulated sources in planned exposure situations other than medical exposure of patients should not exceed the appropriate limits recommended by the Commission*'.

While health physicists usually focus in nuclear facilities on radiological protection, it appears clearly, especially during decommissioning which is most of the time characterised by a highly complex and evolving work environment, that industrial safety is also of high importance (see Section 1.4) and that a global approach for occupational risks management is required ('*The hazards associated with facilities might include chemical, biological and industrial hazards, in addition to radiological hazards, and consideration should be given to achieving a balanced approach to addressing all hazards*')[82].

---

[79] Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities, Specific Safety Guide No. SSG-47, International Atomic Energy Agency.

[80] Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom.

[81] ICRP, 2007. The 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Ann. ICRP 37 (2-4).

[82] Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities, Specific Safety Guide No. SSG-47, International Atomic Energy Agency.

1.5.1 Radiological Protection and decommissioning of nuclear facilities: current practices

Characterisation of the working environment

Radiological protection first relies on a detailed knowledge of actual working conditions which include dose rate (external irradiation) at each workstation, and the fixed and non-fixed level of contamination (internal contamination). This information will allow the assessment, prior to any task in radiological controlled areas, of occupational exposures according to the decommissioning plan. radiological measures can then be implemented to allow for the optimisation of external exposure and prevention of internal and/or external contamination. There are a number of options to consider regarding the radiological measures:

- Workers information and training,
- RP technicians training,
- Biological shielding,
- Chemical system or full system decontamination,
- Remote monitoring,
- Robotic tools,
- Contamination fixatives,
- Ventilation.

Dealing with and working in uncertain conditions

While a detailed knowledge of dose rates and contamination levels is easily accessible during operation (repetition of tasks and jobs, regulatory mapping, feedback experience from decades of operation, etc.), decommissioning may give rise to several technical challenges (outlining the importance of record keeping during operation in order to facilitate decommissioning[83]).

For instance, pipes (or component) cutting may lead to the resuspension of contamination which is fixed on the internal face of the pipes, especially while using hot cutting techniques. Detailed knowledge of this contamination is required to evaluate air contamination and to adapt air monitoring techniques, individual and collective protective equipment, individual monitoring, etc. Decommissioning will also require access to high and very high radiation areas which usually remain closed during operation. Such access will require knowledge on dose rates and (potentially) remote techniques and adequate workers training (based on virtual 3D techniques for instance) that will help to decrease exposure duration if human access is required.

This is highlight by IAEA[84] as follow: '*Although the principles and aims of radiation protection during operation and during decommissioning are fundamentally the same, the methods and procedures for implementing radiation protection may differ during decommissioning owing to differences in the*

[83] IAEA, Record Keeping for the Decommissioning of Nuclear Facilities Guidelines and Experience, TRS-411 (2003).
[84] IAEA Specific Safety Guide No. SSG-49 "Decommissioning of medical, industrial and research facilities" (2019)

physical conditions of the facility, the need for access to highly activated components or contaminated equipment or areas, and the removal of SSCs. During decommissioning, the principal focus of radiation protection is the protection of workers against occupational radiation exposure in planned exposure situations and emergency exposure situations. Special situations might need to be considered, which might require the use of temporary measures and specialised equipment and the implementation of certain non-routine procedures.'

Internal exposure and alpha emitters

For most nuclear facilities and especially NPP which have been sometimes shut down for decades, decommissioning activities will lead to an increase of the potential for internal exposure while most licensees, employers and regulators usually expect to prevent any internal exposure (i.e. no recordable dose). This often requires important changes in the way radiological protection is managed: staffing and organisation[85], workers and RP technicians training, on-site metrology, laboratory tools and staffing, individual respiratory protective equipment, etc.

^{60}Co and ^{137}Cs are easy to detect gamma emitters associated with 1.7 10^{-8} and 6.7 10^{-9} $Sv.Bq^{-1}$ inhalation dose factor. Associated derived air concentration (DAC)[86] are respectively 490 and 1 243 $Bq.m^{-3}$. For ^{241}Am, the inhalation dose factor is 2,7 10^{-5} $Sv.Bq^{-1}$, which equals to a 0,3 $Bq.m^{-3}$ DAC. Inhalation of 741 Bq of ^{241}Am is equal to a 20 mSv dose. This example[87] illustrates challenges associated with the decommissioning of nuclear facilities with alpha emitters contamination, which requires very low detection levels for difficult to measure radionuclides for adequate job planning as well as job monitoring.

Decision taking and graded approach

The optimisation of radiological protection (RP) must be demonstrated. RP staff decisions aiming at reducing collective radiation exposures and preventing internal exposure must be based on a number of parameters considering a complex and changing environment:

- Regulatory requirements,
- Dose saving,
- Dose transfer,
- Generation of conventional and radioactive waste,
- Impact on job duration,
- Costs,
- Nuclear safety,
- Industrial safety (see Section 1.4),
- Generation of liquid and gaseous effluents.

---

[85] Organisation to fight against workers internal alpha contamination in decommissioning works at Saint Laurent A, J. Laurent & al., ISOE Symposium, Uppsala (Sweden), June 2018.
[86] A worker breathing for 2 000 hours a 1 DAC air received a 20 mSv dose.
[87] EPRI Alpha Monitoring Guidelines for Operating Nuclear Power Stations, Technical Report (ID 3002000409), Aug 29, 2013.

Decommissioning projects show that decision and efforts for radiological protection must be commensurate with the level of risk, i.e. applying a graded approach is recommended (‘*A graded approach shall be applied in all aspects of decommissioning in determining the scope and level of detail for any particular facility, consistent with the magnitude of the possible radiation risk arising from the decommissioning*’[88]). Indeed, conservatisms leading to wrong resource allocation should be avoided, which emphasises the need for an in-depth knowledge of the facility characteristics.

Radiological protection and holistic approach for risk management

Decommissioning of nuclear facilities give rise to a number of industrial and radiological risks, which need to be managed according to national regulatory framework and characteristic of the facility. A balance is required in order to deal with a wide spectrum of risks for a given job for example: alpha emitters and asbestos[89] and PAH[90]. Regulations may require implementation of contradictory measures (e.g. shower in RCA for asbestos risk minimisation generating radioactive effluent).

1.5.2 Experiences/Case Studies

1.5.2.1 Connecticut Yankee, USA

The Connecticut Yankee (619 MWe Pressurised Water Reactor (PWR), 4 loops) experience provides valuable information regarding the importance of radiological protection program during decommissioning. ‘*Due to a contamination incident involving two workers that occurred shortly after permanent shutdown of the plant, the U.S. NRC placed Connecticut Yankee (CY) under a Confirmatory Action Letter (CAL). The workers received a significant internal exposure due to the inhalation of airborne contamination. Although not an overexposure, the received doses and the poor HP and Radworker practices greatly concerned the NRC. This letter restricted Connecticut Yankee from performing challenging radiological work until certain improvements in the Health Physics Program had been performed to the satisfaction of the NRC. Once the CAL was lifted, CY could proceed with major decommissioning activities. The CAL was issued by NRC on May 4, 1997 and lifted 14 months later*’. Inadequate radiological protection management may lead to large delays and costs of a decommissioning project.

Results of the CY decommissioning project show a 7.74 man.Sv collective radiation exposure (CRE) over the 1996-2006 period, which is quite comparable to CRE during operation[91]. The idea that external exposure during decommissioning is not comparable to external exposure during operation, leading sometimes to inadequate allocations of resources for RP, is incorrect.

[88] IAEA Safety Standards, Decommissioning of Facilities, General Safety Requirements Part 6. No. GSR Part 6. IAEA, 2014.
[89] Mixed alpha/asbestos risk management at EDF-DP2D, G. Ranchoux, ALARA in Decommissioning and Site Remediation, European ALARA Network Workshop, Marcoule, March 2019.
[90] Restricted clearance - PAH’s posing a challenge in Dismantling, S. Fleck, European ALARA Network Workshop, Marcoule, March 2019.
[91] Connecticut Yankee Decommissioning Experience Report, Detailed Experiences 1996-2006, EPRI Report 1013511, November 2006.

1.5.2.2 Experience in Germany

Based on German experience, GRS provided the following lessons learnt when comparing RP during operation and decommissioning[92]:

- '*Continuous change of the facility due to the decontamination and dismantling activities – systems may not be available anymore, the radiological inventory changes, work instructions requiring adaptations, radiation sources may appear and disappear again;*
- *Increased number of (long-lasting) work activities with interdependencies – high need for coordination of all activities to avoid radiological consequences;*
- *Access to workplaces not accessed during operation and outage – coping with unknown radiological situations;*
- *Need for new or improved cutting and dismantling tools to speed up the decommissioning activities – protective measures need adaptations;*
- *Occurrence of deviations between plans and real situation at the workplace, e.g. due to differences between blueprints and reality, unexpected radioactive material – risk of spontaneous changes of plans without analysis of (safety and radiological) consequences and adaptation of plans and measures;*
- *High volume of material flow, including flow of radioactive material and activated and contaminated components, through the nuclear facility – storage areas, capacities for handling and processing of radioactive material and to control material entering / leaving the radiation-controlled area gain much higher importance;*
- *Depending on the progress of dismantling activities replacement of technical barriers e.g. by administrative barriers – personnel protective equipment becomes more important and human error might have higher impact on safety and radiation protection;*
- *Long-lasting increased number of personnel during all the year in the radiation-controlled area – management of the personnel and equipment is more extensive*'.

The implementation of new techniques requires careful consideration of RP aspects. While laser cutting a big component (internals, vessels, etc.) provides valuable results in terms of effectiveness and efficiency, it also raises RP issues (resuspension of contamination, potential issues associated with water treatment and resins, etc.). Indeed, the LD Safe EC funded research project on laser cutting is paying attention to safety and RP aspects while developing standards for the use of laser technologies for dismantling.

[92] Radiation Protection during Decommissioning of Nuclear Facilities – Experiences and Challenges, J. Kaulard and B. Brendeback, IRPA Glasgow, 2015.

1.5.2.3 Deactivation and Decommissioning Knowledge Management Information Tool (D&D KM IT)

The D&D KM IT is a web-based knowledge management tool developed for the D&D community in collaboration, among others, with the US Department of Energy Office of Environmental Management and the former ALARA Centres at Hanford and Savannah. It was developed to prevent the loss of D&D knowledge gained over the years by DOE and contractors' employees by collecting, consolidating and sharing all valuable information.

Figure 1.5-1 D&D KM-IT website (www.dndkm.org)

As far as RP is concerned, the D&D KM IT provides several practical and technical advice and tools allowing for an improved approach of RP management for decommissioning activities. For instance, while fixative spray may be used to decrease radioactive contamination resuspension, the D&D KM-IT provides a list of fixative used in the nuclear industry to trap radioactive contamination and other hazardous materials which is a great resource for RP technicians [93]. The list provides the name of the product (and company), typical use and some notes.

[93] https://www.dndkm.org/DOEKMDocuments/General/265-fixative%20list%2017.pdf

2. Project Management and costing

For decades, nuclear projects were mainly aimed at building facilities such as NPPs, fuel cycle facilities, waste management storage facilities etc. However, nowadays, decommissioning has become an important part of the project, triggering new challenges in terms of project management and planning as well as costing. Based on its Member States growing experiences and feedback, the international decommissioning community have identified these domains as utmost importance when it comes to the achievement of decommissioning projects as well as effective decommissioning strategies.

In addition to supply chain management, project management is moreover strongly linked with sound cost estimates. It is therefore important to be aware of and use the best methodologies and tools for cost estimation. Similarly, stakeholders responsible for decommissioning projects must take advantage of new technologies in particular when it comes to digital transformation, how they can serve decommissioning projects and which tools are best to use.

Moreover, new technologies and communication techniques should also ensure successful information sharing with civil society to allow the public to be more aware of the challenges and progress of decommissioning projects.

Although it is understood that States have put in place their own decommissioning strategies based on national regulations (cf. supra), best practices should be more and more shared at the international level. International and regional organisations have several programs and initiatives to tackle these issues, share information and develop guidelines based on experiences.

International initiatives

IAEA Initiatives

- IAEA International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities.[94] A session of this workshop was related to contract management and organisation and discussed the different types of contract strategy adopted in the Member States represented at the workshop and explored how risks are managed.

- IAEA –IDN “International Decommissioning Network" was created in 2007, including “Cost estimation" as one of the thematic areas in the programme.

- IAEA-DACCORD collaborative project. Launched in 2012, the Data Analysis and Collection for Costing of Research Reactor Decommissioning (DACCORD) project provides representative input and benchmarking data required for the costing of research reactor decommissioning at preliminary planning stages.[95]

[94] IAEA International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities, Tsuruga, Japan, 2019

[95] Data Analysis and Collection for Costing of Research Reactor Decommissioning. Report of the DACCORD Collaborative Project”. IAEA TECDOC-1832. 2017

NEA Initiatives

- NEA WPDD-DCEG "Decommissioning Cost Estimation Group" was created in 2007 to foster the exchange of information and experience on issues in the cost estimation process.

- In the framework of the NEA CDLM, was created in June 2020 the Expert group on Costing for Decommissioning of Nuclear Installations and Legacy Management (EGCDL)

- International Structure for Decommissioning Costing (ISDC) for Nuclear Installations " jointly developed in 2012 by the NEA, IAEA and EC

European Commission Initiatives

- The EU-funded PLEIADES project (PLatform based on Emerging and Interoperable Applications for enhanced Decommissioning processes) - will develop a new methodology for improving dismantling and decommissioning (D&D) operations in Europe. Specifically, the project will demonstrate an innovative digitally enhanced approach for selected key tasks related to D&D in real life examples from projects in Europe. It will integrate cutting-edge digital support tools into a Building Information Modelling (BIM) technology-based platform, optimising and facilitating the D&D process and leading to coordinated actions. It will also be capable of retrieving and connecting data and calculating simulation results regarding scenario feasibility, waste estimation, radiation exposure, cost and duration [96].

- The EU-funded INN4OGRAPH project (INNOvative tools FOR dismantling of GRAPHite moderated nuclear reactors) will develop a set of tools and methods for dismantling power plant operations. Specifically, it will design 3D modelling of dismantling scenarios as well as measurement tools for mechanical and physical properties. The project's tools and methods will be put to the test at a full-scale graphite power plant demonstrator in Chinon, France, in 2022, facilitating their uptake and further development. The participation of all European graphite reactor operators in the consortium will launch an era of excellence in the graphite reactor decommissioning field[97].

Other Initiatives

- LiveDecom is a research project supported by the Research Council of Norway, which focuses on digitalising the decommissioning (closure) process of nuclear facilities. The project will combine project data and schedules with state-of-the-art visualisation and simulation technologies to provide the user with a comprehensive overview of, for example, the status of the processes and the effects of any changes to the schedule. Unlike conventional project management tools, this system will not only simulate the impact of scheduling changes on staffing, costs, and end dates but also include safety and risk constraints for both employees

[96] "PLatform based on Emerging and Interoperable Applications for enhanced Decommissioning processES | PLEIADES Project | H2020 | CORDIS | European Commission." https://cordis.europa.eu/project/id/899990 (accessed Oct. 16, 2020).
[97] "INNOvative tools FOR dismantling of GRAPHite moderated nuclear reactors | INNO4GRAPH Project | H2020 | CORDIS | European Commission." https://cordis.europa.eu/project/id/945273 (accessed Oct. 16, 2020).

and the environment. The digitalisation of the many parts of a process is made possible by the use of digital twin, sensor, database, visualisation, and risk simulation technologies.[98]

- EU BIM task group promotes the common use of BIM, as 'digital construction'. The group have developed a handbook for the introduction of BIM by the European public sector.

[98] LiveDecom - Digital support for integrated industrial management," IFE. https://ife.no/en/project/livedecom-digital-support-for-integrated-industrial-management/ (accessed Oct. 17, 2020).

2.1 Methodologies and software tools for comparison of alternative decommissioning strategies

A comprehensive assessment of alternative decommissioning strategies is the key step in a decommissioning process. A number of factors like national policies, regulatory framework, suitable decommissioning technologies/techniques, radioactive waste/spent fuel management systems, and health, safety, environmental, social, and financial impacts should be assessed when comparing alternative decommissioning strategies.

For instance, different approaches of size reduction of large components, creates different issues in terms of safety and economics, also the amount of dust and noxious gases produced in the process requires proper ventilation systems, while generation of secondary waste requires specific removal and storage procedures to be solved that has an influence on the overall decommissioning process.

Currently, there are methodologies (such as cost benefit analysis and multi-criteria decision analysis) or software tools that are well advanced and continue to be developed. Also, there are completed and ongoing international initiatives in EC Horizon Euratom Programme and IAEA.

2.1.1 Description of available solutions

DEMplus is a 3D simulation software for interventions in the nuclear field. The software uses technological components developed by CEA. It is a real decision-support tool which responds to current work challenges. It ensures a better safety and waste management, major cost and planning cuts, whilst using the ALARA approach. DEMplus allows collaborators, involved in the same project, to work together in producing reports and schedules. Sensitivity studies can easily be performed allowing an optimum scenario choice, according to the project criteria[99].

Digital Decommissioning is a new software and hardware package that uses digital technology to decommission nuclear facilities efficiently. It is supported by building information modelling (BIM), computer-aided simulation, and virtual reality (VR). The software develops, improves, verifies and visualises the design and process solutions by: detailing the technological processes of the equipment disassembling at the level of operation, obtaining reliable evaluations of generated amounts of radiation waste, generating up-to-date as-built documentation, organising the engineering and technical information about nuclear facilities systematically, manages the information around decommissioning projects and trains personnel[100].

HVRC VRdose is a real-time software tool for modelling and characterising nuclear environments, planning a sequence of activities in the modelled environment, optimising protection against radiation, and producing job plan reports with dose estimates. It offers the possibility to refine the radiological model to improve the accuracy of estimates and configure the dosimetric output provided. The software can also be used as an aid to produce post-work review reports, with real measurements included. Furthermore, the software provides support for presenting information to different types of users for briefing and decision-making, thus serving as an aid to communicate between stakeholders. To support the user in interpreting the results of calculations, the VRdose

[99] "DEMplus for nuclear - site officiel." http://www.orekasolutions.com/demplusfornuclear/demplusfornuclear_en.html (accessed Oct. 16, 2020).
[100] "Digital Decommissioning." http://www.neolant.com/dd/ (accessed Oct. 16, 2020).

Planner provides charts, graphs, and 3D radiation visualisation, updated immediately to reflect any changes to the modelled radiological condition, such as changing shielding materials, and human activities over time. The VRdose Briefer is a dedicated presentation tool for communicating scenarios prepared using the Planner[101].

The CEA created the Marcoule immersive room called PRESAGE, at the end of 2008 in order to validate maintenance or dismantling operations. It is a resource shared by all the CEA decommissioning projects. The PRESAGE room groups all the technologies enabling user immersion and interaction in a virtual environment. Three kinds of immersion are possible: visual, sound, and tactile. To complete VR studies on decommissioning projects, certain steps are necessary to build the VR simulation. To run such simulations, a new software, called iDROP, has been developed by the CEA. This software consists of several real-time modules: collision detection, robotics, virtual human, and dose rate calculation. It allows a scenario global approach, considering all the aspects of a decommissioning project. iDROP takes 3D models, remote handling models and radiological data as input.

N-Visage® Fusion is a unique radiation modelling system for characterising nuclear facilities. The driving concept is data fusion; by combining multiple measurements into a single model. N-Visage makes data easier to navigate and reveals new insights[102].

Some already available software solutions for cost estimation, also has some capabilities for comparison of alternative decommissioning strategies (see Chapter 2.8).

2.1.2 Experiences/Case studies

The IAEA has provided examples of practical experience in reference [103].

[101] "HVRC VRdose," IFE. https://ife.no/en/Service/hvrc-vrdose/ (accessed Oct. 16, 2020).
[102] "N-Visage® Fusion," Createc. https://www.createc.co.uk/innovation/products-technologies/fusion-product-page/ (accessed Oct. 16, 2020).
[103] M Laraia D W Reisenweaver and International Atomic Energy Agency, Selection of decommissioning strategies issues and factors, report by an expert group Vienna International Atomic Energy Agency, 2005

2.2 Methodologies and software tools for project management and performance monitoring

Project Monitoring and the project manager's decision-making processes play a vital part in project management. However, it is a method often overlooked and only done for the sake of fulfilling the requirements of a project management plan. But if put into practice, project monitoring can help project managers and their teams foresee potential risks and obstacles that if left unaddressed, could derail the project. It clarifies the objectives of the project, links the activities to the objectives, sets the target, reports the progress to the management and keeps the management aware of the problems which crop up during the implementation of the project. It supports and motivates the management to complete the project within the budget and on time.

Project Monitoring refers to the process of keeping track of all project-related metrics including team performance and task duration, identifying potential problems and taking corrective actions necessary to ensure that the project is within scope, on budget and meets the specified deadlines. Simply put, project monitoring is overseeing all tasks and keeping an eye on project activities to make sure you're implementing the project as planned.

Usually handbooks for decommissioning concentrate solely on the health and safety aspects of management. As such regulation does not contribute to the question of how to best organize and control a decommissioning project [104]. On the other hand, it is considered that the preparation of a schedule is a well-developed process with capable software that is specifically designed for scheduling and resource loading management[105].

2.2.1 Description available solutions

There are several project management software systems and schedule systems on the market today. The following software is listed in alphabetic order.

Oracle's Primavera P6 EPPM is designed to manage projects of any size with robust, and easy-to-use, Primavera P6 EPPM in the solution for globally prioritising, planning, managing, and executing projects, programs, and portfolios[106].

Microsoft Project is a project management software product, developed and sold by Microsoft. It is designed to assist a project manager in developing a schedule, assigning resources to tasks, tracking progress, managing the budget, and analysing workloads. Microsoft Project and Microsoft Project Server are the cornerstones of the Microsoft Office enterprise project management (EPM) product [107].

[104] European Parliament, Nuclear Decommissioning: Management of Costs and Risks, 2013.
[105] OECD Nuclear Energy Agency, The Practice of Cost Estimation for Decommissioning of Nuclear Facilities, 2015.
[106] Oracle Help Center, Primavera P6 Enterprise Project Portfolio Management (P6 EPPM). https://docs.oracle.com/en/industries/construction-engineering/primavera-p6-project/index.html (accessed Oct. 16, 2020).
[107] Wikipedia, Microsoft Project. Jul. 27, 2020, Accessed: Oct. 16, 2020. [Online]. Available: https://en.wikipedia.org/w/index.php?title=Microsoft_Project&oldid=969763590.

Planisware Enterprise is the integrated solution that brings together budgets, forecasts, schedules, resources, and actuals[108].

A project management information system (PMIS) is used for the coherent organisation of the information required for an organisation to execute projects successfully. A PMIS is typically one or more software applications and a methodical process for collecting and using project information. These electronic systems help to plan, execute, and close project management goals. PMIS systems differ in scope, design and features depending upon an organisation's operational requirements [109].

Multiple software already available for cost estimation, have the in-built capability for additional project management tasks (see Section 2.8).

2.2.2 Experiences/Case studies

The best practices for project management are taken from European Parliament, Directorate-General for Internal Policies, Policy Department for Budgetary affairs[110].

2.2.2.1 France (EDF)

The long-term project vision is focused on the scheduling, reference costs, waste, and the technical reference scenario. It is prepared by the programme manager and by the project managers:

- A data book that is updated every three years and contains: strategic scheduling, important hypothesis, expenses, engineering and operation resources, waste production by project and sub project and spreading until the end of project.
- Risk and solution analysis.

The mid-term vision (5 years) plan includes key milestones and allocated resources:

- Global indicators allowing to control the projects evolution and data book adequacy: working and financial progress.
- Is consolidated by programme management.
- Performs a risk review.

The annual vision 'N+1' (Annual Achievement Contract) includes annual important steps and allocated resources:

- Project weekly meetings to coordinate short term operations,
- Detailed work scheduling update (site, weekly basis),
- Treatment of real-time issues as far as necessary (by useful means).

The Planisware (OGOPA) software was introduced as a necessary supporting element for project planning.

---

108 Planisware, Planisware Enterprise. https://www.planisware.com/enterprise/planisware-enterprise (accessed Oct. 16, 2020).

109 Wikipedia, Project management information system. May 15, 2020, Accessed: Oct. 16, 2020. [Online]. Available: https://en.wikipedia.org/w/index.php?title=Project_management_information_system&oldid=956816529.

110 European Parliament, Nuclear Decommissioning: Management of Costs and Risks, 2013.

2.2.2.2 Germany (EWN Greifswald)

There are currently no formulated requirements in Germany on how the organisation structure and the overall management of a decommissioning project should be designed.

From the very outset, for decommissioning of the EWN reactors, external managers were hired who have long experiences in the project management of large and complex construction projects. With this step, it was assured on the management level that favourable conditions for adapting to the very different task of decommissioning were in place. The typical methodology of project management, the tools and the very different planning and work approaches, as compared with the operational phase were thus introduced from the outset, providing a different general approach to planning and execution. This contrasts with other approaches where the adaptation of the organisation and its work approach was achieved after long periods of slow conversion.

Project management requires continued feedback of experiences that was acquired during task performance. The rationale behind this is that estimates made in the planning phase of tasks require validation. Validated estimates can be used in the planning of similar tasks, thus reducing risks and uncertainties. Such reflection of experiences is highly valuable knowledge for project management purposes.

Project management uses a set of IT planning tools and accounting methods - PMIS. Such a system provides up-to-date information on the status of all relevant decisions, properties, resources, etc. PMIS systems can be applied as an add-on, accompanying other systems, or as stand-alone systems that include all relevant aspects of a project. The PMIS at EWN is of the stand-alone type and was tailored to the specific needs of the decommissioning process. Issues such as 'Work package approval', covering the internal approval process, and 'Mass flow/Disposal' are two aspects that play a specific role in decommissioning projects and would not be found in a traditional or conventional construction project. Another aspect included in EWN’s PMIS is personnel development, relevant for decommissioning, which is not necessarily included in a standard PMIS for a construction project.

2.2.2.3 Slovakia (JAVYS)

In the performance and V1 NPP project management administration JAVYS uses standard software tools for communication and management. The main control system in the company is management program SAP and its sub-modules, including modules for financial planning and costs monitoring of decommissioning projects, including financial management and accounting.

The subject, time and financial recording of V1 NPP projects is implemented in software module SAP PS (Project System). The system provides data for comparison of planned and actual values for the individual V1 NPP projects.

The actual financial transactions are recorded in the company's accounts in SAP FI module (Financial Accounting) and fixed investment assets in accounting module in SAP FI-AA (Asset Accounting).

For planning activities, the software SAP ERP (Enterprise Resource Planning), Controlling module and SAP BPC (SAP Business Objects Planning and Consolidation) is used.

For various subtasks are used advanced extensions SAP BO BI (SAP Business Objects Business Intelligence), SAP BPC (SAP Business Objects Planning and Consolidation), SAP WEBI (Web Intelligence) and SAP CR (Crystal Reports), operating above the central data warehouse.

The main system for communication, progress control of activities, tracking and reporting of tasks, approval and monitoring of billing is IBM Lotus Notes.

The main program used for planning is MS Project, which is used for the planning of decommissioning projects (from the most general top-up tasks to the detailed particular steps for the project or groups of tasks). Meanwhile, the program ARSOZ is used for monitoring of physical and radiological state of material database of nuclear facilities, as well as for issuing of work orders in controlled areas and for work with contaminated materials.

The Oracle database is used for work orders, information administration and documentation. Additionally, MS Office programs are actively used for routine administrative and support work.

2.3 Tools for data collection in the field (e.g. for work monitoring)

The main reason and motivation for data collection is to have up-to-date knowledge of the actual situation on site.

2.3.1 Situational awareness based on field data and models

Producing a BIM model as a digital twin for a nuclear power plant and keeping it up to date even before the start of the decommissioning project in the inventory phase is important and is currently being utilised. The differences may come from the level of BIM maturity, i.e. the extent to which the model covers the entire operation of the plant. The different maturity levels are shown in the Figure 2.3-1 below BIM dimensions and maturity level distribution.

The situation at the plant changes continuously and rapidly during the construction and decommissioning phases compared to the plant's normal operating situation.

The changing snapshot includes things that are part of the plant's BIM model, but additional components, such as work machines, robots, and workers. These new entrants need to be able to work and move safely within the NPP area and therefore the picture needs to be broader than just the plant's BIM model. The multi-actor snapshot can be thought of as an extended real-time BIM model and IT can be combined with new information. Such new ever-changing information includes waste logistics such as driving routes, bypass and turning areas for waste disposal trucks, waste sorting areas, stacking areas, and the working space required by robots and other work machines.

The planners of the decommissioning and dismantling project take into account and make plans for what will happen to the plant itself at any time during the project and also for what happens to the logistics and related infrastructure at different stages of systematic waste management. Thus, after adding the time dimension, the 3D snapshot BIM model becomes a 4D model. Radioactivity/radioactive waste, which is dangerous to the environment, nature and people, brings its own factor to the decommissioning of nuclear power plants, but otherwise things are very similar to normal demolition projects for demanding construction sites, where asbestos, for example, requires special arrangements.

The 4D-BIM model snapshot can be used to monitor compliance with the schedule contained in the model and to ensure that work progresses accordingly. Keeping schedules in a multi-stakeholder project is a very important example one might think of congestion for a facility if one comes on schedule and the other is late and on track.

Keeping the situation up to date requires measurements and observation at the facility and for this purpose manpower or robots and drones can be used or measuring devices, cameras and sensors can also be attached to work machines that are otherwise moving or working in the area.

To compare the current situation of the plant against the 4D model and schedule, automatic image processing and artificial intelligence are needed. For example, a drone or robot goes every night to measure and shoot with 3D sensors, cameras, radiation measuring devices, a thermal camera, and so on, and updates the 4D model for the next day's use.

Augmented reality can be used to enrich a snapshot by bringing elements such as radiation levels or contamination to surfaces and observing in the field through a mobile app or tablet or a window looking at an object. Real-time realism is gained by adding the location of work machines, people moving in the area, etc. based on the data of the position sensors they carry to the video taken by the drone or robot at night and adding and displaying a 4D plan of what to do next. In virtual reality, the situation can be monitored at headquarters or in the design office. But it is worth considering what is the added value compared to a workstation program.

2.3.2 BIM dimensions and maturity levels

Building Information Modelling (BIM) is a process that involves creating and using an intelligent 3D model to inform and communicate project decisions. It has various dimensions:[111]

- 2D 2-Dimensional view
- 3D 3-Dimensional Model
- 4D + Time Schedule
- 5D + Budget (Cost)
- 6D + Facilities management (Maintenance)
- 7D + Sustainability (Life Cycle)
- 8D + Occupational safety and health

The BIM-maturity levels[112] are combinations of BIM dimensions (Figure 2.3-1)

- There is no exchange of information at the lowest level of maturity. Models are not shared and only 2D drawings, 3D models, and other documents are used.
- At the second level, there is already a partial exchange of information. Models are shared but not centralised. The 4D and 5D dimensions of the BIM model are added here at the maturity level.
- The third level of the four-level division already has a common shared model and the 6D dimension will be included.
- At the top we can talk about full integration. Issues of sustainable development and safety as well as well-being have been added to the model. 7D and 8D BIM models are included.

[111] https://cadblogbyamit.wordpress.com/2019/12/01/bim-ds-2d-3d-4d-5d-6d-7d-and-8d-and-benefits/
[112] https://www.united-bim.com/bim-maturity-levels-explained-level-0-1-2-3/

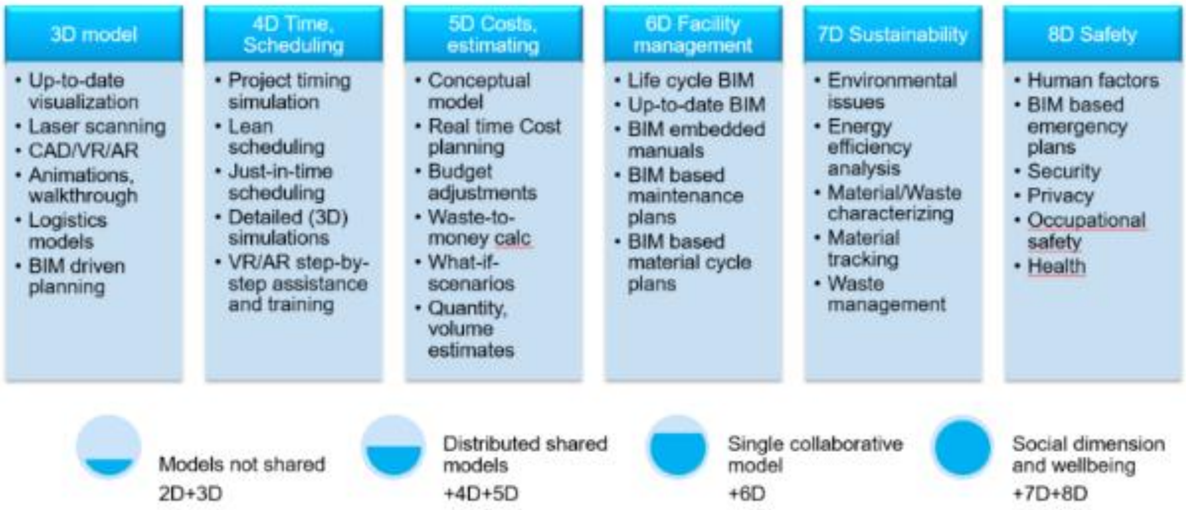


Figure 2.3-1 Combination of BIM dimensions

2.3.3 Safe work and motivated monitored workers

Worker measuring and modelling have raised many doubts within workers and also with management. A high quality of work will be important and demanded in the future: a more interesting working environment, greater autonomy and opportunities for self-development. A central element is human centricity considering workers with different skills, capabilities and preferences. New solutions will empower the workers and engage the work community. Empowering the worker is based on adapting the work to the skills, capabilities and needs of the worker and supporting the worker to understand and to develop his/her competence. Engaging the work community could be based on tools, with which the workers can participate in designing their work and training and share their knowledge with each other. To break the wall, it is important to provide early demonstrations of the ideas and to design them further with the workers in order to find acceptable and ethically sustainable ways for worker modelling. The workers would like to be more involved in the design and have possibilities to impact on their work. There are clear needs for knowledge sharing and adaptive learning solutions that would support personalised competence development and learning while working. An easily accessible platform for knowledge sharing could evolve to a forum where good work practices and ways to solve problems are shared not only within the work community, but also with other stakeholders. The virtual mode/digital twin/BIM could be utilised as platform for participatory design and training.

Augmented reality (AR) tools are promising for knowledge sharing, for assistance and for training. There is a lot of research on AR instructions in industrial work showing that compared to paper-based instructions, AR-based solutions are much faster to use, less errors are made, and the operators appear to accept the technology.

One operator could incorporate one or several of the proposed types: the Super-strength Operator (e.g., using Exoskeletons), the Augmented Operator (e.g., using augmented reality tools), the Virtual Operator (e.g., using a virtual factory), the Healthy Operator (e.g. using wearable devices to track well-being), the Smarter Operator (e.g., using agent or artificial intelligence for planning activities), the

Collaborative Operator (e.g., interacting with CoBots), the Social Operator (e.g., sharing knowledge using a social network) and the Analytical Operator (e.g., using Big Data analytics)[113] .

Over the past decades, advances in personal health technologies have enabled new ways of monitoring human behaviour and vital signals. Today, personal monitoring devices and applications such as wearable motion trackers, heart rate monitors and health-related mobile applications are easily accessible consumer products. Electromagnetic mm wave radars have gained increasing attention for adopting them in remote sensing of vital signs such as heart rate (HR), breathing rate (BR), blood oxygen density etc. For instance, radars can find HR and BR by detecting the chest wall movement. Game industry has catalysed development of eye/gaze tracking systems. Employees could also benefit from the use of personal health technologies to get empowering feedback of their well-being in relation to different jobs. However, significant numbers of employees are not interested in adopting the technologies currently available, or their use declines after some initial enthusiasm[114].

2.3.4 Experiences/ Case studies

Kozloduy nuclear power plant (Bulgaria) was the first NPP decommissioning project in Europe to use information technology to support the back-end stage of nuclear power plant units. The project, conducted by a Russian-German consortium comprising of GC NEOLANT, JSC NIKIMT Atomstroy, NUKEM Technologies GmbH, and EWN GmbH, lasted from 2016 to 2019. The main goal was the development of an equipment dismantling project in the controlled access areas of Kozloduy nuclear power plant, units 1-4.

Digital decommissioning using BIM was integrated. It provided reliable estimates of the amount of radioactive waste generated, up-to-date as-built documentation and the development of 3D engineering and radiation models[115].

[113] Kaasinen et al., 2019; Empowering and engaging industrial workers with Operator 4.0 solutions; Computers & Industrial Engineering; Volume 139, January 2020, 105678, ELSEVIER (2019)
[114] Mattila et al., 2013; Personal health technologies in employee health promotion: Usage activity, usefulness, and health-related outcomes in a 1-year randomized controlled trial; JMIR mHealth and uHealth, 1 (2) (2013)
[115] Kozloduy Nuclear Power Plant, http://www.neolant.com/dd/#id5 (accessed 30.10.2020)

2.4 Digital transformation in decommissioning (big data, business intelligence)

Digitalisation, as defined by GARTNER (2017)[116] is the use of digital technologies to change a business model and provide new revenue and value-producing opportunities; it is the process of moving to a digital business. A practical explanation is: ``*Digitalisation is about how we can use digital technology to do more with less effort and get it done quicker, safer, cheaper and with higher quality*`` (IFE, 2018)[117].

Digital transformation is radically changing how we think and work and gives significant competitive advantages to early adopters. Nuclear decommissioning has been a conservative area due to project managers being cautious about using methods that lack a long track record of successful application and detailed guidance on best practices. However, digital techniques are more commonly being taken advantage of for decommissioning projects. Typical technologies that are being used include: 3D laser-scanning, 3D computer-aided design, building information modelling, digital twins, asset inventories, virtual reality, augmented reality, and collaborative tools for project and resource management. IFE performed a survey, literature study, interviews with experts and group discussions for mapping needs and trends for application of digital technologies in nuclear decommissioning. Conclusions indicated that innovative digital concepts are necessary in case of decommissioning projects with difficult radiological conditions (e.g. Fukushima and Chernobyl), and can significantly enhance 'normal' decommissioning projects, especially if applied early in the process. Some of the cross-cutting activities where significant benefits are expected from higher application of new digitally enhanced techniques are related to early planning capabilities, information centric regulatory interaction, enhanced traceability of decisions and agility (preparedness for changes and emergencies). However, human resource development, and specifically training of decommissioning personnel, is an area that stands out both in terms of technology acceptance, by the intended end-users, as well as technology readiness level for practical field application. The results of the analysis also indicated that, in addition to enhanced training capabilities, application of digital concepts can also contribute to human resources development issues through motivating people (especially the young generation) for starting a career in nuclear decommissioning by providing generally applicable skills. Additionally, the survey revealed that digital technologies are mostly being applied for especially difficult tasks and specific subject areas rather than to gain an integrated holistic view of the whole process. However, it is an increasingly prevailing opinion that digital transformation would provide a much higher return on investment if, rather than on the level of specific tasks, is addressed from a systemic perspective, where people, the organisation and supporting technology are considered as a fully interconnected and interdependent system.

The following provides a summary of current trends for application of novel digitally enhanced concepts enabled by emerging hardware and advanced information technology in the different work phases/tasks of nuclear decommissioning projects.

[116] https://www.gartner.com/en/information-technology/glossary/digitalization, Accessed 22nd March 2021

[117] https://ife.no/en/research/digitalization/ , Accessed 22nd March 2021

2.4.1 Historical Site Assessment (HSA)

Often the first step in decommissioning projects is gathering historical information about the targeted environment. This information can come from a variety of sources including measurement, sampling, and modelling data. An important component of historical information relevant to decommissioning activities may exist as knowledge within the existing or earlier crew of the installation. There is a high potential for losing some part of this knowledge due to downsizing the team in the transition and decommissioning phases. Also, such kind of information is hard to capture using classical methods. In current practice, interviews with staff members are performed with relevant information captured in textual form. Capturing such historical information could be greatly enhanced by the application of user-friendly interactive visualisation of the environment where connections of the information to systems, structures and components (SSC) in the environment or procedures performed in the environment are easily made. Applicability also extends to the improvement of the registration of explicit information (data) relevant for decommissioning since the connection of such data (e.g. radiological contamination) to the environments is also very important.

2.4.2 Characterisation

The next step is the characterisation of the targeted environment to understand the initial conditions and constraints within the site. This includes analysing the information gathered during Historical Site Assessment (HAS) for planning further surveys (measurements and sampling). Concepts enabled by 3D visualisation and analyses of historical data have the potential for optimising the subsequent measurement and sampling by helping in identifying areas where elevated radiological hazards are likely and areas where inadequate historical information is available for planning further surveys. In addition, advanced support systems can improve the recording of information from surveys by providing in-situ guidance to field workers abut planed surveys (e.g. visualisation of place, schedule, and requirements for measurements and samples), as well as allowing connection of the obtained results directly to the SSC in the site.

2.4.3 Strategical planning (decision making)

Once adequate information is available for supporting decisions, strategical planning applies high-level decisions on the decommissioning approaches and methods. This task permits the road mapping of a strategy by providing understanding of the environment’s initial conditions and constraints, as well as the consequences and requirements of different options. Here, new methods of improving communication (shared decisions) between stakeholders can enhance current practice. For finding the best strategy, decision-makers need to be able to analyse and compare different available options in terms of consequences and requirements to constraints set by the availability of resources and national regulations. Hence, advanced support systems offering easy to understand visualisation of conditions, quick safety and efficiency evaluation, and visualisation of decommissioning options are expected to greatly improve future decommissioning strategies.

2.4.4 Detailed job planning

Following decisions on applied decommissioning approach, detailed planning of jobs for implementing the chosen approach will begin. In this step detailed, step-by-step plans are elaborated for the planned jobs for establishing work procedures, as well as determining requirements and resources related to these jobs. Resources include, for instance, time, people (and their expertise), tools, and materials required for implementing each specific job. Digitalised methods have potential for supporting estimation of resources required by jobs, as well as developing job procedures in a user-friendly way, by visualising steps of the work and related radiological conditions. This step may involve extensive utilisation of contractors for implementing jobs of different size. Contractors have different backgrounds and may have limited skills in radiation protection and familiarity with the targeted installation. Advanced visualisation tools would significantly improve job planning by supporting seamless information exchanges between contractors and utilities and ensuring a common understanding of the planned work process and related radiological risks.

2.4.5 Scheduling and resource allocation

In this phase, scheduling of the planned activities begins by distributing resources required by the different jobs by considering a workload balance to achieve an efficient overall work implementation by avoiding any unnecessary delays and breaching of limits. Things that planners (project managers) have to think about include: radiological waste amounts versus storage and transportation capabilities, doses to workers versus acceptable limits, types of radiological waste produced versus waste acceptance criteria, required personnel and expertise versus available stuff and external contractors, etc. Digital techniques have the potential for determining requirements related to planned jobs with low uncertainly and reliable risk estimations.

2.4.6 Training

Decommissioning work involves new and unique activities with an elevated risk of exposure to radiological hazards. As a consequence, for the riskier jobs, for example targeting the more contaminated or active parts of the installation, field workers need additional specialised training. Classical methods applied for training can be inefficient in preparing workers for jobs with higher risk, where an additional detailed understanding of the environment and radiological conditions is necessary. Conventional techniques are often inadequate in preparing workers for possible emergency scenarios. Advanced training methods, such as taking advantage of emerging technologies for immersive presence, are preferred to the more classical methods. While, at present, methods offering physical hands-on training environments provide better skills for field workers for jobs where correct physical interactions of humans with objects (e.g. components, tools) are important, in general, they are also expensive and less effective for learning the work process, familiarising with the environment and radiological conditions, and preparing for emergencies.

2.4.7 Briefing

Decommissioning field workers are briefed before jobs by providing verbal guidance, textual work description, and possibly two-dimensional illustrations (e.g. technical drawings). The resulting material from the application of advanced 3D techniques planning and training can be utilised for improving the briefing of field workers. The application of such techniques provides better situational awareness and understanding of the planned jobs and related risks. In addition, such methods would greatly improve the preservation of experience from completed jobs for use in subsequent jobs and in sharing the experience. For instance, storytelling based on realistic presentations of past incidents can successfully improve the safety culture.

2.4.8 Job execution

Advanced support systems, enabled by emerging hardware (e.g. mobile computing devices), have potential application during the implementation of jobs. In decommissioning, multiple teams may be working in parallel and/or on different shifts in an environment where physical and radiological conditions are continually changing. In such conditions, inadequate monitoring of the radiological conditions (by using outdated information), or inefficient in-the-field information being provided for field workers may result in unexpected situations and delays. Advanced methods based on digital technology have the potential for improving team and environmental monitoring as well as coordination of decommissioning teams.

The above steps may start consecutively during the project. However, there is a strong overlap between them during the project, which may result in the above steps being iteratively repeated. Figure 2.4-1 illustrates the use of digital technologies through consecutive work cycles during a decommissioning project.

Figure 2.4-1 Concepts enabled by advanced technologies for supporting the work cycle in decommissioning

Table 2.4-1 Summary of the state of the art for digital transformation in decommissioning

Methodology	What is working	What is missing	Assessment and Possibility for improvement
State of the art of digitalisation infrastructure in novel generation nuclear power plants (NPP)	Big Data & Business Intelligence (BI) solution management in novel NPP for SSC as well as for processes & costs, spanning the whole NPP lifecycle, including decommissioning	Decision to invest in Big Data & BI solutions for previous/older generation NPP in the decommissioning process	Change management, including challenges for starting digitalisation from scratch in previous/older generation NPP
Identification & prioritization of data sources (dismantled SSC, environment remediation and decommissioning taskforce)	Current digitization practices related to Big Data & BI in novel NPP	Big Data & BI IT infrastructure customisation for previous/older generation NPP in the decommissioning process	(1) Adapt Big Data & BI solutions from operations toward decommissioning; (2) Adapt & design characterisation processes (i.e. manual & automated) for decommissioning inventory (SSC, dismantling, waste, packaging, etc.) (3) Identify & implement infrastructure (e.g. IoT, RFID, etc.) for monitoring decommissioning SSC products & effects related-data (e.g. geolocation, changes over time – radiation, temperature, etc.), environment remediation, as well as effects on decommissioning taskforce

2.4.9 Experiences/Case studies

2.4.9.1 Italian Experience – AIGOR

Sogin is implementing a new integrated platform called AIGOR (Application for Radioactive Objects Management), based on blockchain protocols, which allows the planning and constant control of the treatment, conditioning, characterisation and storage of radioactive waste. The acquisition of the information collected from the interfaces of the various systems involved in the processes of dismantling and waste management will be implemented through an interface bridge that can be integrated with the AIGOR platform. [118]

[118] https://www.youtube.com/watch?v=qubkspvACVw

2.5 Supply chain management for Decommissioning

Typically, a significant share of the decommissioning work is outsourced by the nuclear facility's license holder. Implementation contracts can be the main tool for decommissioning project execution and, therefore, they bring major implications to project management, licensing, funding etc.

During the specific session on the IAEA workshop,[119] special consideration was given to the impacts that contract strategy selection has on the owner/client organisation.

"A good contract strategy will improve supply chain management whilst ensuring delivery for the owner/client at maximum value and minimal cost. In addition, a good contract strategy will support the delivery of best practice for the client and, if applicable, enable innovation from the subject matter experts engaged under the contract. There are several types of contracts used currently in decommissioning, e.g. are shown in Figure 2.5-1 below.

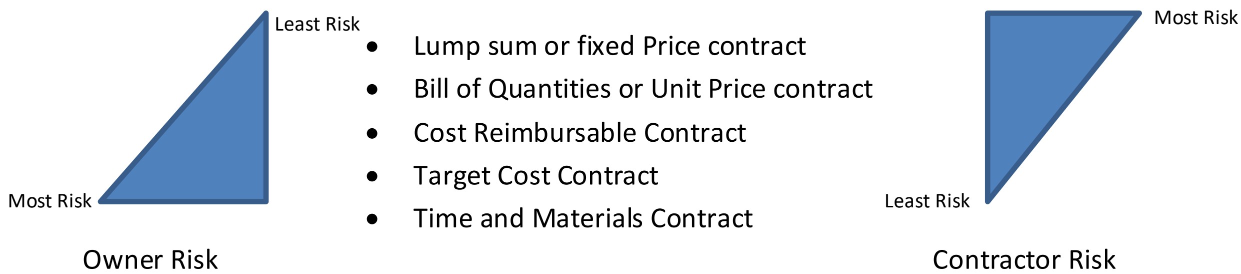


Figure 2.5-1 Type of contracts used currently in decommissioning

For fixed priced contracts, payment is made based on either performance against a costed schedule (sometimes being only specified milestones), or performance based on a costed bill of quantities. The contractor provides a fixed price for the agreed scope and is paid the full amount once the work is complete. Changes are managed by variations to the contract which may increase the cost or extend the programme or both. Penalty clauses may be added by the client for late delivery or other performance issues.

For cost reimbursable contracts, all costs are disclosed and paid (hours at an agreed rate for each labour type and all invoices for purchased items plus an agreed mark-up). Payments are made each period on this cost reimbursable basis until the work is complete.

Target cost contracts have similar payment methods, schedule or bill of quantities but the contractor also discloses all their costs (hours at an agreed rate for each labour type and all invoices for purchased/hired items plus an agreed mark-up). If the contractor can deliver for less cost an incentive can be paid, this results in a benefit to both sides, as the contractor receives an increased payment and the client has the work completed at less cost and usually less time. If the contract exceeds the target cost the incentive is reduced but the contractor never loses money on this type of contract, as full costs plus some fee is always paid if the target is exceeded.

[119] IAEA International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities, Tsuruga, Japan, 2019

Factors that may influence the selection of a contract; one key factor is the intended allocation of risk. As indicated in the graphic above a fixed price contract will result in most of the risk being taken by the contractor, whereas a basic time and materials contract means that the owner or client bears most of the risk. The owner/client can only transfer certain risks to the supply chain and, ultimately, they will always be responsible for the satisfactory implementation of decommissioning and therefore cannot reallocate the associated risk.

The contract strategy will also have a significant impact on the design of the organisational structure for project delivery required by the owner/client organisation and how they intend to manage the knowledge gained during the project.

A fixed price contract will result in an organisation that can provide oversight of the contract with a smaller team having contract management experience. A cost reimbursable contract may facilitate a more collaborative approach to delivery with the client and contractor organisations working together towards a common goal".

This session also went more in detail in the contractual and organisational approaches adopted in UK by Sellafield Limited and in Japan by JAEA and JAPC.

2.5.1 Description

2.5.1.1 Regulation pertaining to supply chain management

Generic requirements set out by the IAEA for supply chain management in the use of nuclear energy (without a specific reference to decommissioning) are [120] [Requirement 11]:

> *Management of the supply chain*
>
> ***The organisation shall put in place arrangements with vendors, contractors and suppliers for specifying, monitoring and managing the supply to it of items, products and services that may influence safety.***
>
> *4.33. The organisation shall retain responsibility for safety when contracting out any processes and when receiving any item, product or service in the supply chain[121].*
>
> *4.34. The organisation shall have a clear understanding and knowledge of the product or service being supplied[122]. The organisation shall itself retain the competence to specify the scope and standard of a required product or service, and subsequently to assess whether the product or service supplied meets the applicable safety requirements.*

[120] IAEA GSR-2 (Pub 1750)

[121] The supply chain, described as 'suppliers', typically includes: designers, vendors, manufacturers and constructors, employers, contractors, subcontractors, and consigners and carriers who supply safety related items. The supply chain can also include other parts of the organisation and parent organisations.

[122] The capability of the organisation to have a clear understanding and knowledge of the product or service to be supplied is sometimes termed an 'informed customer' capability.

4.35. The management system shall include arrangements for qualification, selection, evaluation, procurement, and oversight of the supply chain.

4.36. The organisation shall make arrangements for ensuring that suppliers of items, products and services important to safety adhere to safety requirements and meet the organisation's expectations of safe conduct in their delivery.

Regulating safety is a national responsibility, and many countries have adopted the IAEA's standards for use in their national regulations. For instance, in Finland, the Nuclear Energy Act reinforces these principles:

The licence holder shall be under an obligation to ensure the safe use of nuclear energy. This obligation may not be delegated to another party. The licence holder shall ensure that the products and services of contractors and subcontractors which affect the nuclear safety of the nuclear facility meet the requirements of this Act. [Section 9 of Nuclear Energy Act[123]]

Furthermore, following the regulatory structure applied in Finland, more detailed requirements set by the nuclear regulator. However, concerning supplier management, the content of this requirement is essentially the same as in the Act:

The licensee shall commit and oblige its employees and the suppliers and subcontractors whose involvement affects the safety of the nuclear facility to adhere to the systematic management of safety and quality. [Radiation and Nuclear Safety Authority Regulation on the Safety of a Nuclear Power Plant, STUK Y/1/2018[124]]

Finally, on the lowest (most technical) level of regulation, there are a large number of guidelines issued by the regulator (Ref. [125], in particular, relevant in this context are: *YVL A.3 Leadership and management for safety[126]; YVL A.4 Organisation and personnel of a nuclear facility* [127]; and *YVL D.4 Predisposal management of low and intermediate level nuclear waste and decommissioning of a nuclear facilit*y[128]). These guidelines are non-binding in the sense that the licensee can always propose an alternative way to fulfil the goal of a single requirement, if the achieved safety level is at least as high. A number of specific requirements in the above-mentioned guides reiterate the responsibility of the licensee to ensure the suppliers' ability to act safely in the same manner as the staff of the licensee. E.g., YVL A.3, requirement 402 states:

The licensee is obliged to ensure that the regulatory requirements and guides are complied with. This shall also be taken into account during the procurement of products and services having a bearing on the nuclear and radiation safety of the nuclear facility. It shall be ensured that organisations contributing to the plant delivery or plant modifications understand and comply with the delivery-related

[123] https://www.finlex.fi/fi/laki/kaannokset/1987/en19870990.pdf
[124] https://www.stuklex.fi/en/maarays/stuk-y-1-2018
[125] https://www.stuk.fi/web/en/regulations/stuk-s-regulatory-guides/regulatory-guides-on-nuclear-safety-yvl-
[126] https://www.stuklex.fi/en/ohje/YVLA-3
[127] https://www.stuklex.fi/en/ohje/YVLA-4
[128] https://www.stuklex.fi/en/ohje/YVLD-4

requirements. The licensee shall communicate the requirements to the product suppliers by contractual means (contract documents) and ensure and control the fulfilment of the requirements throughout the supply chain.

Implementation of these requirements into the licensee's management system is discussed in Section 2.5.1.3.

2.5.1.2 Procurement procedures

Any supply contract starts with a procurement procedure executed by the contracting body. Here, not only safety concerns set the boundary conditions for contracting, but there is additional legislation concerning procurement, especially public procurements. The selection of the most suitable procedure depends on the clarity/complexity of the scope and the availability of existing solutions for the purpose. For instance, in the case of unique or rare reactor types to be decommissioned, the dismantling techniques or waste management solutions may require significant additional development. To attract tenderers to offer services in such cases, additional incentive can be provided to them by selecting a procurement procedure that includes an element of development.

2.5.1.2.1 Public procurement procedures

In the case of public contracting bodies, or other bodies operating on public funding, their procurements are usually subject to rules aimed at producing the best value for the use of public funds. On the EU level, the directive 2014/24/EU on public procurement[129] defines the boundaries within which the EU Member States can implement their respective national legislation on public procurements.

The more complex the scope of the procurement, the more important it is to allocate sufficient time and expertise both in the substance matter and procurements to achieve a good contract, which forms the basis for a working relationship with the supplier. The best competence on decommissioning is likely to lie at the suppliers' side and can be utilised for mutual benefit in the procurement by selecting a participatory procurement procedure (e.g., competitive procedure with negotiation, competitive dialogue, or innovation partnership, see Table 2.5-1).

[129] https://eur-lex.europa.eu/eli/dir/2014/24/oj

Table 2.5-1 Procurement procedures according to the EU directive 2014/24/EU.

Procurement procedure	Participation and selection of participants	Prerequisites for application	Benefits	Challenges
Open procedure	Any interested economic operator may submit a tender in response to a call for competition.	The "default" procedure – no specific prerequisites.	Most straightforward procedure with shortest minimum time to complete procurement.	Scope must be clearly defined by the contracting body for fair and transparent comparison of tenders.
Restricted procedure	Any economic operator may submit a request to participate in response to a call for competition [...] by providing the information for qualitative selection that is requested by the contracting authority.	Applicable e.g. if the contracting authority wishes to limit the number of tenderers, or in cases in which technical specifications contain sensitive information, which cannot be published openly (request to participate includes signed NDA).	Helps limiting the number of tenders to be compared (savings in work). Better control of information than in open procedure.	Two-step procedure; increases the minimum time to complete procurement.
Competitive procedure with negotiation	Same as above.	Needs of the contracting authority cannot be met without adaptation of readily available solutions. Scope of contract includes design or innovative solutions. Prior negotiations are necessary because of specific circumstances related to the nature, the complexity or the legal and financial make-up or because of the risks attaching to them. Technical specifications cannot be established with sufficient precision by the contracting authority.	Brings together the knowledge and boundary conditions from all potential tenderers. Can significantly improve the quality of the final Call for Tenders. Competitive nature of tendering is preserved.	Significantly longer process than open or restricted procedure (several phases, more work and time). Requires careful adaptation of the Terms of Reference between negotiations and launching of final Call for Tenders. Attention to be paid to equal treatment of tenderers.
Competitive dialogue	Any economic operator may submit a request to participate in response to a contract	Same as above.	Compared to the above, additional freedom for the	Attention to be paid not to reveal to the other participants solutions proposed or

	notice by providing the information for qualitative selection that is requested by the contracting authority.		tenderer to offer their own optimal solutions. Tenderers can be compensated for their efforts. Competitive nature of tendering is preserved.	other confidential information communicated by a tenderer.
Innovation partnership	Same as above.	Development and purchase of an innovative product, service or works that cannot be met by purchasing products, services or works already available on the market.	Provides a framework and an opportunity for a broad partnership. Tenderers can be compensated for their efforts. Helps attracting tenderers (innovation partners) to develop solutions by lowering their risk. Competitive nature of tendering is preserved.	Little experience on the use of the procedure so far. Uncertainty on the result. Requires correct description of the needs. Additional costs from the compensation of development in case of several innovation partners. Contract conditions, including IPR questions.
(Negotiated procedure without prior publication)	The Directive allows EU Member States to implement on their national legislation a negotiated procedure without prior publication of a call for competition, to be applied in specific cases and circumstances.			

2.5.1.2.2 Procurements by privately owned operators

Privately owned operators have generally more freedom in selecting the method they prefer using in procurements. While they can avoid some of the formalism related to public procurements, the goal is the same: Achieve the best value for shareholder money.

2.5.1.3 Implementation of supply chain management in operator’s management system

The certified quality and environmental management system of a nuclear operator provides a solid basis for the management of the supply chain also in nuclear projects. In practice, similar certifications are required from the suppliers and subcontractors. In addition, reflecting the requirements reviewed in Section 2.5.1.1, it is practically mandatory that suppliers and their subcontractors work under complete control of the licensee's organisation as regards nuclear and radiation safety. Typically, the internal rules and regulations of the nuclear facility define those additional practices, and these rules and regulations must be approved by the nuclear regulator. There must be a clearly defined responsibility for one or several managerial positions in the licensee's organisation to ensure that suppliers and their subcontractors fulfil all safety requirements and that their safety culture is good in general. It is obvious that these practices must be written clearly in the supply contracts, and brought up early during the procurement of services, to make sure that the suppliers and their subcontractors are well prepared already at the time of tendering.

2.5.2 Experiences/Case studies

2.5.2.1 Experience at VTT

In March 2020, VTT Technical Research Centre of Finland Ltd. (state-owned non-profit company, licensee and operator of the permanently shut down FiR 1 research reactor) awarded a contract on *decommissioning services for FiR 1 research reactor and OK3 materials research laboratory, including management of nuclear waste and other radioactive waste*[130]. VTT used to competitive procedure with negotiation, because the legal and technical boundary conditions for the contract (e.g. exact scope of the procurement, licensing questions related to the waste management services) were open at the time of the contract notice. The duration of the procedure was about 11 months (see Figure 2.5-2).

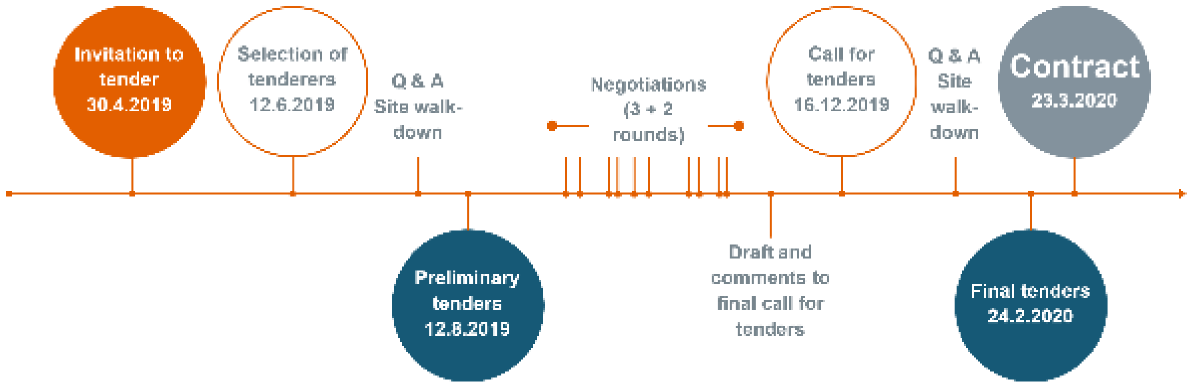


Figure 2.5-2 Timeline of a case example (VTT, Finland) using the competitive procedure with negotiation in contracting decommissioning and nuclear waste management services for a research reactor and radioactive materials research laboratory in 2019–20. Source: VTT

2.5.2.2 Experience at CEA

CEA have over twenty facilities currently under decommissioning on 5 different sites and an associated annual expenditure budget of around 740 M€ which relies on several large companies to assume the role of prime contractors. The contractors need to be notified by a committee for certification of radioactive clean-up companies called "CAEAR" to be allowed to enter contracting competitions in the field of D&D within CEA.

130 procurement notice: https://www.hankintailmoitukset.fi/fi/public/procurement/18939/notice/43638/overview; in Finnish

CAEAR covers several domains:

- Domain D2 : process management or full management of a nuclear facility
    - D2-1 : full management of a nuclear facility
    - D2-2: process management within a nuclear facility
- Domain D3 : clean-up or dismantling operation of a nuclear facility
- Domain D4 : project management, assistance to project management and services or studies contracts for clean-up and dismantling operations of a nuclear facility
    - D4-1 : project management
    - D4-2 : assistance to project management and specific conception studies

CAEAR has the duty to verify and validate the capacities of the companies:
- Quality of work, importance given to safety-security-radioprotection objective
- Skills management, training,
- Management, organisation,
- Safety culture : organisation, individual behaviors,
- Feedback (technical aspects, safety, security, radioprotection, human and organisational factor,)
- Equipment and practices for radioprotection controls,
- Management of subcontractors

CAEAR acceptance is for 3 years and during that period the acceptance can be suspended following errors.

2.6 Methods and tools for communication (public)

The decommissioning of nuclear reactors involves significant technical, environmental and material disposal challenges. As the end of the service life approaches for these facilities, plans for decommissioning of the facilities must be developed and *efforts to familiarise the public* with the decommissioning process and disseminate information on upcoming nuclear decommissioning projects must be made. All this information must be made available for reference use by the media and the public.

To facilitate the continuation of public involvement and participation in the decommissioning process workshops and/or topical conferences to review recent experiences and discuss future decommissioning challenges can be organised. The goal of these meetings are also to disseminate information to the public on the results of recently completed projects (with the main focus on the lessons learned), identify issues of concern (this might be country dependent), and elicit recommendations on future decommissioning operations and associated technical, environmental, socioeconomic and disposition issues. In general, the sets of values below are strongly recommended by experts to be followed when it comes to public information:

- The simplification of the concepts,
- The understanding of the concepts,
- The transparency of the processes,
- The need to harmonise the different sets of criteria for material management.

Few members of the public are familiar with nuclear reactors beyond the general information available from non-technical sources (newspaper, social media, TV). For example, in the USA the *Public*

Electronic Reading Room provides access to the NRC's new records-management system[131] whilst the Norwegian Nuclear Decommissioning (NND) public information centre[132] is supporting planned decommissioning activities in Norway. Although not required by regulations, public meetings organised by technical experts in the vicinity of the facilities to keep the public informed is recommended. Information made available to the public should be transparent, understandable and accessible.

It is important that the workers and the public are aware of the precautions taken during decommissioning activities. It must be emphasised that no unwarranted doses will be incurred by the public during the activities.

To increase confidence basic information on radiation and the associated risks during decommissioning should be made public. Even if the existing regulations and practices used during decommissioning protect the workers and the population, it remains that the public needs to be informed of those measures and that its protection is real. In the USA, workers at the NPPs, or members of the public that have specific concerns of a safety-related nature, can bring safety concerns directly to NRC[133].

Confirmation of the compliance of the decommissioning activities with the regulations should be part of the information given to the public.

Harmonisation of processes, strategy, and criteria within the EU is necessary to prevent public rejection of decommissioning projects[134] since there could be public concern as to how individual country's decommissioning strategy was decided and why it is different from others. It is useful to identify the causes of variations, and to communicate these to the public.

Studies have shown that the various strategies for material management (disposal and replacement or recycle and reuse) can have a different impact on public opinion. This should be considered when deciding between different options, e.g. recycle in the nuclear industry, recycling in the non-nuclear industry, conditional release, and traceability[135]. It should be noted that difficulties have been met in the acceptance of cleared material by scrap dealers or commercial smelters who refuse this kind of raw material for their production.

Information should also be available on international waste and material management, such as transportation, conditioning, disposal, and recycling between countries. Among this information, there needs to be information regarding the financing plans. Sound decommissioning financing will increase the public acceptance of legacy waste. However, it needs to be emphasised that since only 21 reactors have been decommissioned worldwide (as of 2018) there is little historical data to help estimate costs.

---

[131] http://www.nrc.gov/NRC/NUREGS/BR0010/index.html
[132] https://www.norskdekommisjonering.no/about-nnd/
[133] https://www.nrc.gov/docs/ML0037/ML003726190.pdf.
[134] http://www.wenra.org/harmonisation/working-group-waste-and-decommissioning
[135] https://hrcak.srce.hr/ojs/index.php/rgn/article/view/5222/pdf)

Since the start of decommissioning programmes, several major decommissioning projects have been completed and several others are underway or moving forward. The lessons learned from these are of extreme value. The purpose of the reports from the Decommissioning Lessons Learned Database is to provide support and/or give recommendations not only in the decommissioning but also in the planning phases. The links below provide quick access for the public to the reports derived from the decommissioning lessons-learned database: 2007[136], 2006[137], 2005[138], 2004[139], 2002[140] and 2013-2016[141].

2.6.1 Experiences/Case studies

2.6.1.1 Norway's approach - the NND public information centre

The timely establishment of suitably equipped public information centers supporting planned decommissioning activities has proven to be of key importance for ensuring that activities related to decommissioning and waste disposal can progress smoothly without generating public disturbance, ultimately leading to delays and cost overruns. In addition to ensuring a positive public attitude, public information centers can also greatly improve the confidence of all stakeholders in that the funds are being optimally used and safety is being upheld to high standards.

The Norwegian Nuclear Decommissioning (NND) organisation is establishing a (physical) showroom at its premises where it can show to the public the decommissioning process of different sites. The NND info center is dedicated to public awareness of nuclear facilities and decommissioning progress and is heavily relying on IFE’s unique competence in Norway for decommissioning information centers. IFE’s competencies combine digital visualisation technologies with 3D safety modelling and information, a systemic approach to integrating information centers into organisational processes, as well as real-life experience from decommissioning information centers abroad.

In 2019, NND ordered a series of 3D-models from IFE to supply their new showroom. The first phase of the project resulted in two important outcomes:

(i) a software platform that can combine geographical and facility data, as well as procedure simulations in an information rich user-interface,

(ii) a maturing procedure for safety and security evaluation of data/information to be presented within the platform. The main aims of the on-going phase of the project are 1. enriching the content within the platform (including the national repository) while 2. also further improving the platform itself (based on end-user feedback) and 3. the process for safety and security evaluation of the content. This second phase also aims at defining a process from a safety and security perspective for maintaining a public and a non-public version of the platform.

For the information to be presented in public in a securely, each model had been assessed and controlled according to laws and regulations. The laws governing information concerning nuclear facilities and critical infrastructure are the Norwegian Security Act (Sikkerhetsloven) and the

[136] https://www.nrc.gov/docs/ML0734/ML073460310.pdf
[137] https://www.nrc.gov/docs/ML0734/ML073460307.pdf
[138] https://www.nrc.gov/docs/ML0734/ML073460306.pdf
[139] https://www.nrc.gov/docs/ML0734/ML073460304.pdf
[140] https://www.nrc.gov/docs/ML0734/ML073460299.pdf
[141] https://www.nrc.gov/docs/ML1608/ML16085a029.pdf

Norwegian Atomic Energy Act. Additionally, the information was considered according to the IFE Information Security Handbook.

During the public visits, NND will present the process for regulating decommissioning, its current plans for it and will use IFE's 3D platform for promoting the approach for applying cross-cutting techniques for supporting efficiency, safety and transparency of Norwegian decommissioning activities. A portion of the available time will be devoted to questions and answers. Comments and questions were submitted in writing before or post-visit.

Figure 2.6-1 Concept demonstration figure for 3D simulation supported public decommissioning information centre

2.6.1.2 Chernobyl Decommissioning Visualisation Centre (CDVC) supported by Norway

The Norwegian Ministry of Foreign Affairs (NMFA) funded in the period 2006-2016 an assistance project at the Chernobyl Nuclear Power Plant (ChNPP) in Ukraine. The goal of the project was to assist in the decommissioning of the site through better planning and training by establishing the Chernobyl Decommissioning Visualisation Centre (CDVC) based on Virtual Reality (VR) technology.

The CDVC enables the ChNPP staff to plan procedures for various work tasks involved in dismantling the facility with a special focus on the tasks in the radioactive parts of the plant. The CDVC consists of the following software developed by the IFE: ChNPP ProCre, ChNPP Planner and ChNPP PCT.

- ChNPP ProCre makes it possible for a group of engineers and other specialists to test out plans and procedures for various work tasks involved in dismantling a nuclear facility.
- The ChNPP Planner allows users to plan the work by simulating tasks in a VR model of the work environment and by using mannequins to represent workers. The ChNPP Planner then calculates in real-time the radiation exposure for the given scenario. The user can view dose distribution graphs to determine if the doses are within acceptable limits and to identify when and where high radiation doses were incurred. The ChNPP Planner thereby supports the principle "as low as reasonably achievable" (ALARA).

- ChNPP Procedure Creator and Trainer (ChNPP PCT) can be used for training personnel doing work tasks before actually doing the tasks in real life. One or more trainees can do collaborative training on a predefined work procedure in a VR environment.

Figure 2.6-2 The user interface in ChNPP Planner

CDVC supports ChNPP in preserving the decommissioning expertise and knowledge at the plant from a long-term perspective. The CDVC can be a pedagogical means for the ChNPP personnel in training the dismantling procedures. At the same time, CDVC provides the decommissioning project team with an effective medium in presentations to the public as well as for communicating with the management and the licensing authorities[142].

2.6.1.3 France's approach – National and local initiatives

French National Initiatives:

The law of 13 June 2006 on transparency and nuclear safety, known as the "TSN" law, provides that everyone has the right to be informed about the risks associated with nuclear activities. It is for this purpose that the High Committee on transparency and information on nuclear safety (HCSTIN) has been created, as a body for information, consultation and debate on the risks associated with nuclear activities and the impact of these activities on human health, environment and nuclear security. The HCSTIN is composed of 40 people from the National Assembly, the Senate, Local Information Committees, associations, nuclear operators, national unions, experts on science, risk prevention, with around four meetings a year.

Another initiative is the National plan for management of radioactive materials and waste (PNMGDR) as discussed in Section 1.2.2.1.

Local Initiatives:

Established in 2000, ANCCLI is the National Association of Local Information Committees and Commissions. It brings together 34 Local Information Commissions. In France, each nuclear installation

[142] https://www.youtube.com/watch?v=txnoidoZzEk&feature=share

has a Local Information Commission (CLI). The CLI has a dual mission: to inform the population about nuclear activities and to ensure permanent monitoring of the impact of nuclear installations.

Members are elected and consist of the departmental council, and territorial bodies, members of environmental protections association, representatives of professional unions, representatives of local economic interests, physician, and experts. It is consulted for any project related to the nuclear installations.

CEA local Initiatives in Marcoule:

CEA focused on communication with the public, for example, the Marcoule site region where most of the D&D projects take place. An example of a large project, include the public inquiries in 2014.

CEA have dedicated units as communication tools: Visiatome, opened in 2005, and InfoDEM, opened in 2013. The main aim of these initiatives have been to inform the public about nuclear, not to convince or change their political opinion and to provide information to children about the application of science. The local community's perception of Marcoule has changed following these initiatives. Additionally, there has been a positive impact on families, through educational workshops for children.

2.7 Methodologies and guidance for cost estimation

Reliable cost estimating is one of the most important elements of decommissioning planning. Decommissioning cost estimates may serve a variety of purposes and can have a wide variation depending on the decommissioning strategy adopted, the assumed end state and differences in basic assumptions as well as the context in and purpose for which the estimates were prepared.

Estimate accuracy evolves with the project maturity, improving as more detailed information becomes available. It is important to note that cost estimates should fit their intended purpose and be appropriate for the stage of the facility's lifecycle for which they are produced.

Decommissioning cost estimates are performed by many organisations in the nuclear industry (e.g. operators, contractors, regulators). There is no universally accepted standard methodology at present for developing decommissioning cost estimates, which present considerable variability in formats, contents and practices.

In recent years, given the considerable increase in the number of nuclear facilities to be dismantled, has arisen the need to make the methodologies and tools as much homogeneous as possible.

The need for standardisation was especially gathered by the NEA which, together with the IAEA, intensified its efforts in the sector and from 2006 to 2012 published numerous reports on the costs of decommissioning produced by Working Groups formed by international specialists in the sector.

Furthermore, IAEA provided some requirement and suggestions for financing and cost estimating of the decommissioning activities in the General Safety Guides part 6 [143], and in the Specific Safety Guides [144] and [145]. They provided general requirements and suggestions for funding and financing decommissioning, the related responsibility and the identification of costs that should be included in the estimations.

Particularly notable is the establishment of the new "International Structure for Decommissioning Costing (ISDC) for Nuclear Installations ", jointly developed by the NEA, the IAEA and the EC to enhance consistency and improve comparability of estimates across countries [146]. The ISDC provides general guidance on developing decommissioning cost estimates and, through its itemisation, a tool either for cost estimation or for mapping estimates onto a standard, common structure for comparison purposes. This document constitutes a revision of the previous publication of the known NEA as the "Yellow Book" of 1999.

Similarly, in the framework of the Committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM), an expert group on Costing for Decommissioning of Nuclear Installations and Legacy Management (EGCDL) was created in June 2020, the EGCDL will address costing issues for decommissioning of nuclear installations and legacy management. Despite such efforts, it is worth to

---

143 IAEA Safety Standards - Decommissioning of Facilities - General Safety Requirements Part 6 No. GSR Part 6 – 2014

144 IAEA Safety Standards - Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities – Specific Safety Guide No. SSG-47 – 2018

145 IAEA Safety Standards - Decommissioning of Medical, Industrial and Research Facilities Specific Safety Guide No. SSG-49

146 NEA - International Structure for Decommissioning Costing (ISDC) for Nuclear Installations - 2012

mention that there in not an internationally agreed consensus on how to develop cost estimates. Furthermore, it is recognized the current difficulties to make comparison between planned or completed project due to a number of reasons[147] of which confidentiality is the major one.

2.7.1 Description of the methodologies

As previously reported, there is no universally accepted standard for developing and classifying cost estimates.

Various organisations use the concept of classification or class of estimate to describe the quality of the underpinning data, the completeness and reliability of the estimate.

For example, AACE International [148]establishes standards for the accuracy of cost estimates that are based on the degree of known information at the time of the estimate. These classification and cost estimating methodologies found general application in engineering, procurement and construction. Here below the AACE estimate classification.

	Primary Characteristic	*Secondary Characteristic*		
ESTIMATE CLASS	**MATURITY LEVEL OF PROJECT DEFINITION DELIVERABLES** Expressed as % of complete definition	**END USAGE** Typical purpose of estimate	**METHODOLOGY** Typical estimating method	**EXPECTED ACCURACY RANGE** Typical variation in low and high ranges at an 80% confidence interval
Class 5	0% to 2%	Concept screening	Capacity factored, parametric models, judgment, or analogy	L: -20% to -50% H: +30% to +100%
Class 4	1% to 15%	Study or feasibility	Equipment factored or parametric models	L: -15% to -30% H: +20% to +50%
Class 3	10% to 40%	Budget authorization or control	Semi-detailed unit costs with assembly level line items	L: -10% to -20% H: +10% to +30%
Class 2	30% to 75%	Control or bid/tender	Detailed unit cost with forced detailed take-off	L: -5% to -15% H: +5% to +20%
Class 1	65% to 100%	Check estimate or bid/tender	Detailed unit cost with detailed take-off	L: -3% to -10% H: +3% to +15%

Figure 2.7-1 AACE estimate classification

IAEA and NEA have developed in the context of nuclear decommissioning a classification based on three types of cost estimates and each type have a different level of accuracy. These cost estimate types are the following:

- **Order-of-Magnitude Estimate**: One without detailed engineering data, where an estimate is prepared using scale-up or -down factors and approximate ratios. It is likely that the overall scope of the project has not been well defined. The level of accuracy expected is -30% to +50%.
- **Budgetary Estimate**: One based on the use of flow sheets, layouts and equipment details, where the scope has been defined but the detailed engineering has not been performed. The level of accuracy expected is -15% to +30%.
- **Definitive Estimate**: One where the details of the project have been prepared and its scope and depth are well defined. Engineering data would include plot plans and elevations, piping

[147] NEA - Costs of Decommissioning Nuclear Power Plants – 2016
[148] AACE International Recommended Practice n. 18R-97 “Cost estimate classification system – as applied in engineering, procurement and construction for the process industries” – March 2019

and instrumentation diagrams, one-line electrical diagrams and structural drawings. The level of accuracy expected is -5% to +15%.

It is apparent from these estimate types and the associated levels of accuracy that, even in the most accurate case, a definitive estimate is only accurate to -5% to +15 %.

It is crucial that the cost estimator needs to exercise his/her judgment as to the level that the input data will support. The estimator includes sufficient margin (or contingency) to account for a potential budget overrun.

With reference to cost estimating methodology according to IAEA and NEA documents [149], [150] there are five different approaches:

1. **Bottom-up technique**: Generally, a work statement and specifications or a set of drawings are used to extract ("take off") material quantities required to be dismantled and removed and unit cost factors (UCFs) (costs per unit of productivity – per unit volume or per unit weight) are applied to these quantities to determine the cost for removal. Direct labour, equipment, consumables and overhead are incorporated into the UCFs. The process involves breaking the project down into its smallest work components or tasks, assigning the work into a work breakdown structure (WBS), estimating the amount of labour, materials and consumables to accomplish each task, the duration of each task and then aggregating them into a full estimate
2. **Specific analogy**: Specific analogies depend on the known cost of an item used in prior estimates as the basis for the cost of a similar item in a new estimate. Analogous estimating uses a similar past project to estimate the duration or cost of the current project. Adjustments are made to known costs to account for differences in relative complexities of performance, design and operational characteristics. It may also be referred to as ratio-by-scaling. Specific analogy estimating requires a detailed evaluation of the differences between a similar past project and the current project. Adjustment for these differences is an important element of this approach. It includes size differences, complexity differences, labour cost differences, inflation/escalation adjustments and possibly regulatory differences
3. **Parametric:** Parametric estimating requires historical databases on similar systems or subsystems. Statistical analysis may be performed on the data to find correlations between cost drivers and other system parameters, such as units of inventory per item or in square metres, per cubic metres, per kilogramme, etc. The analysis produces cost equations or cost estimating relationships (CERs) that may be used individually or grouped into more complex models.
   A parametric cost estimating model is made up of one or more algorithms or CERs that translate technical and/or programmatic data (parameters) about an activity into cost results. The algorithms are commonly developed from regression analysis of historical project information however other analytical methods are sometimes used. The models are very useful for cost and value evaluations early in the project life cycle when not much is known about the project scope. The models are dependent on the many assumptions built into the algorithms. Also, the validity of the model is usually limited to certain ranges of parameter

[149] IAEA-TECDOC-1476 - Financial aspects of decommissioning - November 2005
[150] NEA - The Practice of Cost Estimation for Decommissioning of Nuclear Facilities - 2015

values. For example, size differences of 100% between the past project and the current project would not be reasonable. Due to these limitations and constraints, it is incumbent upon the user to thoroughly understand the basis of a parametric model.

4. **Cost review and update:** An estimate may be developed by examining previous estimates of the same or similar projects for internal logic, completeness of scope, assumptions and estimating methodology. This approach applies to updating a previous estimate to the current estimate and generally does not involve size difference considerations.
5. **Expert opinion**: This may be used when other techniques or data are not available. Several specialists may be consulted iteratively until a consensus cost estimate is established.

The most widely adopted internationally and the most common methodology to calculate a detailed site-specific cost estimate is the bottom-up technique.

Table 2.7-1 Summary of the methodologies for cost estimation

Methodology	What is working	What is missing
Bottom-up	It could be applied to Decommissioning project with a clear project scope and available inventory data Most accurate as it accounts for site-specific radiological and physical inventory. Relies on UCFs.	Requires detailed description of inventory and site-specific labour, material and equipment costs for the UCFs
Specific analogy	Applied to similar decommissioning projects in different time period Accurate if prior estimates are appropriately adjusted for size differences, inflation and regional differences in labour material and equipment	Adjustments as noted may require detailed documentation and introduce approximation that reduce accuracy
Parametric	Applied to Decommissioning project where historical data are available, but the project scope is not clear Suitable for use for large sites where detailed inventory is not readily available. Suited for order of magnitude estimates	Approximations based on areas or volumes introduce additional inaccuracies. There is no way to track actual inventory. Not suited for project planning of work activities
Cost review and update	Applied to Decommissioning estimate update Suitable for large sites where detailed inventory is not available. Suited for update of previous estimates, or order of magnitude estimates.	There is no way to track actual inventory. Generally, not suited for project planning of work activities
Expert opinion	Applied to Decommissioning projects when no other techniques are available Suitable when expert opinion of the specific work is available. Can be used for estimating productivity of smaller tasks based on expert's experiences	Expert opinion may not be specific to the work activities. May not reflect the radiological limitations of the project

To offer international actors' specific guidance in preparing quality cost and schedule estimates also increasing data traceability, in 2015 NEA published the guide [151] where are detailed four basic elements of a cost estimate:

[151] NEA - The Practice of Cost Estimation for Decommissioning of Nuclear Facilities - 2015

1) Basis of Estimate (BoE): it is the foundation upon which the cost estimate is developed. It is based on the currently applicable decommissioning plan or decommissioning concept for the facility. A typical list of items that might be included in the basis of estimate are shown in the following:

    - assumptions and exclusions;
    - boundary conditions and limitations – legal and technical (e.g. regulatory framework);
    - decommissioning strategy description;
    - end point state;
    - stakeholder input/concerns;
    - facility description and site characterisation (radiological/hazardous material inventory);
    - waste management (packaging, storage, transportation and disposal);
    - spent fuel management (activities included into a decommissioning project);
    - sources of data used (actual field data vs. estimating judgement);
    - cost estimating methodology used (e.g. bottom-up, specific analogy);
    - contingency basis;
    - discussion of techniques and technology to be used;
    - description of computer codes or calculation methodology employed;
    - schedule analysis;
    - uncertainty and management of risk.

    A well-documented BoE should fully describe the boundaries of the project scope and define basis for the cost estimating process, including the consideration of estimating uncertainty and risk.

2) Structure of estimate: it is constructive and helpful to group elements of costs into categories to better determine how they affect the overall cost estimate. To that end, the work scope cost elements are broken down into activity-dependent, period-dependent, and collateral costs and special items. Contingency, another work scope element of cost, may be applied to each of these elements on a line item basis because of the unique nature of this element of cost.

3) Work Breakdown Structure (WBS) and schedule: the WBS is used to categorise cost elements and work activities into logical groupings that have a direct or indirect relationship to each other. The work groupings are usually related to the accounting system or chart of accounts used for budgeting and tracking major elements of the decommissioning costs.

4) Uncertainty and Risk Analysis: Risk analysis is a means of dealing with decommissioning project problems that extend beyond the project scope, the risk potentially causing an increase in cost or an opportunity resulting in a decrease in costs. Risk analysis has become an integral part of cost and schedule estimating in recent years.

An example of cost estimate methodology is the “Unit Cost Factors using Bottom-up Technique”. This methodology provides detailed activity-dependent cost estimate by breaking down the decommissioning project into a series of work activities according to the defined WBS. For each activity a unit cost factor is estimated.

The UCF is developed from a description of the activity to be performed, the estimated time to perform the activity under ideal conditions, the estimated productivity or work difficulty factor (WDF), the applicable crew composition and number of workers of each category, and the equipment and consumables required to perform the activity:

UCF = (sum of labour cost + equipment +consumables cost) / unit quantity

Labour cost = (estimated time for activity X WDF X crew cost/hour) / unit quantity

The Work Difficulty Factor (WDF) is defined as:

WDF = % increase in time for the activity for the degree of difficulty expected

The application of work difficulty factors is intended to account for the productivity losses associated with working in a difficult or hazardous environment. The approach is widely used at operating power plants to account for difficulty in performing maintenance activities during outages. The application of this methodology to decommissioning activities is a natural and reasonable extension of this work adjustment factor. The NEA document [152] gives a reference for value ranges for different work difficulty factors.

Once the UCF are determined they are applied to the entire inventory of systems and structures to determine the activity costs and the duration to perform them in a defined sequence.

Costs for activities (removal of pipe, valves, pumps, tanks, heat exchangers, ducting, electrical conduit and cable trays) are estimated by the following formula:

Activity cost = inventory quantity X unit cost factor

The Bill of Quantities (BoQ) of each facility is developed from the site-specific information for the facility.

2.7.2 Experiences/Case studies

There are several NEA-IAEA technical and guidance documents in which experiences, methodologies and case studies are presented.

NEA report [153], published in 2016, reviews nuclear power plant decommissioning costs and funding practices adopted across NEA member countries, based on an analysis of survey data collected through an NEA questionnaire. The report highlights that cost data are estimates related to future projects, with the exception of the José Cabrera NPP in Spain (single PWR unit) that was undergoing decommissioning, and for which cost figures refer partly to expenditures incurred for completed tasks and partly to estimates for outstanding activities.

NEA report [154] was prepared in 2015 to offer international actors' specific guidance in preparing quality cost and schedule estimates to support detailed budgeting for the preparation of decommissioning

[152] NEA - The Practice of Cost Estimation for Decommissioning of Nuclear Facilities - 2015
[153] NEA - Costs of Decommissioning Nuclear Power Plants – 2016
[154] NEA - The Practice of Cost Estimation for Decommissioning of Nuclear Facilities - 2015

plans, the securing of funding and decommissioning implementation. The guide is based on current practices and standards in several NEA member countries.

Regarding the cost estimating methodologies and tools some national practices are reported in the NEA report [155] and summarised in the following Table 2.7-2. Due to the time that has passed since the table was produced an update of the table is required.

Table 2.7-2 National cost estimation practices[156]

Country	Belgium	France	Italy	Japan	Spain	Sweden
Topics						
Organisation of costs	Most use a work breakdown structure for their cost estimates but not all follow the Yellow Book List of Cost Items:					
	Based on Yellow Book.	Own WBS.	Own WBS.	Own WBS.	Uses Yellow Book.	Uses Yellow Book.
Cost estimating methodologies and tools, and their impact on the life cycle of the project.	National guidelines in most countries allow some degree of operator discretion as to choice of cost calculation method.					
	ONDRAF/NIRAS can specify general information requirements though not how material is collated or methods used.	Phased approach in developing estimates with increasing accuracy from opportunity study to implementation (CEA).	Cost methodology developed by SOGIN.	Linear approximation related to weight of materials from decommissioning and individual accumulation method (METI)	Cost methodology developed by national waste management organisation (ENRESA).	Cost methodology developed by national waste management organisation (SKB).
Where estimates go wrong – review of experiences.	No broad experience in comparing estimated costs to actual costs.	Changes in original strategy and end-point conditions represent one of the greatest factors.	Unexpected regulatory prescriptions and time for approval of general and specific licences, causing delays and cost increases.	Changes in scope (e.g. changes in the cost of disposing of decommissioning waste because of the introduction of a clearance option.	Changes in scope are a highly sensitive factor, particularly regulatory or other legal requirements cause delay.	No experience in comparing estimated costs to actual costs.

Table 4.b **National cost estimation practices**

Country	Canada	Germany	Netherlands	Slovak Republic	United Kingdom	United States
Topics						
Organisation of costs	Most use a work breakdown structure for their cost estimates but not all follow the Yellow Book List of Cost Items:					
	Own WBS.	Own WBS.	At operator's discretion.	Uses Yellow Book	Own WBS.	At operator's discretion.
Cost estimating methodologies and tools, and their impact on the life cycle of the project.	National guidelines in most countries allow some degree of operator discretion as to choice of cost calculation method:					
	Licensees have discretion to use their own cost methodology, but requires review and acceptance by regulator	Owners/licensees have discretion to use their own cost methodology.	No specific information.	OMEGA code based on Yellow Book.	Owners/licensees have discretion to use their own cost methodology	Requires life cycle planning – suggests use of worst case financial scenarios for uncertainties.
Where estimates go wrong – review of experiences.	Decommissioning experience is extremely limited at this time in Canada.	Annual re-basing to minimise scope changes and regular updating of unit costs.	Changes in the decommissioning plan can have a major impact on costs.	No changes in scope anticipated at this time.	Annual re-basing to minimise scope changes.	Changes in cost are highly sensitive to scope changes.

155 NEA – Cost estimation for decommissioning – 2010

156 NEA – Cost estimation for decommissioning – 2010

2.8 Software for cost estimation

Cost estimation is a very important element of decommissioning planning. While preliminary decommissioning cost estimates could be done without specialised software, such software is required if reliable, informative, and dynamic results are to be obtained.

At present there is no universally accepted standard for software for decommissioning cost estimates, therefore most countries implement their cost methodology and associated computer software for estimating decommissioning costs [157].

Given the considerable increase in the number of nuclear facilities to be dismantled a few international initiatives has emerged. The IAEA developed a software tool, CERREX (Cost Estimation for Research Reactors in Excel) to be used for estimating costs for research reactor decommissioning. This software code is suitable for use on a wide range of facilities and structures and provides cost estimates according to ISDC format, which is a standardised format for comparison and validation[158].

CERREX was intended as a tool that would not require significant training or cost estimating experience; however, feedback indicated that significant support was required to enable its effective use. Currently, there is no international initiative regarding cost estimation for bigger facilities (e.g. NPPs). Considering these issues there might be a potential area for international collaboration.

2.8.1 Description of the already available solutions

Despite the lack of international initiatives, different proprietary software for cost estimation is available. Examples of available software is listed in alphabetic order.

AquilaCosting Software is the software that calculates the costs for decommissioning any type of nuclear facility (for example, power plants, research reactors, and laboratories). Aquila can be used in decommissioning planning to determine the final price for the final phase of the nuclear facility life-cycle i.e. for the transition period after final shutdown and for final decontamination, dismantling, demolition, site restoration, processing of all generated radioactive and non-radioactive waste and all related managing and supporting activities [159].

CORA-CALCOM is a database-supported program system for the planning of decommissioning and dismantling projects for nuclear facilities. It enables the creation of a project structure, the determination of costs, personnel expenses and the recording of components with integrated waste disposal planning [160].

In the United States, utilities have relied on consultants' computer codes written specifically for decommissioning cost estimating, such as the code DECCER. This code is a bottom-up code using unit

---

[157] OECD Nuclear Energy Agency, Cost estimation for decommissioning: an international overview of cost elements, estimation practices and reporting requirements, 2010.

[158] International Atomic Energy Agency, Data analysis and collection for costing of research reactor decommissioning, 2018.

[159] «AquilaCosting Decommissioning & Waste Management Costing Software», http://aquilacosting.com/about-the-software (accessed Jul. 09, 2020).

[160] CORA-CALCOM Database-supported program system for the planning and cost calculation of the decommissioning of nuclear facilities, http://www.siempelkamp-nis.com/fileadmin/media/Englisch/Download/NIS_PundD/CORA_CALCOM_en.pdf (accessed Jul. 09, 2020)

cost factors for estimating decontamination, removal, packaging, shipping and disposal. It also calculates management costs, undistributed costs, scrap, salvage, and contingency. [161]

DeCAT Pro (Decommissioning Cost Analysis Tool) allows users to easily evaluate the cost implications of various decommissioning options and to update these costs as facilities and equipment are removed or added to a site, and as waste disposal costs, labour rates and strategies change. This program provides detailed reporting for decommissioning funding requirements as well as providing detail project schedules, cash-flow projections, staffing levels, and waste volumes by waste classifications and types [162].

DECRAD (DECommissioning and RADiation) software is for planning, assessment of different strategies/alternatives and uncertainty analysis for decommissioning and radioactive waste management; it evaluates the radiological, technological and economical parameters of a system dismantling process. It is designed to support detailed dismantling projects of separate buildings and entire nuclear facilities as well as to plan integrated waste disposal for nuclear facilities [163].

DEXUS consists of a database system, evaluation and optimisation system, VR and visualisation system, and data management system. The CAD data, volumetric data and activity inventory data is input to the visualisation system to show the complex structure and radioactivity. The data is also related to the decommissioning plan evaluation and optimisation system. The output of the decommissioning plan is also visualised by the system. Moreover, an advanced simulation such as interference checking during the dismantling plan is included. This means that a more precise evaluation of workload, simulation of workers and safety check at the dismantling plan by using virtual reality technology is expected and the result can be reflected on the evaluation system. These sub-systems are used for the planning stage of a decommissioning project [164].

ETE EVAL software uses approximately 20 standard scenarios for which lists of operations have been defined. The ratios (labour hours, dose, waste produced) assigned to these operations are then applied to the parameters of the facility. Virtually all existing CEA facilities have been assessed this way for budgetary purposes. Preliminary design studies are used before initiating a decommissioning project, as they are based on the intended strategy, site-specific facility information, and organisational requirements. The costs are broken down into categories of work, subcontractor supervision (with and without the prime contractor), waste, contracting authority, and operation of the facility[165].

OMEGA is, used in Slovakia, for cost estimating follows the sequence of preparation of the inventory database (in three parts – technology inventory, building inventory, and radiological parameters), preparation of the database of technical and economic parameters (unit cost factors), creating the

[161] OECD Nuclear Energy Agency, Cost estimation for decommissioning: an international overview of cost elements, estimation practices and reporting requirements, 2010.
[162] Decommissioning Cost Estimation Software: Radiation Safety & Control Services." http://www.radsafety.com/decommissioning-cost-estimation-software.php (accessed Jul. 10, 2020).
[163] ŠIMONIS A., A. SIRVYDAS, G. POŠKAS. The software DECRAD justification report. LEI report TA-14-13.10, 2010
[164] Y. Iguchi, Y. Kanehira, M. Tachibana, and T. Johnsen, "Development of Decommissioning Engineering Support System (DEXUS of the Fugen Nuclear Power Station," Journal of Nuclear Science and Technology, vol. 41, no. 3, pp. 367–375, Mar. 2004, doi: 10.1080/18811248.2004.9715497.
[165] OECD Nuclear Energy Agency, Cost estimation for decommissioning: an international overview of cost elements, estimation practices and reporting requirements, 2010.

calculation options with the OMEGA code (based on the selected strategy), optimisation of the calculations (critical path rescheduling and resource loading), and final review and presentation of selected options[166].

The successor of OMEGA is eOMEGA software. The basic idea behind the eOMEGA development is to merge the advantages of two existing matured solutions: the decommissioning costing software OMEGA and the web-based ADIOS platform with tools and processes to implement any web-based software solution with user-friendly interface. Moreover, the eOMEGA costing platform is intended to be one of the modules of the universal eOMEGA platform which also covers the other activities within the back end of nuclear power engineering. The methodology for decommissioning cost calculation in the eOMEGA is in line with the international recommendations and best practices, is completely based on the ISDC and is applicable for any type of nuclear facility. Previous experience with development of decommissioning inventory data, determination of input data, calculation of waste management parameters using a unique system simulating the material and radioactivity flow were implemented to the eOMEGA. Main goal of the eOMEGA costing platform is to provide flexible, user-friendly and internet-based tool for transparent decommissioning costing in the ISDC format for any stakeholders involved in the preparation of decommissioning projects. Costing cases may be graded from preliminary up to detailed costing at the level of final decommissioning planning. Access by any authorities/stakeholders may facilitate building confidence in the decommissioning cost estimates and understanding the cost elements [167].

[166] OECD Nuclear Energy Agency, Cost estimation for decommissioning: an international overview of cost elements, estimation practices and reporting requirements, 2010.
[167] Zachar, M., Daniska, V., Hrncir, T., Daniska, D., & Zubcak, P. (2016). eOMEGA - ISDC decommissioning costing platform. Collaborative solutions for current & future trends 3rd Canadian conference on nuclear waste management, decommissioning and environmental restoration, p. 555, Canada: Canadian Nuclear Society, 2018.

2.9 Development of mechanisms for cost benchmarking

In recent years considering the number of nuclear power plants entering the decommissioning phase, the need of developing more reliable and accurate decommissioning cost estimate is increased significantly in the nuclear industry and the interest in cost benchmarking is growing consequently.

This higher interest in 'benchmarking' applied to decommissioning costing reflects increasing attention on understanding variations between cost estimates and apparent escalation of decommissioning costs. A related interest is in improving the performance and ensuring value-for-money in the delivery of decommissioning projects and services.

In comparison with other industry sectors, such as oil and gas, where cost benchmarking approach is more mature, nuclear decommissioning industry need to expand and improve this process. A key element is the awareness within nuclear industry of potential added value of developing and implementing cost benchmarking in the context of nuclear decommissioning. Benchmarking would provide data for wide groups of stakeholders that have different interests adding value in a variety of ways, such as for example the need for authorities and regulators to establish financing requirements or the need of operators to increase cost reliability and reduce cost overruns.

Developing and implementing benchmarking approach in decommissioning cost estimates means a systematic analysis of estimated vs. actual cost/other estimates, identifying relationship between values, and collecting and comparing all the information on which estimates are developed and cost is incurred.

In general, application of cost benchmarking approach in the estimating phase represent an opportunity for cost savings in relation to future projects, applying the lessons learnt from previous projects.

Cost benchmarking could identify key cost drivers and how they affect the current cost estimates as well.

To date a number of challenges, need to be addressed before benchmarking approaches for NPP decommissioning costing can be practically implemented. These challenges arise in part because the key relevant project and cost data currently is not readily available. In the context of nuclear power plant decommissioning, the problem here is two-fold:

- Firstly, there is a heavy dependence on cost estimates rather than actual cost data because of the relatively limited experience in actual NPP decommissioning; and
- Secondly, where there is actual experience, the access to actual decommissioning project cost data is limited, not least of all because of strong sensitivity around sharing of such data.

In 2017, the NEA’s Decommissioning Cost Estimation Group (DECG) launched a project on aspects of benchmarking in decommissioning costing with the aim of:

- Identifying possible benchmarking approaches and discussing their specific application to decommissioning costing, including the ‘added value’ in developing decommissioning cost benchmarking approaches;

- Discussing prevailing barriers to the necessary information and data sharing required for decommissioning cost benchmarking and exploring what is needed to facilitate development of benchmarking approaches in nuclear decommissioning costing (a possible 'road map'); and
- Possible benchmarking exercises or case studies in order to develop and illustrate decommissioning costing benchmarking concepts and methodologies, if suitable cost estimate and actual cost data is made available.

The results of the work conducted by the NEA group are summarised in the report "Cost Benchmarking for Nuclear Power Plant Decommissioning"[168].

2.9.1 Description of the mechanisms for cost benchmarking

Benchmarking may be accomplished by several methods including:

- comparisons with other studies;
- comparisons to actual field experiences;
- comparisons to decommissioning costing formulae.

The selected method for, and the quality of, the comparison will depend on the quality of the information available and the degree of detail provided for comparison. When possible, all three methods should be used.

Comparison with other studies is the most direct method for experienced estimators to validate the cost and schedule estimates. Generally, estimating consulting companies have an inventory of previous estimates that were prepared for other clients and can review those estimates against the current estimates.

Other applicant/licensee estimators may have to rely on published information in literature, papers presented at conferences, or handbooks.

When making comparisons to other studies it is important to ensure the baseline estimates conform to the same assumptions and boundary conditions as in the estimate under review, or to be aware of how any differences in these may impact on the estimates being compared. The basis of estimate for both studies must be compared in detail and any differences noted for the comparison.

[168] NEA - Cost Benchmarking for Nuclear Power Plant Decommissioning 2019

Table 2.9-1 . Summary of the methodologies for cost benchmarking.

Methodology	What is working	What is missing
comparisons with other studies	It is the most direct method for validate cost estimates. It has a larger availability of data than other methodologies/approaches.	The context where studies are performed (such as, assumptions, boundary conditions) affects the comparison. Any difference in basis should be noted and taken into account.
comparisons to actual field experiences	Cost are compared to those of actual decommissioning project experienced by operators Comparison between estimate cost and actual cost from other project is an opportunity to consider in the estimates the lesson learned from the field.	Completed project cost data is currently very limited. Some costs of dismantling work have been reported in the literature, but in general they contain only summary level cost data. Actual costs could be affected by issues: the type of contracts used to accomplish the decommissioning work, specific elements of decommissioning costs are handled differently in different countries in accordance with national policies, low breakdown of actual costs (total costs and few major cost elements)
comparisons to decommissioning costing formulae	Costs of some activities could be estimated using formulae, that required availability of inventory and unit cost factor It is a fast way for cost estimate	Inventory and unit cost factor could vary due to the site-specific information. Any difference in basis should be noted and taken into account.

The work carried out by the NEA's Decommissioning Cost Estimation Group (DCEG) published in the report [169] analysed the more developed cost benchmarking practices in other industries to evaluate the useful models for developing cost benchmarking for NPP decommissioning.

The examples discussed in the report highlight two broadly different approaches that might be taken in cost benchmarking:

a. Project level: a comparison of overall cost and scheduling for one project with the overall cost and scheduling for another. This technique is offered by Independent Project Analysis Global (IPA Inc.) for oil and gas and process industries, and by the FMI for shipbuilding. The approaches are typically aimed at informing the decision executive, as well as the authority and regulator stakeholder groups.
b. Activity level: a comparison at the line-item level within a project's cost and schedule estimate – i.e. comparison of one activity of the project against a similar activity in another project – and

[169] NEA - Cost Benchmarking for Nuclear Power Plant Decommissioning 2019

aggregation of the results. Turner and Townsend, the ICMS, IPA Inc. and the FMI offer this type of analysis.

The use of standard, work breakdown structures, cost breakdown structures and product breakdown structures greatly simplifies cost benchmarking. The international cost benchmarking approaches are typically supported by a common cost-reporting structure.

In case of NPP decommissioning, the ISDC [170] provides a standardized list of decommissioning cost items with a hierarchical common reporting structure capable of allowing the comparison of the costs themselves. The first and second levels being aggregations of the basic activities identified on the third level. The cost associated with each activity may be subdivided according to four cost categories (see Figure 2.9-1).

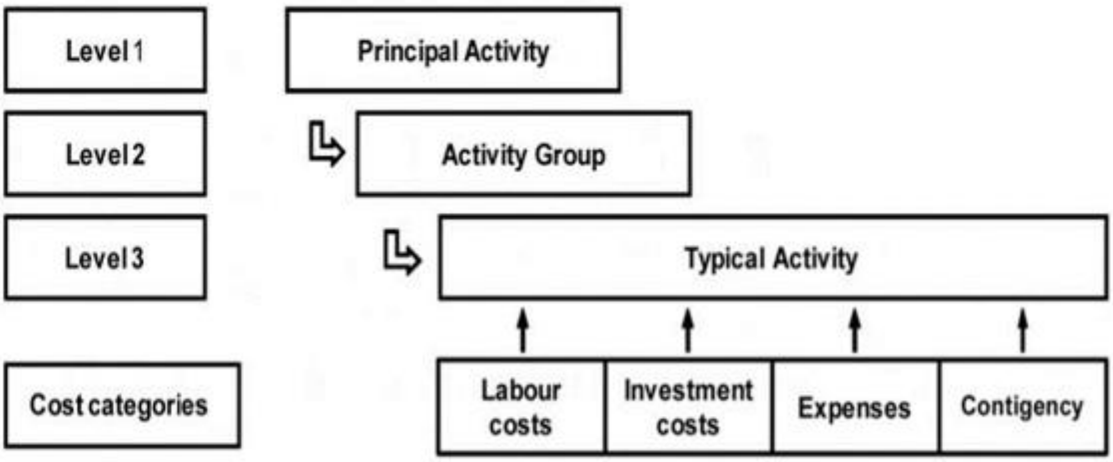


Figure 2.9-1 Cost categories

The items of the highest level of aggregation, Level 1 - Principal Activity, are listed in the following:

01 – Pre-decommissioning actions.

02 – Facility shutdown activities.

03 – Additional activities for safe enclosure and entombment.

04 – Dismantling activities within the controlled area.

05 – Waste processing, storage and disposal.

06 – Site infrastructure and operation.

07 – Conventional dismantling, demolition and site restoration.

08 – Project management, engineering and support.

09 – Research and development.

---

[170] NEA - International Structure for Decommissioning Costing (ISDC) for Nuclear Installations - 2012

10 – Fuel and nuclear material.

11 – Miscellaneous expenditures.

The ISDC was developed primarily as the international standardised structure to present costs for NPP decommissioning projects, but it would be suitable for use in the normalisation of data in the cost benchmarking context but could be also adapted to the decommissioning of other types of nuclear facilities.
Barriers to the introduction of cost benchmarking in the decommissioning of nuclear power plants (NPPs) are discussed in the NEA report [171] and they mainly result from a lack of collected data on actual costs, as well as perceived obstacles to sharing this data with others.

In order to overcome these barriers, it will be necessary to address the apparent absence of an appropriate organisation within the nuclear industry, which will ultimately be needed to enable cost benchmarking and to facilitate the sharing of data.

2.9.2 Experiences/Case studies

Examples of experiences in the use of the ISDC for cost benchmarking were presented, for example, in Costs of Decommissioning Nuclear Power Plants [172], which also includes some principles and results of conversions from different cost estimation presentation formats into the ISDC.

Although the ISDC was developed in a way that could be adapted to the decommissioning of other types of facilities, only the IAEA has done so at the international level for research reactors, in the context of the DACCORD programme (IAEA, 2017).

To date, there are no international agreed cost structures for the decommissioning of other types of nuclear facilities, or for the provision of radioactive waste infrastructure.

Another example of the use of the ISDC structure was to present the total cost and contingencies for the whole decommissioning phase is reported in the Decommissioning study of Forsmark NPP [173]

[171] NEA - Cost Benchmarking for Nuclear Power Plant Decommissioning 2019
[172] NEA - Costs of Decommissioning Nuclear Power Plants – 2016
[173] Westinghouse Electric Sweden AB - Decommissioning study of Forsmark NPP - June 2013

2.10 Methods and tools for sensitivity and uncertainty analysis in cost estimation

Sensitivity and uncertainty analysis is a key element of cost estimation process and it is commonly agreed that uncertainties and risks need to be evaluated in decommissioning projects and in related cost estimates. Over the years, a number of different approaches have been developed to perform uncertainty analyses and incorporate the results into decommissioning cost estimates. The ISDC [174], published in 2012, provides a useful common reporting format for decommissioning costing. It builds up a deterministic estimate starting from the scope and assumptions set out in a detailed basis of estimate (BoE) including the reporting of contingency within the defined project scope as part of the project baseline estimate. However, even if ISDC may be used as a good foundation for cost calculations relating to out-of-scope uncertainties, the ISDC itself does not address probabilistic methods or associated presentation formats for their inclusion in decommissioning estimates.

In 2014 the NEA and IAEA initiated a joint project to facilitate preparation and presentation of nuclear decommissioning cost estimates, complementing the ISDC cost presentation format, describing approaches to estimating uncertainty and to risk analysis. The project concluded in 2016 and the results are published in the report [175]. The report describes the different elements of a decommissioning cost estimate and provides suggestions for incorporating and presenting uncertainty and risk in a way that is compatible with the International Structure for Decommissioning Costing (ISDC). It offers an approach to treating uncertainties reflecting current good practices in cost estimating. Specifically, it describes how uncertainties in decommissioning cost estimation can be addressed using standardised methods of estimating uncertainty and risk analysis. Its recommendations aim at enabling better consistency of application of the treatment of risk and uncertainty in the preparation of decommissioning cost estimates.

In general there are a wide variety of approaches for presenting the different elements of a cost estimate currently in use, and the details may differ considerably between countries, organisations and estimators. For the purpose of the document, we refer to the same approach and key terms definition as presented in the aforementioned report [176] considering such definition is very useful for the reader and that is as much as possible consistent with the ISDC.

Despite the multiple variations, all the several approaches necessarily have some core elements in common. These common elements are illustrated in the following Figure 2.10-1 and Figure 2.10-2 [177]

[174] OECD/NEA (2012) - International Structure for Decommissioning Costing (ISDC) for Nuclear Installations
[175] OECD/NEA (2017) - Addressing Uncertainties in Cost Estimates for Decommissioning Nuclear Facilities
[176] OECD/NEA (2017) - Addressing Uncertainties in Cost Estimates for Decommissioning Nuclear Facilities
[177] OECD/NEA (2017) - Addressing Uncertainties in Cost Estimates for Decommissioning Nuclear Facilities

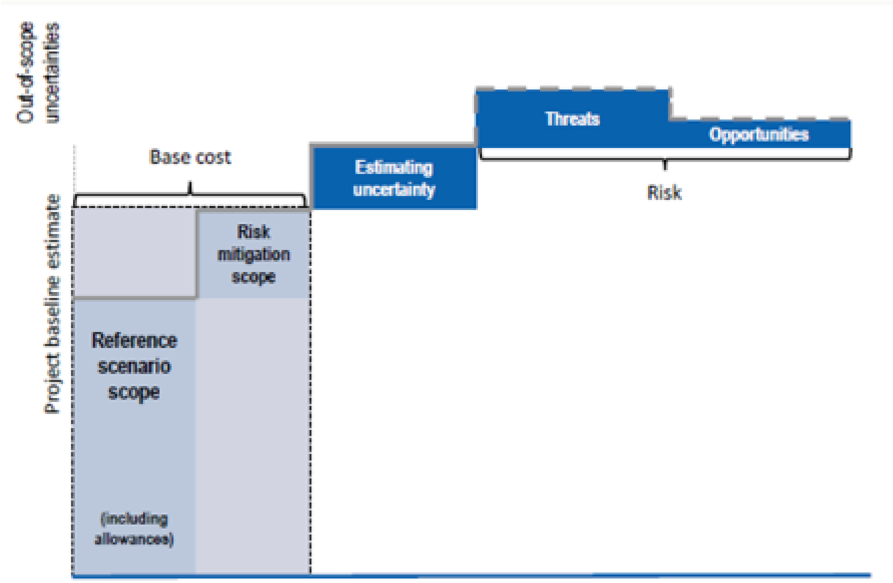


Figure 2.10-1 Generic example of the elements of a cost estimate

The Figure 2.10-1 shows the cumulative impacts as cost components are added or subtracted. Such a presentation is useful for understanding how an initial value (for example, the reference base cost) is affected by a series of positive and negative cost elements.

The following Figure 2.10-2 shows the basic elements of a cost estimate. This is an example based on the ISDC structure, with risk elements added:

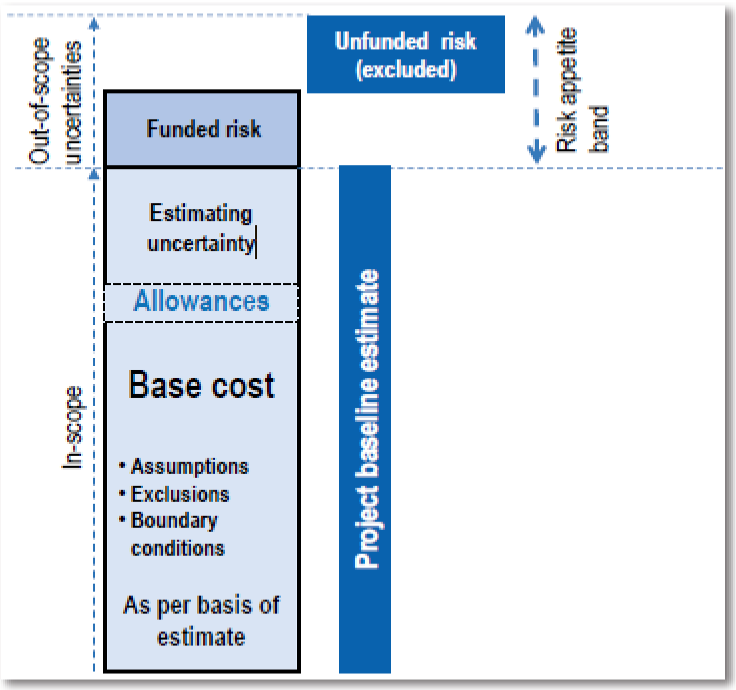


Figure 2.10-2 Basic elements of a cost estimate with risk elements added

In particular it is important to mention the definition of following terms:

The "*Base cost*" is estimated cost of the base scope of the project as defined by the BoE, without any provision for estimating uncertainty or out-of-scope uncertainties.

The "*risk mitigation scope*" derives from an iterative process of scope refinement or optimising of the initial project scenario. Where an initial assessment reveals potential events and outcomes that may be seen as intolerable or undesirable, these may be better dealt with by adding appropriate risk mitigation scope to the original base scope, rather than by being addressed separately as potential out-of-scope risks.

The "*Estimating uncertainty*" is a provision for uncertainties that are associated with the defined project scope (i.e. are considered to be in-scope), as identified by the BoE, and are part of the project baseline estimate. Specifically, this is a provision for uncertainties associated with conduct of work under other than the ideal (theoretical) conditions used to derive the project base cost. Within ISDC, this is referred to as the "contingency" and it is assumed to be fully spent during execution of the project.

The "*Project baseline estimate*" is the estimated cost of the base scope of the project as defined by the BoE, including provision for the estimating uncertainty. It excludes provision for any risks considered beyond the defined project scope, but includes any added risk mitigation scope.

The "*Out-of-scope uncertainties*" are the uncertainties which lie above the project baseline estimate as they are considered beyond the defined project scope. In this report, these are referred to as risk. Out-of-scope uncertainties can be funded or remain unfunded.

Putting all elements together allows the production and presentation of a cost estimate that is able to integrate treatment of issues of scope maturity, uncertainty within the defined project scope, and out-of-scope risk. This integrated approach is illustrated in the following Figure 2.10-3 [178].

[178] OECD/NEA (2017) - Addressing Uncertainties in Cost Estimates for Decommissioning Nuclear Facilities

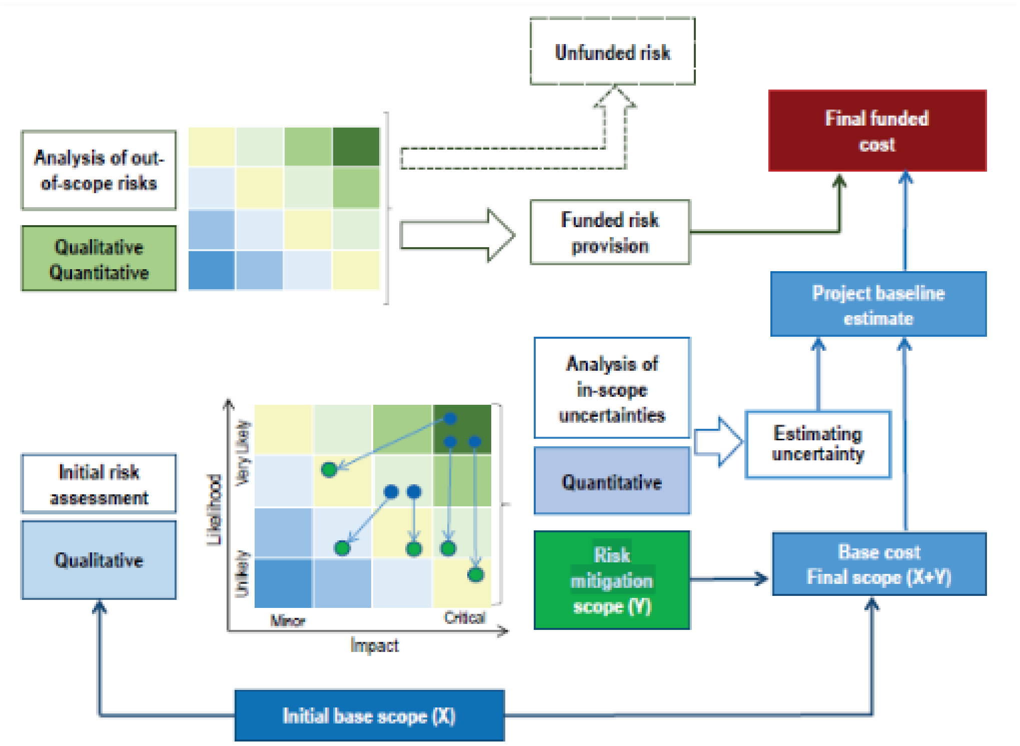


Figure 2.10-3 Final funded cost, including addition of a funded risk provision to the project baseline estimate

2.10.1 Description of the methodologies and tools

Internationally there are many methodologies and tools to perform sensitivity analysis and calculate uncertainties.

In general, sensitivity analysis is the study of how the output of a mathematical model or system (numerical or otherwise) can be related to variances in its input variables. By means of this analysis, insight is provided into how and to what extent changes in particular variables may influence the model outputs. The BoE is designed around a set of boundary conditions that defines what work packages are to be produced. This is based on a single reference scenario and results in a project baseline estimate cost covering the in-scope elements. Cost modelling can therefore be used to conduct a sensitivity analysis of the project baseline estimate to particular input parameters such as labour rates or waste package disposition costs. By changing key parameters one at a time, this will reveal what the key cost drivers are and enable more analysis of options, opportunities and risk mitigation. For example, a sensitivity analysis for an NPP decommissioning project might consider variation of key input parameters related to alternative arrangements for inter alia:

- organisational transition activities (operations to decommissioning);
- make versus buy (supply chain) decisions;
- waste packaging, treatment, storage optimisation and accessibility of waste disposition routes;
- critical path analysis of the reactor and any other decommissioning schedule optimisation;
- labour costs for different staffing options.

A further sensitivity analysis can be conducted specific to risks and out-of-scope events. In this case, a useful insight might be obtained as to where it may be possible to prioritise effort to reduce risk and hence reduce cost and schedule durations associated with the baseline.

With reference to uncertainties analysis, the process foresee the calculation of *estimating uncertainty* and *out-of-scope uncertainty*.

Estimating uncertainty - provision for uncertainties that are associated with the defined project scope:

The *estimating uncertainty* value may be derived deterministically, by percentages or by probabilistic means where a Monte Carlo analysis is performed or by combination of these techniques as described in [179].

Estimating uncertainty is determined through analyses of the input data, e.g. physical, radiological, decommissioning process and economic parameters. Estimating uncertainty does not include any provision for potential scope change from external factors (out-of-scope), such as impacts of changing regulations, major design or project scope changes, catastrophic events (force majeure), labour strikes, variation in site conditions (expected vs. actual), or external project funding (financial) limitations. These need to be separately considered as risks beyond the defined project scope for which additional provision (funded risk) might be required.

Several approaches may be used for calculating the estimating uncertainty:

Application to an entire decommissioning project

The estimating uncertainty is defined on the basis of expert judgement or by means of other methods as a percentage of the base cost. This methodology is based on the practical experience of the estimator and the estimating uncertainty derived in this manner may itself be quite variable from estimator to estimator.

This approach has been used historically because of limited experience of using analytical methods for calculating estimating uncertainty provisions. Typical contingency levels may range from -5% to +15% for detailed cost estimates made at the outset of decommissioning projects, to -15% to +30% for preliminary cost estimates [180].

Application to groups of decommissioning activities

This approach reflects the fact that different uncertainties in input data can be identified for different groups of activities in the decommissioning process. These uncertainties may also have different weight of influence. For example, separate percentages for estimating uncertainty for highly radioactive component removal (reactor vessel and internals), lower percentages for less radioactive piping, components and building demolition. Where this approach is followed, estimating uncertainty for individual activities may vary widely, depending on the specific methodology used, the experience of the estimator and the groupings of the activities. Following this approach individual activity

[179] OECD/NEA (2012) - International Structure for Decommissioning Costing (ISDC) for Nuclear Installations
[180] OECD/NEA (2010) – Cost estimation for decommissioning: An International Overview of Cost Elements, Estimation Practices and Reporting Requirements

uncertainties may range from 10% to 75%, depending on the degree of difficulty judged to be appropriate by the cost estimator [181].

This is an elaboration of the first approach and offers a greater level of detail. However, it is subject to similar limitations as the first approach.

Application to individual decommissioning activities

The basis of this approach is an application of estimating uncertainty for each decommissioning activity on a line item basis using a bottom-up estimating method.

This approach ensures consideration of individual conditions and characteristics of specific decommissioning activities. However, this approach requires considerable effort, detailed information and knowledge about the planned decommissioning project.

Application of estimating uncertainty for each decommissioning activity on a line item basis using a bottom-up estimating method is the most detailed of the three approaches.

As reported above whatever methodology is used, it is important to fully document the approach applied to calculate the estimating uncertainty provision for a better understanding and confidence in the overall cost estimate.

Cut-off scope uncertainties - provision for risks beyond the defined project scope:

For uncertainties beyond the defined project scope, risks can be either funded or unfunded and it depends by the risk appetite. The risk appetite is the factor that determines which point in a range of cost outcomes is to be used for the funded risk provision.

Also for these kind of uncertainties, provisions can be calculated using either a deterministic approach, or by using a probabilistic approach.

It should be noted that a prerequisite of addressing risk beyond the defined project scope requires understanding of the BoE, assumptions and exclusions; and how estimating uncertainties within the defined project scope have been treated.

The analysis of out-of-scope risk and how this can be used to derive a cost provision in the final cost estimate is made of 4 stages according to [182]:

- Stage A - Risk review and prompt lists: the process to identify and explore the risks, examine all risk events qualitatively to determine the low, medium and high-risk events.
- Stage B - Opportunity review: the process to identify beneficial outcomes (opportunities) that are also generated in the risk analysis.
- Stage C - Generating scenarios: The development and use of different scenarios is important but is an optional step.

[181] OECD/NEA (2010) – Cost estimation for decommissioning: An International Overview of Cost Elements, Estimation Practices and Reporting Requirements
[182] OECD/NEA (2017) - Addressing Uncertainties in Cost Estimates for Decommissioning Nuclear Facilities

- Stage D - Out-of-scope risk analysis: The starting point for the analysis is a risk (and opportunity) register with impacts and probabilities that have been reviewed, agreed and assigned. These risks can impact on cost, or schedule or both.
- A number of analysis approaches may be applied or a combination of these techniques:

Option 1 – Probabilistic risk assessment

The benefit of a probabilistic approach is that it should facilitate conduct of an unbiased analysis. The quality of the output data is however greatly dependent on the input data and for that reason it is often not a recommended approach for projects of low scope definition and maturity. A further issue is how this method treats high-impact low probability events and whether these events may dominate the s-curve (cumulative probability) shape and in doing so mask more realistic outcomes.

Option 2 – Apply a factor tied to past experience of similar activities

The nuclear decommissioning industry has a limited history of recent nuclear power plant (NPP) dismantling work. Completed project cost data is currently very limited. Some costs of dismantling work have been reported in the literature, but in general they contain only summary level cost data. As the NPP decommissioning industry matures, it is anticipated that more of this detailed information will become available.

Option 3 – National factors (custom and practice)

For some project types and for some countries it is normal to apply a multiplying factor or lump sum additional provision to cover the costs associated with out-of-scope uncertainties, e.g. the addition of a 20% or 30% uplift. This is typically done to address issues of potential funding shortfalls and/or experience of project cost overruns, for example relating to incomplete or immature definition of scope, and where there is a limited basis for the detailed analysis of specific risks. However, as this approach is likely to be inaccurate and misleading as to project outcomes since it is arbitrary and not related to an analysis of the specific risks that may impact the project. Accordingly, this practice is being phased out in most countries, and is being replaced by use of appropriate specific risk analyses.

Option 4 – Qualitative risk assessment (step1)

Qualitative risk analysis is the process of assessing and combining risk probability of occurrence and impact; in this kind of analysis risks are “manually” classified by raw types of impacts and probability (and not by really computed values with respect to the whole project and the side effects of other risks).

Option 5 – Quantitative risk analysis (step2)

The main objective of the quantitative risk analysis is to appraise the cost value coming from negative risk or the revenue of positive risk. Not all the identified risks are considered in the quantitative analysis, and the focus is on strategic risks. A first selection is done by the qualitative analysis results. Within this step the risk effect is costed as the calculated expected monetary value (EMV), i.e. the expected monetary value of the risk: EMV = cost impact × probability.

According to risk appetite and risks analysed, estimator will determinate which risks are to be categorised as funded risk and which will remain as unfunded risk. The additional cost provision for funded risk above the project baseline estimate can be included in the estimate to yield a final funded cost.

Once all these evaluations have been performed the cost estimates should therefore contain a number of specific elements:

- the cost as it relates to the "base scope";
- the estimating uncertainty;
- the funded risk.

Table 2.10-1 Summary of the methodologies for sensitivity and uncertainty analysis in cost estimation

Methodology/Tool	**What is working**	**What is missing**
Sensitivity analysis (one-factor-at-a-time [OFAT])	It identifies key input parameter which have greater impact on total cost.	Does not offer insights into simultaneous changes of input parameter Does not result in a distribution representing potential range of costs (it does not look at the probability)
Probabilistic risk assessment	Probabilistic risk assessment should facilitate conduct of an unbiased analysis. Calculates probability density functions (PDF) and cumulative distribution functions (CDF)	Outputs is affected by input data, so this technique is not recommended for project with low scope definition and maturity. Needs to identify parameters by the use of historical data or expert opinion
Deterministic: Apply a factor tied to past experience of similar activities	It is based on actual costs and field experiences. It could be faster than other methodologies.	Limited quantity of data arising from completed projects and that in general contain only summary level cost data.
Deterministic: National factors (custom and practice)	It is useful where there is a limited basis for the detailed analysis of specific risks. It could be faster than other methodologies.	This approach is likely to be inaccurate and misleading as to project outcomes
Qualitative risk assessment	Good for screening level assessments when comparing/screening multiple alternatives or for when sufficient data is not available, or as first analysis step.	Results are based on subjective measures. Possible side effects are not taken into account.
Quantitative risk analysis	Able to calculate the cost value coming from negative risk or the revenue of positive risk. Detailed approach and results.	Not all the identified risks are considered in the quantitative analysis, and the focus is on strategic risks. Require large amount of historical information.

2.10.2 Experiences/Case studies

The IAEA has developed a risk analysis framework specifically for risk management in decommissioning projects in the context of a project on Risk Management for Decommissioning (DRiMa). This is based on the IAEA Safety Standards on Decommissioning of Facilities (IAEA, 2014) and ISO Standards on risk management (ISO, 2009a, 2009b and 2009c). The project provides recommendations on how to perform such risk analyses by considering the decommissioning plans, from initial to final versions, up to the implementation of the decommissioning and dismantling actions.

In 2017 was published the joint report "Addressing Uncertainties in Cost Estimates for Decommissioning Nuclear Facilities" by the Nuclear Energy Agency and the International Atomic Energy Agency. The purpose of this report is to describe the treatment and presentation of uncertainty and risk in nuclear decommissioning cost estimates, based on experience in participating countries and current good practices. In Appendix A, a case study "Calculating the final funded cost for a reactor decommissioning project" is provided.

In the frame of "Data Analysis and Collection for Costing of Research Reactor Decommissioning" (DACCORD) project, launched in 2012 by IAEA, was performed a simplified sensitivity analysis of the total decommissioning cost to different five input parameters: labour rate, inventory, duration (ISDC 06 & 08), waste management unit factors and decommissioning unit factors. Results were published in project report[183].

[183] IAEA (2017), Data Analysis and Collection for Costing of Research Reactor Decommissioning, TECDOC- 1832, IAEA, Vienna, www-pub.iaea.org/MTCD/Publications/PDF/TE1832_web.pdf.

3. Human resources management

With the large amount of plant shutdowns approaching, there is a great need for staff with adequate competence to handle the upcoming decommissioning projects. At the same time, having enough competence with the right quality is seen as a major challenge for the industry. In a survey, 200 nuclear experts in the international decommissioning industry were asked about the biggest challenge for the nuclear decommissioning sector. The most frequent answer (37% of respondents) was "lack of trained and qualified staff"[184].

Competence needs will vary across different decommissioning phases and it is important that the right competence is available at the right time. IAEA's general safety requirements for decommissioning state that individuals performing decommissioning actions shall have the necessary skills, expertise and training to perform decommissioning safely. And it is the licensee's responsibility to ensure that staff are properly trained and qualified[185]. The skills, knowledge and attitudes developed and used during operation of nuclear facilities will not be sufficient for decommissioning those same facilities. This means that specific decommissioning training, or hiring of external staff, is needed to ensure staff have the right competence for the decommissioning phase.

International initiatives

International initiatives related to Organisation models:

- OECD NEA No. 7374 (2018): organisation transition from operations to decommissioning
  The objective of this report is thus to inform decision makers in decommissioning, decommissioning planners, as well as regulatory bodies, on the key aspects of such a “transition” and of preparing for decommissioning during the last years of operation and after cessation of operation.

- EU EUR 27902 EN (2015): decommissioning & environmental remediation, resources per D&ER stage
  The document aims to support the development of adequate policies in IAEA Member States for decommissioning and environmental remediation, addressing in essence the following three fundamental questions:
  - What are the motivations for implementing decommissioning and environmental remediation?
  - What are the main constraints hindering progress of decommissioning and environmental remediation programmes?
  - What are the solutions for overcoming these constraints, taking account of experience from programmes under implementation?
  This document was prepared in close collaboration between the European Commission's Joint Research Centre and the IAEA.

[184] Nuclear Energy Insider (NEI) (2012). UK & Europe Nuclear Decommissioning Market Survey 2012
[185] International Atomic Energy Agency (IAEA) (2014). Decommissioning of facilities. General Safety Requirements Part 6. Vienna, Austria: International Atomic Energy Agency

- EU IP/D/CONT/IC/2013_054 (2013): best practices for decommissioning The study identified best practices in the organisation of the decommissioning projects in Germany and France. The comparison with the three eastern European countries identified several areas where the process organisation should be urgently improved and a clearer attribution of responsibilities is required.

- UK URN - 12D / 436 (2012): nuclear supply chain constellations
  The Nuclear Supply Chain Action Plan sets out a series of actions to provide greater clarity about the forward pipeline of both public and private procurement contracts in the nuclear sector. It includes a commitment by the NDA and the majority of its key contractors to use Contracts Finder to provide details of major contracts.

- IAEA TECDOC 1394 (2004): stakeholders & organisational issues *Planning, managing and organising the decommissioning of nuclear facilities: lessons learned*
  The objective of this document is to encourage the development and improvement of decommissioning planning and management techniques. The focus is on organisational aspects, to reduce the duplication of effort by various parties through the transfer of experience and know-how and to provide useful information for those Member States planning or implementing decommissioning projects. The document summarises the reported experience in the planning and management of decommissioning. It is particularly aimed at decision makers, plant operators, contractors, and regulators involved in the planning and management of decommissioning activities. This is particularly applicable to nuclear installations, which are approaching the end of their operating lives.

- IAEA TRS 399 (2000): organisational adaptation per decommissioning stage LINDER International initiatives related to Knowledge management
  The report covers organisational aspects of decommissioning and describes factors relevant to the planning and management of a decommissioning project. It identifies the general issues to be addressed and provides an overview of organisational activities necessary to manage a decommissioning project in a safe, timely and cost-effective manner.

International initiatives related to Knowledge management

- IAEA: knowledge management, competence for contractor personnel, plant information models (2017, 2020)
  The first publication[186] provides a methodology to enable knowledge loss risk management to ensure safe, reliable and efficient operation of nuclear facilities. It focuses on aspects of knowledge loss risks associated with employee attrition and provides guidance to mitigate them. The publication also provides examples of best practices (case studies) of effective knowledge loss risk management gathered from the nuclear power plants and nuclear related organisations.

[186] IAEA Nuclear Energy Series NG-T-6.11 Knowledge Loss Risk Management in Nuclear Organisations

The purpose of the second publication[187] is to provide an overview of PIMs, emphasize the importance of their application in support and management of design knowledge throughout the nuclear power plant life cycle and present an overview of a knowledge-centric plant information model that builds on the basic concept of a PIM. The target users of this publication are decision-making organisations in Member States having experience with nuclear power programmes and those embarking on new nuclear power programmes.

- Hitachi-GE Nuclear Energy, Ltd. (Hitachi-GE) used an in-house survey and interviews conducted in 2017 to identify the primary challenges surrounding technical knowledge transfer in its internal operational reforms[188]. The results were published, and the article describes the knowledge management initiative that has been undertaken since 2017 on the basis of "transferring knowledge from person to person," "connecting people," and "connecting people with information," and the plans for the future.

- KM support system for nuclear decommissioning in Japan[189]
  The decommissioning of a nuclear facility is a long term project, handling information which begins from the design, construction and operation. Moreover, the decommissioning project is likely to be extended because of the lack of the waste disposal site especially in Japan. In this situation, because the transfer of knowledge and education to the next generation is a crucial issue, integration and implementation of a system for knowledge management is necessary in order to solve it.

- The H2020 ENEN PLUS (ENEN+) Project has the primary motivation to substantially contribute to the revival of the interest of young generations in the careers in nuclear sector. This is to be achieved by pursuing the following main objectives:
  - Attract new talents to careers in nuclear.
  - Develop the attracted talents beyond academic curricula.
  - Increase the retention of attracted talents in nuclear careers.
  - Involve the nuclear stakeholders within EU and beyond.
  - Sustain the revived interest for nuclear careers.

- The European Human Resources Observatory for the Nuclear Sector (EHRO-N) was established in 2011, in the framework of the EURATOM treaty, to determine the situation of nuclear-educated Human Resources, skills and competences in Europe, assess the trends and suggest policy options for improvement. Since its launch, EHRO-N has become a widely recognised and appreciated instrument and role model, inspiring a vast number of interested stakeholders. Its mission is to provide qualified data on human resources needs in the nuclear field within the European Union and high-level expert recommendations on EU-wide nuclear Education and Training action, thus promoting lifelong learning and cross border mobility.

---

[187] IAEA-TECDOC-1919 Application of Plant Information Models to Manage Design Knowledge through the Nuclear Power Plant Life Cycle

[188] Hitachi GE Vol. 69, No. 4 (2020) Knowledge Management for Transferring Nuclear Industry Technical Knowledge to Next Generation

[189] Yukihiro Iguchi, Satoshi Yanagihara ICONE23-1004 INTEGRATION OF KNOWLEDGE MANAGEMENT SYSTEM FOR THE DECOMMISSIONING OF NUCLEAR FACILITIES

- The European credit system for vocational education and training (ECVET) is a powerful tool for increasing cross-border cooperation in education and training. The aim of the European Credit system for Vocational Education and Training (ECVET) is to:
    - make it easier for people to get validation and recognition of work-related skills and knowledge acquired in different systems and countries – so that they can count towards vocational qualifications
    - make it more attractive to move between different countries and learning environments
    - increase the compatibility between the different vocational education and training (VET) systems in place across Europe, and the qualifications they offer
    - increase the employability of VET graduates and the confidence of employers that each VET qualification requires specific skills and knowledge.

International initiatives related to General education for decommissioning

- ELINDER presents a modular, coherent and commonly qualified training programme in nuclear decommissioning. The target groups for ELINDER are students at the end of their education cycle, young professionals at the start of their career and experienced professionals and managers who change their career orientation towards nuclear decommissioning. "Metrology for Waste Characterisation and Clearance" is a Specific, topical training module for "specialisation in decommissioning". Experts from JRC explain the current practices and the developments in the following fields:
    - Radiological measurement principles
    - Destructive assessment techniques
    - Non-destructive assessment techniques
    - Measurement validation and statistics
    - New developments in waste characterisation
    - Waste and material clearance approaches
    - Metrology networks

  People are then able to elaborate and monitor characterisation plans in a nuclear installation or clearance/release plans of a nuclear infrastructure.

- CLP4NET, the Cyber Learning Platform for Network Education and Training, is sustained by the IAEA and allows users to easily find educational resources related to topics ranging from nuclear energy, nuclear safety and nuclear science and technology. It contains instructor-led courses and e-learning self-study resources and is provided to the interested public as a cost-free service.

  CLP4NET aims to facilitate sustainable education in the nuclear sector by empowering web-based development and dissemination of high-quality e-learning resources and learning environments, in a way that is cost-effective, scalable and easy to use. Its main components are:

  - Self-directed Learning Management System – designed to provide e-learning self-study materials for a wider audience
  - Instructor-led Learning Management System – developed to support and enhance instructor-led training courses for closed groups of students with online learning management features

- The NEA launched the Nuclear Education, Skills and Technology (NEST) Framework in partnership with its member countries to help address important gaps in nuclear skills capacity building, knowledge transfer and technical innovation in an international context. The NEST Framework is developed as an NEA joint undertaking gathering private and public organisations from interested countries (not-necessarily NEA member countries).

  It entered into force on the 15 February 2019. The goal of NEST is to:

  - energise advanced students to pursue careers in the nuclear field by proposing a multinational framework among interested countries to maintain and build skills capabilities;
  - establishing international links between universities, academia, research institutes and industry;
  - attracting scientists and technologists from other disciplines to examine nuclear technology issues and involving such actors in the resolution of real-world problems.

- Meet Cinch Project: In 2010–2016 a series of two “CINCH projects” – CINCH-I: Cooperation in Education in Nuclear Chemistry, and CINCH-II: Cooperation and training in Education in Nuclear Chemistry – was supported within Euratom FP7. The projects aimed at mitigating the special skill-based deficits within nuclear chemistry at masters and doctorate levels and the decline of number of staff qualified in this field. The projects were built around the well-proven five-phase (Analysis, Design, Development, Implementation, Evaluation) Systematic Approach for Training (SAT) developed by IAEA; while CINCH-I dealt with the first three phases of the process, CINCH-II concentrated on the Implementation. Additionally, evaluation mechanisms were proposed and tested on the pilot courses developed during the projects.

- In addition, there are several national centralised training organisations (Argonne NL (US) - Facility Decom, CICET-Russia, KSU – Sweden, WAKASA centre FIHRDC – Japan, …)

3.1 Organisation models (staff and resources)

In addition to planning the technical aspects of the decommissioning process, special attention should be paid to the organisational aspects that can affect safety and efficiency at least as much as the technical issues. When it comes to the hierarchical origination of the decommissioning team, while there are international guidelines that licenses can consider (see Figure 3.1-1), there are also differences from project to project and from country to country. The International Atomic Energy Agency (IAEA), which has conducted several workshops to collate decommissioning experience worldwide, states that there is no optimum organisational structure for decommissioning, "except the need to have a dedicated decommissioning team with sufficient resources" [190]. It seems to be clear that rather than trying to come up with international best (optimal) organisation models for decommissioning, application of more holistic approaches to planning the total decommissioning project and managing the change from operation to decommissioning anticipating potential problems and developing solutions ahead of time taking advantage of both available in-house as well as external experience, tools and technologies, can help to reduce the likelihood and impact of these human factors challenges, and to avoid unnecessary costs and delays. However, there is consensus on some general guidelines to be considered when planning the organisation of the decommissioning team to be. Such general good practices are listed below.

It is essential both to ensure that a suitable, motivated workforce for the decommissioning process itself is available, but also to maintain motivation, focus and safety at the operating plant until it is fully shut down. To reduce the negative effects associated with change, the management must communicate clearly and regularly about the plans for decommissioning. To reduce uncertainty about their future, staff must also be supported in competence and career development.

[190] IAEA-TECDOC-1394 (2004) Planning, managing and organising the decommissioning of nuclear facilities: lessons learned

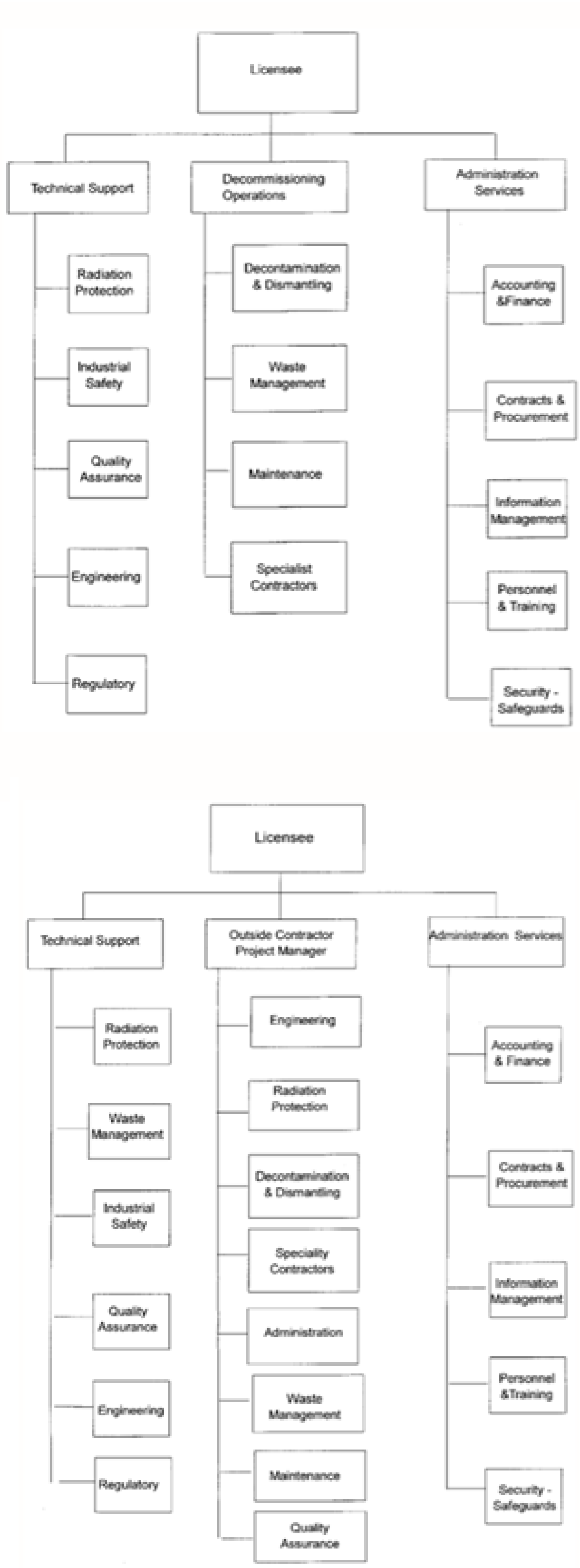


Figure 3.1-1 Licensee performing decommissioning organisation (upper) and licensee with outside contractor performing decommissioning (lower).

During the operation phase, the plant organisation is optimised for operation. But an operating organisation is not the most efficient way to run a decommissioning project, and the organisation must change to handle the demands of decommissioning. Some of the functions that the organisation relies on when operating the plant will still be applicable when going into the decommissioning phase; for example, the maintenance function can be used for dismantling the plant. However, some functions will be new, and some functions will no longer be needed.

Once the required functions are identified, it must be decided what is the most efficient way to fill those functions. Which tasks and responsibilities should be performed by contractors, and which should be performed by internal staff? And where is the most efficient way to conduct the tasks, e.g. in the field or from central support?

When an organisation needs to change, it is usually because the present way of reaching its business objectives is not working satisfactorily (anymore), or because the business objectives have changed. In both cases, deciding, planning and implementing the required change can be seen as developing new capabilities.

Any organisational capability can be said to consist of four interdependent capability elements: People, Processes, Technology, and Governance. When developing new organisational capabilities, it is not uncommon to experience that this was more difficult to achieve than foreseen. Or even, that evaluating whether the new capability hits the mark is rather hard to do. One reason is that the following traps are frequently fallen into:

- We mature some elements of the capability more than others, forgetting that the capability never improves beyond its weakest element.
- We discover interdependencies between capabilities or between capability elements too late.
- We fail to clarify the ambition of the capability early enough.

Planning for decommissioning should be performed by a dedicated group of people with appropriate knowledge of the plant. The size of the decommissioning planning group may vary; for example, the Oyster Creek plant in the US had a small group of 8 – 10 people[191], whereas the Chooz A plant in France had a larger group of 21 people[192]. Regardless of the number of people, the group should be made up of existing plant staff that have the vital expertise and knowledge that will be required for decommissioning planning[193].

The Maine Yankee plant in the US ensured that all disciplines were represented within the group responsible for initial decommissioning planning. It was considered important to ensure that all disciplines were involved, even for issues not directly relevant to a specific discipline or department. *"This is because in decommissioning it is not always obvious how a seemingly unrelated task/decision*

---

[191] Decommissioning Pre-Planning Manual, Technical report, 2000. available online epri.com/research/products/1003025

[192] Schmidt, G.; Ustohalova, V.; Minhans, A., 2013. Nuclear Decommissioning: Management of Costs and Risks, Study by Öko-Institut Darmstadt on behalf of the European Parliament's Committee on Budgetary Control, IP/D/CONT/IC/2013_054, Brussels

[193] Ljubenov V., Decommissioning Planning for the RA Research Reactor at the Vinča Institute. In Proceedings of Lessons Learned from the Decommissioning of Nuclear Facilities and the Safe Termination of Nuclear Activities, Athens, 11 – 15 December 2006

could affect other departments, and also because unique and better solutions/approaches to problems were offered by those not directly related to the issue"[194].

The planning team should be independent from the operating organisation and should be dedicated to the job of planning the decommissioning process. A separate organisation is required because decommissioning is a fundamentally different process from operation of the plant, with a very different end goal, and therefore the team should be allowed to focus fully on this process, without having to worry about conflicting goals from operation of the plant. The decommissioning team should also be separated from the operating team to ensure that decommissioning planning "*does not interfere with the plant organisation's priority of safe and reliable operation*"[195].

While there seem to be no standard recipes for the optimal organisation of the decommissioning team, including the optimal balance of in-house and contractor staff, it is important that the contracting strategy is adopted to the chosen level of contractor involvement in order to account for the difference in sharing the project risks between the contractor and licensee.

The following description is based on the IAEA report from the International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities[196].

Setting appropriate contract strategies is vital to the successful delivery of a programme of work such as decommissioning nuclear reactors. A good contract strategy will improve supply chain management whilst ensuring delivery for the Owner / Client at maximum value and minimal cost. In addition, a good contract strategy will support the delivery of best practice for the client and, if applicable, enable innovation from the subject matter experts engaged under the contract. There are several types of contracts used currently in decommissioning that are described in Section 2.5 Supply chain management for Decommissioning.

3.1.1 Experiences/Case studies

3.1.1.1 OECD NEA No. 7374 (2018): CEA

The aim of this report is to inform regulatory bodies, policy makers and planners about the relevant aspects and activities that should begin during the last years of operation and following the end of operation. Compiling lessons learnt from experiences and good practices in NEA member countries, the report supports the further optimisation of transition strategies, activities and measures that will ensure adequate preparation for decommissioning and dismantling.

3.1.1.2 IAEA TECDOC 1394 (2004): various decommissioning experience (international)

The objective of this document is to encourage the development and improvement of decommissioning planning and management techniques. The focus is on organisational aspects, to

[194] Maine Yankee Decommissioning Experience Report, Detailed Experiences 1997 - 2004

[195] Electric Power Research Institute «Preparing for Decommissioning: The Oyster Creek Experience. Technical Report 1000093. EPRI, 2000 »

[196] International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities, Tsuruga, Fukui-ken, Japan 11-14 November 2019

reduce the duplication of effort by various parties through the transfer of experience and know-how and to provide useful information for those Member States planning or implementing decommissioning projects.

3.1.1.3 IAEA TRS 399 (2000): organisational adaptation per decommissioning stage

The report covers organisational aspects of decommissioning and describes factors relevant to the planning and management of a decommissioning project. It identifies the general issues to be addressed and provides an overview of organisational activities necessary to manage a decommissioning project in a safe, timely and cost-effective manner.

3.2 Methods and software tools for knowledge management (e.g. competence preservation)

Now it is widely recognised that decommissioning has to be regarded as integral part of the lifecycle of a nuclear installation and has to be considered in the plant management strategy from the very early (conceptual design) phase. This trend is also reflected in national regulations and international guidance. Risk and safety management has a strong focus in this research activity. Efficient risk management during the decommissioning phase requires a high-level knowledge management and organisational learning process during the pre-decommissioning phases of a nuclear installation. The foundation for the information needed for safe decommissioning is provided by the data collected during plant design and operation. Therefore, the data requirements for the decommissioning phase need to be taken into account from the design phase onwards. In practice, the same data are also required for safe and efficient operation, in particular during outages, maintenance and upgrades.

The rapid growth of nuclear industry-initiated efforts towards managing nuclear knowledge. Now, intensifying international D&D activities generate there is a strong need for efficient management of knowledge in nuclear decommissioning. Suboptimal decisions and incidents resulting from inadequately high-level management of relevant knowledge may have severe health consequences to field workers and serious long-term consequences to the public and the environment. However, management of plant data for decommissioning involves a number of challenges that need to be addressed in future strategies for plant information management for decommissioning:

- Compared to design and operation of nuclear installations, current open international experience in decommissioning, combined with the strong differences in national policies, is insufficient for giving rise to standard international decommissioning procedures.
- Due to the possibility of long-term adverse effects to the population from suboptimal production and handling of the resulting radiological waste, very long-term safety and environmental awareness is required while the time and cost constraints are also very important.
- Human and organisational factors play a major role in the management of knowledge due to changing organisation, staff and daily tasks.
- There is a very high probability of losing knowledge from the operational phase important for decommissioning planning and implementation, due to staff retirement due to the downsizing entailed by decommissioning projects.
- Knowledge important to decommissioning produced during the pre-decommissioning phases need to be retained far beyond the working life of a single human for application in the decommissioning phase. For instance, there is a high probability of losing knowledge associated of systems and components that are no longer needed.
- There is an increased probability of losing knowledge due to low recruitment into related programmes. In spite of an improving tendency, decommissioning is still not attractive for career starters due to the “distractive” nature of the related activities.

In addition, similar to the operational phase of a nuclear installation, but possibly in a more accentuated way, knowledge relevant for decommissioning is complex, and is strongly dispersed with parts originating and/or stored in a large variety of places. For instance, knowledge important for decommissioning is produced during the lifetime of a nuclear installation by a number of different organisations and companies. During hand- and turn-overs of the installation much of knowledge important for safe and efficient decommissioning process does not get transferred or is transferred in a suboptimal way mainly due to the lack of standardisation and intellectual property issues.

Information available for informing the decommissioning process is also distributed (accessible) within a very large number of different systems, making it difficult for decision makers to combine all this information into a comprehensive understanding of the conditions and constrains for the work to be planned. This results in a serious amount of rework and reinvention, as well as decisions based on inadequate or incorrect information.

Information available for decommissioning planning is also usually vast and visualised in a non-user-friendly way, making it difficult to decision makers to understand the overall conditions described by a combination of all this information. In addition, information important for control of the process and decisions for continuation is dynamically varying, as the target environment, to which the information is connected to, is continuously changing as the result of the on-going D&D activities. Hence, informed every day decisions need to be supported by up-to-date, non-conflicting information describing the current conditions and constraints.

Furthermore, in addition to the mostly explicit knowledge composed of data, scientific information, and methods (technological and organisational know how), knowledge relevant for decommissioning has a very significant implicit part as well, residing in the experience and skills of humans. Experience is very hard to capture, is often neglected as part of the knowledge needed to be preserved for decommissioning and is very easily lost during changes in staff.

Rapid advancement in 3D computer-aided design (CAD), building information management (BIM) (including its adoption by the nuclear industry under the name KPIM - knowledge centric plant information management), as well as 3D digital simulation technologies have opened new dimensions for managing knowledge about nuclear installations throughout their whole lifecycle. Information conserved in Virtual Power Plants (see Figure 3.2-1) from the early design phase are expected to radically improve efficiency and safety of the decommissioning by supporting informed decision making, reliable cost estimation, and risk prediction during the planning phase, and significantly lowering the effects of unexpected conditions and events during the decontamination and dismantling phase.

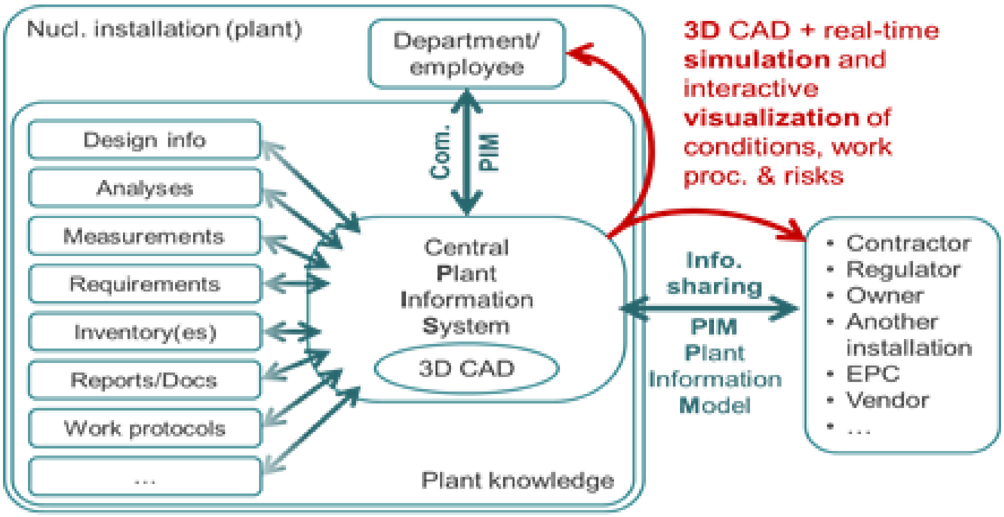


Figure 3.2-1 Integrated management of plant information supporting continuity between decommissioning stakeholders

For new builds, taking into account requirements for eventual decommissioning is an integral part of the design phase and commissioning. However, for nuclear facilities to be decommissioned in the near future, the new design knowledge preservation possibilities offered by modern information technology supporting transfer of knowledge relevant for decommissioning throughout the life cycle was not an option. During the operational phase, the design information is strongly supplemented with information, some of which has high potential for improving decisions during the decommissioning phase. Knowledge produced or captured during the design and operational phase that has importance for decommissioning originates from various sources and has very diverse nature. Some of this knowledge is explicit data resulting from measurements or modelling. A part of this data can greatly support decisions during decommissioning. Such information, however, is usually stored in multiple sources typically with little integration (linking) between the different information sources, and many times stored in not automatically searchable paper-based archives.

On the other hand, the design base of a nuclear installation encompasses a very extensive amount of information and not all that information is relevant or useful during the decommissioning phase. Due to the unavailability of relevant historical information in an easy to access form, combined with the advancement of new technology supporting reconstruction of the necessary facility information (e.g. 3D laser and radiological scanning), there is an increasingly prevailing trend for enabling modern BIM/KPIM based knowledge management methods through a combination of digitization of historical information and digital reconstruction of new missing or hard to access information. There is also an increasingly clear trend for application of such modern knowledge management techniques in a holistic way, as compared to the current prevailing trend for application of the technique for specific tasks the most common of which is training of personnel for safety critical jobs (see Figure 3.2-2).

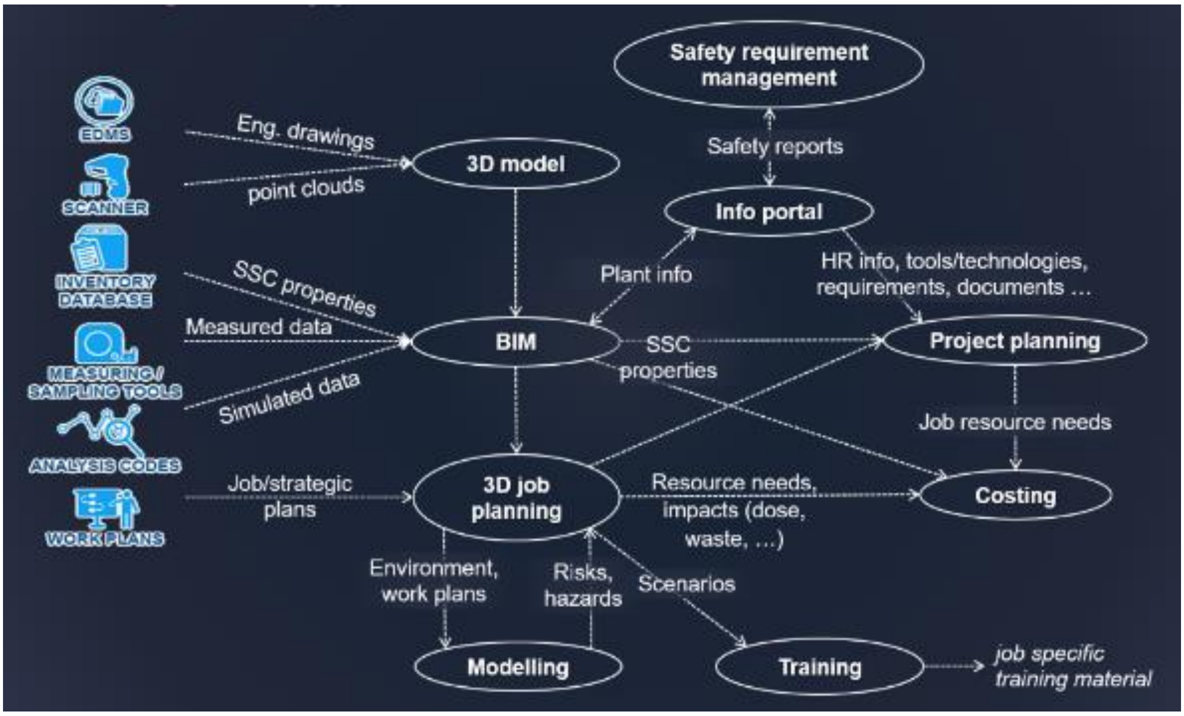


Figure 3.2-2 Integrated (holistic) knowledge management concept for nuclear decommissioning

3.2.1 Experiences/Case studies

3.2.1.1 Norwegian Experiences

In 2019, the Institute for Energy Technology (IFE) and the Norwegian Nuclear Decommissioning Authority (NND) established a cooperation focused on "Best Practice" within planning and practical issues related to decommissioning of nuclear facilities.

IFE will utilise the expertise within the international research projects (the Halden Reactor Project, PLEIADES, PREDIS, RoboDecom and LiveDecom) and experience from decommission planning activities in Russia and Ukraine to assess this type of information. The results from this work will be to identify good practice and to share this learning within the organisation to be built-up for decommissioning.

3.2.1.2 UK Experiences

In 2019, the National Nuclear Laboratory (NNL) completed a benchmarking exercise to explore the Knowledge Management (KM) activities and approaches of organisations from across the UK and internationally.

The purpose of the exercise was to identify good practice and to share learning across NNL, the organisations involved, and wider.

Twelve organisations were engaged, 13 including NNL, with headcounts ranging from 600 to 90,000. The organisations covered the nuclear, energy and environment sectors and included a number of national laboratories. The report[197] is an anonymised summary of the information collected.

[197] https://www.nnl.co.uk/wp-content/uploads/2020/04/NNL-Knowledge-Management-Benchmarking.pdf

3.3 General education for decommissioning

The decommissioning of nuclear facilities is an industrial activity that is growing worldwide, creating job opportunities but also requiring skilled workers. Most decommissioning programmes are implemented over several years via a sequence of projects and activities of different types. This explains, at least in part, the variety of the skills required.

In the transition phase from the closure of an installation to its decommissioning, part of the competences can be acquired by professional conversion of part of facility operating staff. Experience showed that while the early stages after shut-down may resemble the activities during a normal outage, soon the operator`s staff must adjust to accept that the facility will never operate again. Because of the complexity of work, the operator will have to develop plans for retention of essential workers, for retraining workers in new skills, hiring new workers and contractors and plans for oversight of contractors.

Over the past decades, European companies have been more and more involved in decommissioning projects that are targeted at delivering an environmentally friendly remediated end-product such as a fully restored green field site that can be released from regulatory control. European companies have developed strong know-how and today Europe can position itself at the top level in the world decommissioning market. However, in view of the expected expansion of the activities, efforts are necessary to maintain this leading position and, in particular, to ensure and share the underpinning knowledge, skills and competences[198] .

Education and training of young scientists and researchers is also a key element in the nuclear decommissioning sector. A survey[199] of the education and training opportunities in Europe shows that the evolution of nuclear decommissioning activities over the last decades has triggered the development of several programmes, particularly in the three main 'nuclear' EU countries: France, Germany and the UK. Education in decommissioning and waste management is currently provided as follows:

- PhD programmes and dedicated Professorships in decommissioning linked to engineering (an example is in Germany the 'Professorship on Decommissioning of Conventional and Nuclear Facilities' at the Karlsruhe Institute of Technology (KIT));
- two to three year or postgraduate taught masters courses focussed on decommissioning knowledge (examples are in the UK the one year 'MSc in nuclear decommissioning and waste management' at the University of Birmingham or in France the 'ITDD Master – ingénierie, traçabilité et développement durable' at the Université J. Fourier in Grenoble);
- dedicated modules in decommissioning integrated in a more general Master course in nuclear science or nuclear engineering (examples are in Belgium the 'Belgian Nuclear higher Education Network or BNEN', in the UK the 'Nuclear Technology Education Consortium or NTEC' modules at various Universities, or in France the 'Nuclear Sciences and Technologies engineering' degree sharing courses at CNAM/INSTN (50%) and apprenticeship in industry (50%));

198 PREDEC 2016 «Education and Training in Decommissioning, February 16-18, Lyon, France»
199 EU report EUR 27460 EN – 2015 « Education and training in nuclear decommissioning »

- Bachelor's degree with specialisation (about one year) in decommissioning (examples are in France the courses on decommissioning and waste management at the University of Caen and the University of Nîmes).

3.3.1 Experiences/Case studies

3.3.1.1 Education and training for decommissioning in France

The two main actors in training are INSTN (CEA's Education and training entity) and the Orano Training Institute, both having a large trainee base and covering a comprehensive curriculum with many items of high relevance for decommissioning.

The need for dedicated decommissioning-specific training on a national level is generally accepted, with an urgent need for engineers and technicians in the fields of safety, radioprotection and scenario building before decommissioning operations. French owners/operators support a national programme to assist the transition from operation to decommissioning, including the creation of a reference for decommissioning specific skills, as well as deployment of large-scale training programmes responding to skill gaps identified against this reference. The current national programme includes 12 training programmes in French, 3 in English and provides 6 professional (French) certificates. All the three French owners/operators (CEA, Orano, EDF) increasingly use advanced 3D simulation-based technologies in their training programmes (as well as other activities) related to decommissioning. CEA's Visiatom (since 2005) and Ifodem (since 2013) are information centres responding to the clear need for promotion, explanation and communication to the general public, students, and professionals about nuclear decommissioning. France is in strong collaboration with relevant international organisations (IAEA, ENEN, EC) and initiatives (e.g. ELINDER).

3.3.1.2 Education and training for decommissioning in Germany at Karlsruhe Institute of Technology (KIT)

Karlsruhe Institute of Technology (KIT) established a Competence Center for Decommissioning in order to secure the existing specialised know-how related to decommissioning and to extend this knowledge in an application-specific way. The Center relies on the already available comprehensive expertise and a highly performing infrastructure. Innovative dismantling technologies, radiation protection, radiological characterisation and decontamination of contaminated plant components, and interdisciplinary technology assessment are among the central activities (http://www.kit.edu).

3.3.1.3 Education and training for decommissioning in Nordic Countries

The DigiDecom© annual workshops (which started in 2017) organised by IFE and supported by the Norwegian Nuclear Decommissioning (NND) agency bring together a multidisciplinary group representing the professional community working on implementation and oversight of decommissioning for discussing opportunities and lessons learned from innovative methods for knowledge management, training and education in nuclear decommissioning.

In 2018 a workshop was organised within the OECD Halden Reactor Project (IFE) and the Nordic Nuclear Safety Research Forum. There special focus was on bringing stakeholder organisations closer together through digitally enhanced innovative concepts. This workshop also hosted the first meeting of a nuclear decommissioning advisory group to be launched by the OECD-HRP programme. Examples for specific subjects addressed by the group:

- Collaborative development of guidance for practical application of regulation.
- Application of advanced information systems for demonstrating regulatory compliance.
- Joint development of case studies with digital support concepts.
- Establishing and testing digital experience-based training methods.
- Joint development of e-Learning material for nuclear decommissioning.
- Interfacing big contractors with the regulators through digital safety demonstration methods.
- Collaborative testing of new decommissioning technologies using digital twins.

3.3.1.4 Education and training for decommissioning in UK

The Centre for Innovative Nuclear Decommissioning (CINDe) was established in 2017, led by the UK National Nuclear Laboratory (NNL) working in collaboration with Sellafield Ltd (SL), the University of Manchester, Lancaster University, the University of Liverpool and the University of Cumbria and based at NNL Workington, Cumbria.

CINDe is a PhD hub with the researchers performing their projects while being primarily based in industry, and with reach back to academic expertise within the university sector.

The purpose of CINDe is to provide additional underpinning and innovation to the Research and Development (R&D) needs for the Sellafield and wider nuclear industry decommissioning mission by:

- Bringing together leading academics, NNL and SL to perform innovative R&D to support the national decommissioning mission
- Enabling access to NNL facilities to university partners
- Allowing more frequent interactions between the academic partners and industry to enable closer more effective relationships working relationships to be developed
- Enhancing technical reputation through publication of high quality, peer reviewed scientific journal publications
- Bringing new talent to the industry and building skills in the next generation nuclear R&D skills pipeline in West Cumbria

CINDe currently has 16 PhD researchers in place at Workington, with ongoing recruitment for a further cohort to start in October 2021. The team are a vibrant, mutually supporting community of researchers working on related PhD projects to support decommissioning operations in the nuclear industry, with a particular emphasis on the Sellafield challenges. Because of the diversity of the decommissioning challenges, the CINDe team is both multi-national and multidisciplinary bringing a rich variety of expertise and perspectives to address the decommissioning issues.

UK Research Council Funded PhD Programmes

- In addition, UK Research and Innovation has funded a number of nuclear related research programmes, working with industrial partners over the last 15 years. These include]DIAMOND (Decommissioning, Immobilisation and Management of Nuclear wastes for Disposal)
- DISTINCTIVE (Decommissioning, Immobilisation and Storage soluTIons for NuClear wasTe InVEntories)

- and the current TRANSCEND (Transformative Science and Engineering for Nuclear Decommissioning) programmes

These industrial -academic partnership programmes have all contributed to developing solutions to decommissioning challenges in parallel with training of researchers to PhD level.

The UK Nuclear Decommissioning Authority, also funds work for the development of PhD level researchers, through its bursary award scheme, which it competitively tendered each year.

3.3.1.5 Education and training for decommissioning in Italy

The Radwaste Management School (RMS) is the training centre of the Sogin Group and ensures high-level professional training, promoting managerial and technological innovation based on the experience and specialised know-how in the field of decommissioning and radioactive waste management.

Founded in 2008, it is also open to professionals coming from institutions and companies and represents a reference point for safety management in industrial processes.

The RMS is part of the strategic assets for achieving the mission of Sogin and Nucleco. Nuclear decommissioning and radioactive waste management are, in fact, complex activities which develop over a long period of time and require strong expertise and multidisciplinary skills, encompassing civil engineering, design of large components, radiation protection and innovative technologies.

The development of highly specialised know-how is part of the Sogin strategy to guarantee maximum safety and implement an integrated knowledge management, education and training system. This is done in the light of transferring skills to future operators and satisfying the increasing knowledge demand in the sector both at an international and national level.

3.3.1.6 Education and training for decommissioning supported by Norway

Norway's cooperation with Russia on nuclear safety started in the beginning of the 1990s and has since 1995 formally been organised under the Norwegian government's Action Plan for Nuclear Safety. The Action Plan is managed by the Norwegian Ministry of Foreign Affairs (MFA) which grants funding to relevant projects. The role of the Norwegian Radiation Protection Authority (NRPA) is to oversee the projects implementation and report to the MFA whether the intended goals of the projects are being reached. The Action Plan focused specifically on nuclear safety and covered projects implemented in the period 2005-2009 at the two nuclear power plants in north-west Russia, Kola and Leningrad Nuclear Power Plants.

The Norwegian Ministry of Foreign Affairs (NMFA) funded in the period 2006-2016 an assistance project at the Chernobyl Nuclear Power Plant (ChNPP) in Ukraine. The goal of the project was to assist in the decommissioning of the site through better planning and training by establishing the Chernobyl Decommissioning Visualisation Centre (CDVC) based on Virtual Reality (VR) technology. The CDVC can be a pedagogical means for the ChNPP personnel in training the dismantling procedures and supports ChNPP in preserving the decommissioning expertise and knowledge at the plant on a long-term perspective. For more details see section 2.6 Methods and tools for communication.

3.4 Methodologies and tools for task specific training

Decommissioning involves different kinds of activities, and thus requires different skills and training than operation[200]. In comparison to the characteristics of work during power operation, decommissioning work involves more unique, non-routine tasks; the radiological risks are changed, and the radiological conditions are less predictable; industrial risks are increased, and the working environment is more uncertain. The work is performed by a smaller stable resource pool, supported to a larger extent by contractor staff. The focus of work should be on project completion rather than production. Specifically, the training requirements for personnel undertaking decommissioning activities should focus on project management skills; individual tasks and their interrelationship; pre-job preparations and risk assessment; waste categorisation and minimisation; and preparedness for radiological and industrial risks.

A combination of different competences is needed in teams that conduct the decommissioning work: plant knowledge; technical skills and soft skills; leadership skills; safety-oriented skills and a sound safety attitude; as well as more general decommissioning skills and a decommissioning mindset. Training should take into consideration the different backgrounds and skills of contractors and permanent staff from the operator company. Permanent staff will need training focusing on changing from an operational to a decommissioning mindset and the changed risks during decommissioning. Contractor staff will need training focused on radiological and industrial risks as well as on the plant-specific conditions. The permanent staff and contractors should also learn to work together as an integrated team.

Leadership is an important factor for creating and supporting change. Good leaders can help align the staff with the goals and approaches associated with decommissioning. Leadership development is therefore an essential element of a decommissioning organisation. However, leadership may not only be a management task. It is up to each person in the organisation to take a leadership role when needed. Therefore, training should also include elements of leadership.

Hence, improved methods for assessing and providing training with a strong team focus are required for enhancing integration of teams with a combination of permanent staff and contractors. Training methods must also ensure building a decommission mindset on the team level and provide an arena for developing and practicing leadership skills.

On a more technical level, training methods currently applied in decommissioning need to be improved in general for providing a better situation awareness and emergency preparedness for decommissioning jobs. In decommissioning, field workers, with limited radiation protection skills, have to penetrate areas where there is an increased risk for exposure to radiological hazards from different pathways. Field workers will have to perform new types of jobs in unwanted radiological environments that are continuously changing. In addition, due to uncertainties in the radiological information about the environment, workers may be exposed to unexpected exposure situations. Current hands-on, and theoretical training methods focusing on preparedness for radiological work and emergencies are suboptimal. Theoretical training based on classical methods does not provide

[200] https://www.nrc.gov/waste/decommissioning/program-docs.html

adequate preparedness for critical work entailing elevated radiological risks. Hand-on training is expensive but efficient for preparedness of the workers for normal work with non-significant hazard levels. However, such methods do not allow field workers to experience and practice work in increased radiological hazard conditions in planned or unexpected stressful emergency situations. Training methods enabled by advanced simulation and visualisation techniques are increasingly available with the emergency of supporting hardware. However currently available underlying mathematical simulation techniques allowing real-time interactive visualisation of radiation exposure conditions need further research.

The next paragraphs will address opportunities in application of real-time radiological risk assessment modelling and 3D visualisation (e.g. Virtual Reality) aided advanced training systems for field workers. It is worth mentioning that, in addition to advancement of the technology required for such systems, human factors orientated research into identifying best practices for integrating such systems into organisational practice is also required due to the lack of extensive experience from application of such systems in the field.

Based on extracted data from the 58 studies over a publication period of 30 years Virtual Reality based training is used in the following domains and for the following training objectives:

(1) Industrial training dealing with training employees working in assembly lines, carryout maintenance and oversee safety critical operations.

(2) Firefighting training dealing with training firefighters for search and rescue operations

(3) Safety and emergency preparedness training dealing with training users for safe operations and emergency preparedness

(4) Healthcare training dealing with all sorts of training associated to the healthcare industry from surgery training to general clinical competence for doctors, nurses

(5) Space training dealing with training astronauts for carrying our missions in space

(6) Defense training dealing with training in defense sector including military, air force and naval officers training

There are large differences between decommissioning organisations in terms of the reason for decommissioning, the time between shutdown and decommissioning, the decom strategies used, the availability of intermediate and long-term waste storage, and the proportion of contractor or internal staff. This diversity across different organisations implies a need for training that is tailored for the specific competence needs of each organisation. A thorough mapping of competence needs for each decommissioning phase is needed in order to plan what kind of training to implement for each phase. Although the specifics of the training programs will vary from one decommissioning organisation to another, the types of training required for decommissioning will be similar. Planners will need to understand different decommissioning techniques and methods, as well as have good project management skills, and lessons learned from other projects can provide valuable insights. Decommissioning staff also needs training for the specific jobs to be performed, an understanding of specific decommissioning risks and hazards, and training for handling unforeseen events and emergencies. There is likely to be changes in organisation during decommissioning, with different combinations of contractor and internal staff. To avoid incidents due to misunderstanding, there is also a need to train collaboration skills. Correct and detailed documentation of plant history and radiological conditions is needed as a basis for both planning and training.

It is evident that VR has been applied in numerous training contexts ranging from military training, firefighting, medical to many industrial trainings. The studies done on aerospace training using VR clearly found out that VR is the most suitable way of training when training in real environments is not available. In addition, VR-based training was also found to be a good alternative training for working at hazardous environments such as firefighting. When it comes to nuclear industry, on-the-job training is a commonly used training method. Here, learners are placed with mentors, experienced workers, who model for and supervise the learner as they perform tasks. Simulators are also widely used in the industry to train workers where the actual environment is too dangerous to expose the workers. Work environments and skill requirements for decommissioning operation comprises of all the above. This makes VR an ideal way of training for decommissioning operations. VR Head Mounted Displays (HMDs) provide full-scale virtual mock-ups offering natural navigation and interaction. They also provide accurate sense of scale and depth. Combining this immersion with the visualisation of radiation and other possible hazards in decommissioning provides a very powerful tool for training. The next steps in VR-based training are to develop metrics that are sensitive enough to gauge the levels of skill improvement and provide active feedback to the trainees to enable self-regulated learning. Combined with accurate measurement of training outcomes and active feedback, VR HMD technology-based training could decentralise the decommissioning training and make training available anywhere, anytime.

Figure 3.4-1 3D real-time simulation aided VR training for decommissioning

Table 3.4-1 Summary of the methodologies and tools for task specific training

Methodology	What is working	What is missing	Assessment and Possibility for improvement
Experimental testing/ User Experience	VR proofs of concept/demonstrators for hardware & software lab testing with a single participant at the time	(1) Scenarios closer to real-life operations; (2) Preponderant multiplayer collaborative scenarios & tasks	Leveraging VR training benefits in the decommissioning organisation
Assessment of VR readiness of the decommissioning organisation	VR infrastructure & know-how in the research organisation	VR technology & knowledge transfers toward the decommissioning organisation	(1) Identifying points of contact supporting VR in the decommissioning organisation; (2) Set up change management for VR digital solutions
Simulated on-the-job-training	VR generic training environments in terms of content (i.e. training concepts, procedures, purposes), presentation (i.e. 3D, 360 pictures & videos; text, CAD, analytics, etc.), and interaction (i.e. joypads, VR controllers, etc.)	Relevant and detailed on-the-job VR training materials; these should be prioritized according to the decommissioning organisational needs	(1) Knowledge & requirements elicitation from decommissioning experts and taskforce; (2) Setting up VR on-the-job-training at the organisational level
Benchmarking of AR, MR and VR for best on-the-job-training fit	AR, MR & VR capabilities of research organisation	Understanding of appropriateness of best-fit technology related to specific decommissioning scenarios & tasks	(1) Staged approach in the decommissioning organisation from VR toward AR & MR; (2) Categorisation of workforce skills and training delivery through VR (i.e. off-site) first, then through AR & MR (i.e. on site)

3.4.1 Experiences/Case studies

3.4.1.1 Framatome, Exelon, EPRI: VR immersive training (Reuters Events (2020)[201]

A survey of recent efforts in the nuclear industry revealed that learning by doing and interactive & immersive cost-effective training in plant digital twins including radiation exposure are among the main benefits of VR training. Lessons learned from Framatome, Exelon and EPRI also mention a set of state-of-the-art features that may be integrated within virtual reality (VR) and also in augmented (AR)

[201] Reuters Events (2020) Virtual Reality Transforming training in the nuclear industry https://oberontech.com/featured-offers/vr-whitepaper

and mixed reality (MR) training environments (the umbrella of VR, AR and MR is called extended reality XR), such as:

- scenario simulation
- location awareness within the VR plant
- recording & playback of user behavior within the VR simulations
- unlimited use in training of the VR web-based online simulations
- rapid deployment at scale of VR simulations (i.e. web-based online training solutions)

Such features are foreseen to enhance training proficiency and training effectiveness, improve trainees' engagement, improve plant operations, enhance worker safety, and reduce costs.

3.4.1.2 IFE VR training (1999-present)

IFE has conducted studies on VR-based training since the late 1990s. The results indicate that VR can provide a useful medium for training spatial skills. The results from studies in the early 2000s also document that the effect of the type of technology used for 3D simulation-based training, although all forms (desktop PC, non-stereoscopic large screen, stereoscopic large screen, and HMD) were found to be generally effective. The results indicated that sense of presence and performance are linked. More immersive technology gives a more engaging experience, which may lead to deeper processing of the information and to better retention and transfer of the material[202] [203].Today, the cost proposition of using an HMD, in addition to the vastly improved quality of the HMD-based experience means that cost of adequate display technology is no longer the hurdle that it was in the past.

Without offering underpinning recent scientific evidence from a systematic study, IFE's assumption based on earlier work[12,13,204] and recent experience[205] (i.e. testing VR decommissioning training scenarios with radiation protection experts, and using these scenarios remotely online in the DigiDecom ELINDER 2020 training), is that, for situation awareness, agility (understand and react), and procedural training for safety sensitive work in nuclear decommissioning (typically work in the radiologically active zone), methods based on advanced digital technology (including 3D simulation-based) are superior to traditional training methods in terms of: a) *effectiveness* (including measurability of training effect), b) *recall decay, cost-efficiency*, c) *overall time required for developing and competing the training*, d) *portability* (reusability at other sites and projects), and e) *flexibility* (efforts required for re-design for other training objectives).

In addition, the level of technology acceptance of digitalised training methods is also increasing (internationally). Outside of the specific area such as situation awareness, agility and procedural

---

[202] Sebok, A., and Nystad, E. Training in Virtual Reality: A Comparison of Technology Types. (HWR-734). Halden, Norway: OECD Halden Reactor Project.
[203] Nystad, E., Drøivoldsmo, A. and Sebok, A. Use of radiation maps in a virtual training environment for NPP field operators (HWR-681). Halden, Norway: OECD Halden Reactor Project.
[204] Sebok, A., and Nystad, E. Training in Virtual Reality: Qualitative Results from a Comparison of Technology Types. (HWR-768). Halden, Norway: OECD Halden Reactor Project.
[205] Stephane, L., Renganayagalu, S.K. (2020). Novel insights from VR use cases and experiments for training in decommissioning (HWR-1305). Halden, Norway: OECD Halden Reactor Project.

training, or for non-safety sensitive work, or training including development of psychomotor skills, 3D simulation-based methods may be not be cost-efficient in comparison with traditional human resource development methods. It is, however, worth mentioning that the increasingly capable simulation-based digital training technologies are rapidly pushing down the cost of deployment and increasing the effectiveness of the solutions. It can be concluded that VR training must have an important role in developing human resources for the decommissioning activities to come if we intend to successfully mitigate the foreseen shortage of highly skilled professionals for decommissioning work.

4. Characterisation during decommissioning

International initiatives

IAEA Initiatives

➢ Source Book of the IAEA, EC and NEA References in Decommissioning[206]

This source book addresses all aspects of Decommissioning. In the field of characterisation, answering to the question "what procedures of radiological characterisation (of facilities, buildings, land) for decommissioning should be implemented, including situation after severe accident of nuclear installation? ", it mentions:

- The Workshop on Radiological Characterisation for Decommissioning organised by NEA in 2012 at Studsvik, Sweden, 17 -19 April 2012 where participants shared current practices, lessons learned and innovation in radiological characterisation for decommissioning of nuclear sites and facilities[207] .
- The status report on Radiological Characterisation for Decommissioning of Nuclear Installations

The 2014 version of this source book mentions that this document is updated yearly and available to the wider public through the NEA website. But no update seems to have been made since 2014.

➢ IAEA International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities[208]

The report of his workshop gives short state of the art and lessons learned on methodologies for Characterisation of structures and building before dismantling in Japan, UK, Germany and Italy.

➢ IAEA report "Decommissioning of Research Reactors: Evolution, State of the Art, Open Issues[209]

IAEA report "Decommissioning of Research Reactors: Evolution, State of the Art, Open Issues" lists some recommendations:

- General management issues. Establish a decommissioning project team well before shutdown. Utilising the experience of the old operating staff is beneficial, but proper management also includes retraining in new skills and attitudes. Stakeholder dialogue should be emphasized throughout the whole project.

---

[206] WPDD - Source Book of the IAEA, EC and NEA References in Decommissioning, NEA/RWM/WPDD (2014)

[207] Workshop organized by NEA, Studsvik Nuclear, SSM, SKB and SWAFO at Studsvik, in Nyköping, Sweden, 17 - 19 April 2012. Proceedings are available at http://www.oecd-nea.org/rwm/wpdd/rcd-workshop/index.html.

[208] IAEA International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities, Tsuruga, Japan, 2019

[209] IAEA International Workshop on Preparing for Implementation of Decommissioning of Nuclear Facilities, Tsuruga, Japan, 2019

- Technical decommissioning planning is an established process, but the planning should start years before shutdown to enable smooths transition from operation to decommissioning. Preparatory work before shutdown also speeds up the overall project.
- Management of spent nuclear fuel and other radioactive decommissioning waste requires that the waste end point and national regulations are known before starting the decommissioning project.
- When choosing the applied technologies, one should consider especially reliability, ease of maintenance and generation of secondary waste.
- Sharing experiences and learning from other projects is essential.
- Cost estimates are often underestimated. Using robust and state-of-the-art techniques reduce this risk. Benchmarking the cost estimates with experiences from other decommissioning projects improve the accuracy.

➢ IAEA report on in- situ Analytical Characterisation of Contaminated Sites [210]

This publication represents a comprehensive review of the in situ gamma ray spectrometry and field portable X ray fluorescence analysis techniques for the characterisation of contaminated sites. It includes papers on the use of these techniques, which provide useful background information for conducting similar studies, in the following Member States: Argentina, Australia, Brazil, Czech Republic, Egypt, France, Greece, Hungary, Italy, Lithuania, Montenegro, Spain, United States of America and Uruguay.

➢ Report on use of Scaling Factors (SF)[211]

This publication assists Member States with the formation and implementation of efficient strategies for safe and cost effective classification and disposal of nuclear wastes. The new SF methodology exploits known quantifiable ratios of DTM to easy-to-measure (ETM) nuclides so as to facilitate radioactive waste processing. This publication contains guidance and case studies from Member States where the technique has been successfully deployed.

➢ IAEA Project “DACCORD”[212]

Link between waste inventories and global cost estimation of decommissioning

➢ IAEA Project on Graphite Characterisation

[210] In Situ Analytical Characterization of Contaminated Sites Using Nuclear Spectrometry Techniques. Review of Methodologies and Measurements. AQ-49. IAEA, 2017.
[211] « Determination and Use of Scaling Factors for Waste Characterization in Nuclear Power Plants », IAEA Nuclear Energy Series, 2009
[212] “Data Analysis and Collection for Costing of Research Reactor Decommissioning. Report of the DACCORD Collaborative Project”. IAEA TECDOC-1832. 2017.

In the last twenty years there have been organised and collaborative efforts to address the challenge of irradiated graphite waste management. These include four IAEA projects[213], [214], [215], [216]. These projects and the work of the partners contributing to these projects have been instrumental in understanding the characterisation requirements for irradiated graphite and establishing best practice in some areas.

NEA Initiatives

- NEA Report on R&D and innovation needs for decommissioning Nuclear facilities[217]

This report of 300 pages and more than 700 references addressed 2 aspects of Characterisation:

- Characterisation and survey prior to Decommissioning where it discussed challenges, current guidance, applicable current innovative technologies and Research & Development being conducted.
- Site characterisation and environmental monitoring

After update of the report[218], 4 challenges related to Characterisation were finally retained, out of a list of 7 challenges:

(1) PR decommissioning characterisation (Modelling concrete characterisation at depth and techniques for Hard to detect RN (alpha and pure beta) in solids with no dissolution)
(2) Use of remote sensing and satellite
(3) *Use of robotics*,
(4) Modeling mobile nuclides,
(5) Statistical modelling and sampling,
(6) P*rioritisation on waste management,*
(7) *Site remediation,*

- NEA TGPFD[219]

The Task Group on Preparing for Decommissioning during Operation and after Final Shutdown (TGPFD) involved regulators, nuclear operators and independent experts who reviewed between March 2015 and December 2017 strategic aspects to optimise preparations for decommissioning

[213] International Atomic Energy Agency, 2006. Characterisation, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. IAEA-TECDOC-1521
[214] International Atomic Energy Agency, 2010. Progress in Radioactive Graphite Waste Management. IAEA-TECDOC-1647
[215] International Atomic Energy Agency, 2016. Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal. IAEA-TECDOC-1790
[216] Wickham, A., Steinmetz, H.-J., O'Sullivan, P., Ojovan, M.I., 2017. Updating irradiated graphite disposal: project 'GRAPA' and the international decommissioning network. J. Environ. Radioact. 171, 34–40
[217] Report on R&D and innovation needs for decommissioning Nuclear facilities, NEA, 2014
[218] Needs and emergency technologies for decommissioning, Gerard Laurent, In Solutions, Norway, digidecom 2017
[219] NEA No. 7374, Preparing for Decommissioning During Operation and After Final Shutdown, OECD 2018 Working Party on Decommissioning and Dismantling. (WPDD).

from the last years of operation onwards. The report provided is based on case studies in Canada (Whiteshell), France (CEA Grenoble, EDF FBR and PWR), Sweden (Vattenfall Ringhals), Switzerland (Muehleberg), United Kingdom (Thorp), Spain (José Cabrera), USA (Connecticut Yankee, Vermont Yankee),

It gives general recommendations on:

- Definition of Data quality objectives
- Initial categorisation of the facility based on the evaluation of available historical information.
- Planning and carrying out characterisation during the final years of operation, insisting on putting in place a framework for management of the information
- Definition of nuclide vectors by material source and time

➢ NEA Strategic guidance on strategies for radiological characterisation[220]
  - Good practices for radiological characterisation for decision makers
  - This document underlines the role and significance of radiological characterisation in decommissioning and some key aspects of its implementation. It identifies the differences during the various phases of a nuclear installation. Even if it is more focused on building and equipment characterisation, concepts, recommendations and lessons learned apply to contaminated soil and groundwater.

➢ NEA: Radiological Characterisation from a Material and Waste End-State Perspective [221], [222], [223], [224]
  - Good practices and practical advices covering all stages of the characterisation process for implementers
  - Presentation of lessons learned in a regulatory perspective + general advices
  - Mention of a review of radiological characterisation practice across the UK nuclear industry

➢ NEA EGFWMD[225]

---

[220] Radiological Characterization for Decommissioning of Nuclear Installations", by Task Group on Radiological Characterization and Decommissioning (TGRCD) from 2011 at Working Party on Decommissioning and Dismantling (WPDD), NEA/RWM/WPDD (2013)

[221] "Radiological Characterization from a Material and Waste End-State Perspective: Evaluation of and international Questionnaire by the NEA Task Group on Radiological Characterization and Decommissioning" (NEA, 2016)

[222] NEA No. 7373: Radiological Characterization from a Waste and Materials End-State Perspective: Practices and Experience, by Task Group on Radiological Characterization and Decommissioning (TGRCD) from 2014 at Working Party on Decommissioning and Dismantling (WPDD), © OECD 2017

[223] Best practice on facility characterization from a material and waste end-state perspective, Matthew EMPTAGE, WPDD Task group on RCD, NEA, International Symposium on preparation for Decommissioning (PREDEC), Lyon 2016,http://www.oecd-nea.org/rwm/wpdd/predec2016

[224] Characterization: Challenges and opportunities - A UK perspective, Matthew EMPTAGE, EA, UK, Lyon, PREDEC 2016

[225] "Management of Radioactive Waste after a Nuclear Power Plant Accident", *NEA No. 7305, 2016. https://www.oecd-nea.org/rwm/pubs/2016/7305-mgmt-rwm-npp-2016.pdf*

The NEA Expert Group on Fukushima Waste Management and Decommissioning R&D (EGFWMD) was established in 2014 to offer advice to the authorities in Japan on the management of large quantities of on-site waste with complex properties and to share experiences with the international community and NEA member countries on ongoing work at the Fukushima Daiichi site. This report provides technical opinions and ideas from experts on post-accident waste management and R&D, as well as information on decommissioning challenges. Lessons learned from case studies (e.g. Three Mile Island and Chernobyl) including handling of any environmental contamination and the current status of related waste management are used to develop recommendations on physico-chemical characterisation including non-radiological hazards alongside radiological.

The group provided in 2016 a strategic approach to the Japanese government for effective management of radioactive waste related to Fukushima Daiichi

- NEA EGCUL[226]

This group addressed:

- Knowledge and experience for characterisation on a large amount of unknown waste
- case studies from France, Japan, Russian Federation, Ukraine, UK
- International feedback on Japan's developed characterisation methodology at 1F

The development of a reliable and efficient characterisation and categorisation methodology is a common challenge in the fields of post-accident radioactive waste management and decommissioning of complex sites. Following recommendations given in previous working group EGFWMD, the Japanese Nuclear Damage Compensation and Decommissioning Facilitation (NDF) requested the RWMC[227] to further assist in developing an integrated methodology for managing a large amount of radioactive waste with unknown properties, focusing on radiological characterisation.

The Expert Group on Characterisation Methodology of Unconventional and Legacy Waste (EGCUL) worked from 2018 to 2020 to share state-of-the-art knowledge and experience in characterisation with case studies from Japan, Russian Federation, Ukraine, United Kingdom and France.

The key issues and challenges discussed were classified as the "enablers", and the "technical aspects" as reported in the following Table 4-1.

Table 4-1 Key issues and challenges identified by EGUL

Enablers	Technical aspects
• Radioactive waste management framework. • Waste classification and categorisation. • Waste Acceptance Criteria for the treatment, storage and disposal of UL Waste.	• Characterisation project plan including sampling and analysis. • Development of Nuclide Vectors (fingerprints)

[226] EGCUL: RWMC Expert group on characterization Methodology of unconventional and legacy waste
[227] https://www.oecd-nea.org/rwm/rwmc/

• Defining End-states. • Defining priorities on complex sites and safety analysis.	• Characterisation reporting and review. • Retrieval, conditioning and packaging of waste.

EGCUL proposed a decision tree to be specifically designed in each situation and a phased approach integrating the wider management program and taking into account stakeholders' input as shown in following Figure 4-1 and Figure 4-2.

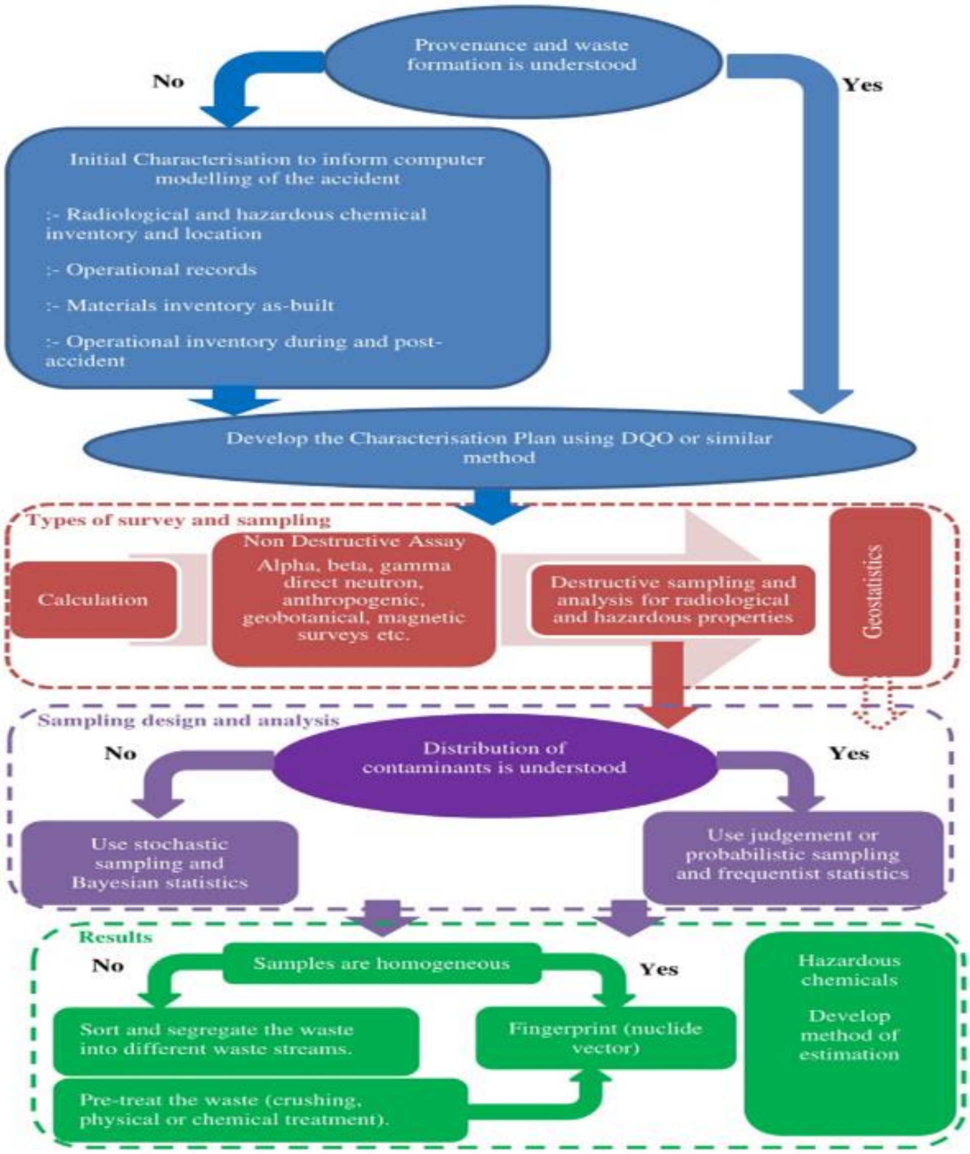


Figure 4-1 Decision tree proposed by EGCUL

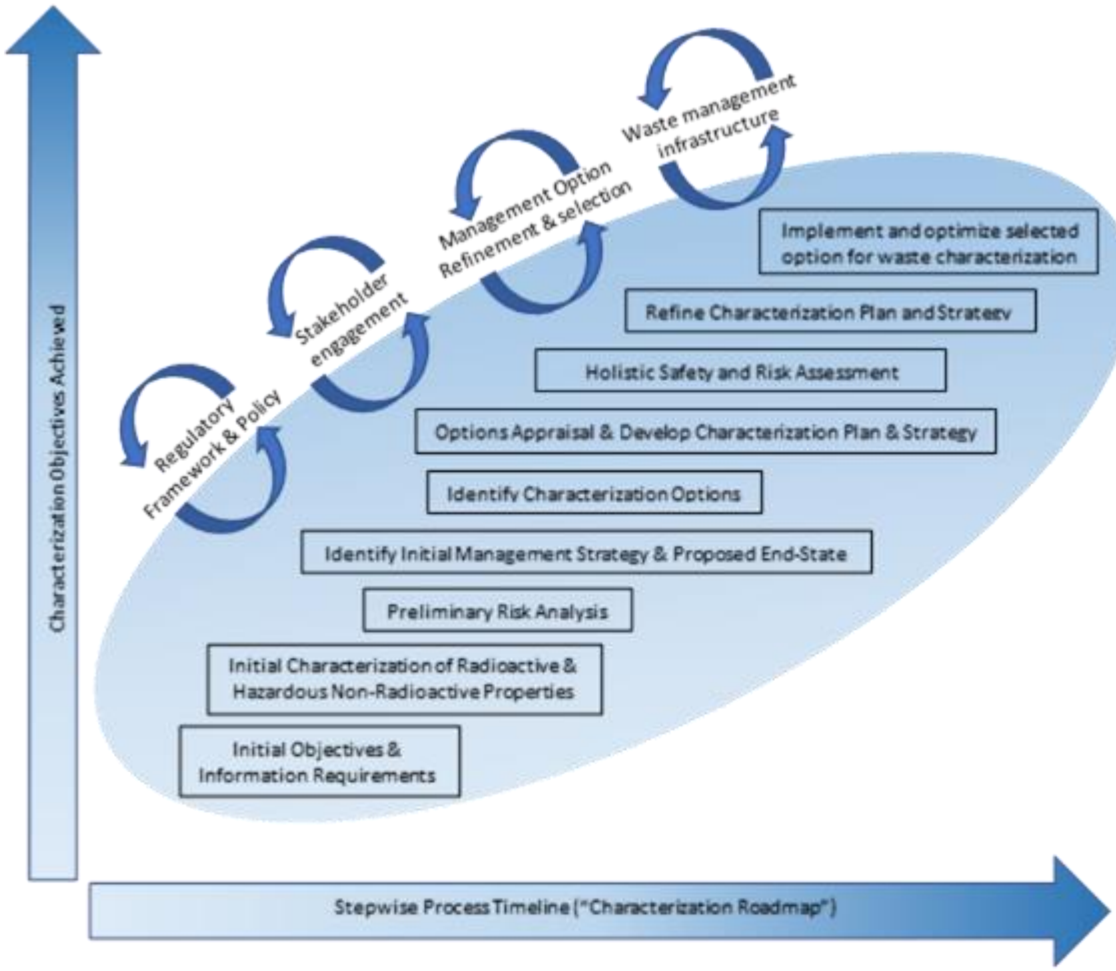


Figure 4-2 Phased approach proposed by EGCUL

The conclusion of NEA Working group EGCUL was that the radioactive characterisation methodologies designed for conventional waste (operational or decommissioning waste produced as part of planned operations or decommissioning) can be applied to Unconventional or legacy waste with some adaptations and enhancements. It is important that the radioactive characterisation method is supported by a flexible legislative and regulatory framework when there are significant uncertainties in the composition of the waste you are dealing with. These uncertainties need to be communicated in a way that all stakeholders understand so that there is no ambiguity in the overall strategy and that a "life cycle", "holistic" approach to characterisation can be developed. This will ensure that the available resources are optimised and enable a comprehensive R&D programme to be developed to support radiological characterisation and ultimately decision-making.

The Expert Group has identified the following areas for further development where guidance is missing:

- Setting a clear strategy underpinned by a regulatory framework to implement the policy.
- Having an integrated waste management strategy to optimise resources and time.
- Having a clear and flexible waste classification system and WAC such that UL waste could be included and that addresses radiological and other hazardous content of the waste in a proportionate manner.

- Having a flexible and adaptable sampling methodology.
- Understanding the end state goal prior to undertaking characterisation activities.
- Early and open dialog with all stakeholders.
- Understanding that characterisation is a continuous process.
- Having a clear review and validation process.

European Commission Initiatives

➢ EU-H2020 INSIDER[228]

INSIDER aims at developing and validating an improved integrated characterisation methodology and strategy during nuclear decommissioning of nuclear power plants, post accidental land remediation or nuclear facilities under constrained environments. It is based on different new statistical processing and modelling, coupled with present and adapted analytical and measurement methods, with validation through 3 case studies:

- Decommissioning of a back/end fuel cycle and/or research facility (Liquid waste storage tanks at JRC-ISPRA)
- Decommissioning of a nuclear reactor (Biological shields at SCK-CEN Belgian Reactor BR3)
- Post accidental land remediation (Contaminated soils at CEA)

The overall project methodology is based on common case studies in the form of inter-laboratory comparisons on matrix representative reference samples and benchmarking. Industrial partners (selected D&D actors) in close cooperation with major EU R&D organisations will drive comprehensive and realistic conclusions formalised in guidelines, recommendations and elements for pre-standardisation initiatives.

Deliverables give an overview of ongoing decommissioning projects within EUG member states, Japan and Ukraine, with their applicable regulations related to Decommissioning, their requirements and their practices regarding characterisation process and already developed guidelines for characterisation of soils and polluted infrastructures

Work Package 3[229] is drafting a strategy for data analysis and sampling design for initial nuclear site characterisation in constraint environments before decommissioning, based on a statistical approach:

- Development of a strategy for data analysis and sampling design, referring to state-of-the-art techniques, and provide guidance to the end user through an application in which the strategy contents can be explored in a user-friendly way.
- Implementation of the strategy when working out the methodology for the different test cases, in order to test its adequacy and identify potential flaws.
- Guidance summarising all the findings in a comprehensive data analysis and sampling design strategy

[228] EU H2020, Improved Nuclear SIte characterization for waste minimisation in DD operations under constrained EnviRonment (INSIDER), https://cordis.europa.eu/project/id/847641 , project website at http://insider-h2020.eu/
[229] https://insider-h2020.sckcen.be

WP5 is analysing the existing systems and methodologies for carrying out in situ measurements in constrained environments, aiming to classify and categorise these environments[230].

Within H2020 INSIDER project, the main objective of work package 3 (WP3) is to draft a sampling guide for initial nuclear site characterisation in constrained environments, before decommissioning, based on a statistical approach. The second task of WP3 aims at developing a strategy for sampling in the field of initial nuclear site characterisation in view of decommissioning, with the most important goal to guide the end user to appropriate statistical methods (including, but not limited to those identified during the first overview task) to use for data analysis and sampling design. To aid the end user in applying this strategy, a user-friendly application for guiding the end user through the contents of the strategy and the initial characterisation process is also developed[231].

- EU-H2020 MICADO[232]

MICADO project is addressing characterisation of packaged waste for the in-field Waste Management (historical waste retrieval operations and waste from decommissioning). Thus in this chapter we are addressing the methodology part and complementary information about technology for waste characterisation is given in chapter 8.6.

The project starts with the statement that "The absence of a consistent and straightforward solution to characterise all types of materials, along with the lack of an integrated solution for digitizing the enormous amount of data produced, is a critical issue. Now the systems rely on the operator's ability to maintain high operational skills and quality assurance with precision measurements that unfortunately today very often are associating high uncertainties not allowing therefore a real optimisation of the waste. The use of several un-automatised instruments implies taking many notes and inserting them into specific ad-hoc format and on a database manually, without the possibility to combine data including previously available legacy data's if present."

A WP is dedicated to better address the end-users and stakeholders needs (scheduled for delivery by July 2020).

MICADO will deliver, by 2022, a turnkey solution called Radiological Characterisation & Monitoring System (RCMS) DigiWaste toolbox, aiming at:

- faster execution of radiological measurements,
- optimised characterisation of a nuclear waste package combining non-destructive methods and tools that are already used as reference

[230] INSIDER WP5 (in situ measurements): developed activities, main results and conclusions Margarita Herranz, Raquel Idoeta, Khalil Amgarou, Frédéric Aspe, Csilla Csöme, Sven Boden and Marielle Crozet, https://doi.org/10.1051/epjn/2019061

[231] Development of a user-friendly guideline for data analysis and sampling design strategy Yvon Desnoyers1 and Bart Rogiers, EPJ Nuclear Sci. Technol., 6 (2020) 16 , https://doi.org/10.1051/epjn/2020006

[232] EU H2020, Measurement and Instrumentation for Cleaning And Decommissioning Operations (MICADO), project website at https://www.micado-project.eu/

- accurate tracking and long-term monitoring of nuclear waste
- efficient digitization of the full characterisation and logistical processes.

MICADO project is focusing on implementing a digitization process with integration of high TRL technologies (see Table 4-2).

Table 4-2 MICADO technology chart

Associated with document Ref. Ares(2019)3007608 - 08/05/2019

TECHNOLOGY CHART
Technologies integrated in the RCMS DigiWaste toolbox - MICADO Project

Technology	Type of Technology	Actual TRL	Recent Scientific Conferences Community & Peers	Acceptance / Recognitions / Awards	Commercially available	WP
Active Neutron Measurement	Quantification of nuclear materials like Pu in non-concrete packages (in complement or replacement to passive neutron measurement)	TRL 6	[22], [23]	Fully accepted technique (e.g. at ORANO La Hague)	No versatile system commercially available	5
Passive Neutron Measurement	Quantification of nuclear materials like Pu in non-concrete packages (in complement or replacement to gamma-ray spectroscopy)	TRL 8	[24], [25], ANIMMA 2015, 2017	Fully accepted	Technology already used industrially but no existing commercial system available for a large range of waste packages	5
ExpressIF	Expert system including AI algorithms and based on fuzzy logic	TRL6	Fuzz IEEE 2018, IPMU2018	Used in different applications	Versatile tool already used for several industrial applications	9
Waste database	Database developed for specific applications in the nuclear waste management field	TRL8	World Nuclear Exposition 2018	Used in different applications	Yes	9
Photofission	Quantification of fissile material in large concrete nuclear waste packages	TRL 6	ANIMMA 2013/2017, IEEE/NSS 2016/2018	Invited presentation in ANIMMA 2013	Technologies ready for industrialization	6
SiLiF	Neutron monitor during interim storage and/or transportation	TRL 6	[26] [27] [28] [29] [30] and IAEA Int. Conf. on Physical Protection of Nuclear Material and Nuclear Facilities 2017	verified in a certified neutron field at PTB Metrology Institute [28]	not yet	7
SciFi	Gamma ray dose rate monitor during interim storage and/or transportation	TRL 6	[31] [32] [33] [34] [35]	published in a book chapter [31]	not yet	7

Table 1: Technology Readiness Level and description of the technologies evaluated for the MICADO project with their most recent results.

➢ H2020 CHANCE[233]

[233] https://www.chance-h2020.eu/

Provides State of the art (current applications + ongoing developments) for the following techniques:

- High energy X-ray imaging (radiography and tomography)
- Gamma-ray spectroscopy
- Passive neutron measurement
- Active neutron interrogation
- Active Photon Interrogation
- Prompt Gamma Neutron Activation Analysis
- Fast Neutron Analysis with the Associated Particle Technique
- Beryllium characterisation by photon activation analysis
- Calorimetry
- Muon imaging
- Cavity Ring Down Spectroscopy
- Combination of measurement methods: Emission - Transmission Computed Tomography (ECT-TCT), Combined imaging-gamma-neutron systems, Coupling of non-destructive and destructive methods

R&D in CHANCE is focused on 3 innovative technologies to complete existing radioactive waste characterisation techniques: calorimetry, muon imaging and Cavity Ring Down Spectroscopy (CRDS) to complete existing radioactive waste characterisation techniques, once the waste is in a drum. For instance, in the frame of WP3, calorimetry is combined with gamma spectrometry and passive neutron coincidence counting to reduce uncertainties due to gamma and neutron attenuation in dense and rich-in-hydrogen waste matrixes like cement, bitumen, rubble, dry or wet soils

But it also addressed in situ measurement systems in deliverable (D2.3)[234], with transportable technologies already implemented or under development:

- transportable passive neutron measurement systems used at Cadarache for the characterisation of legacy waste
- Examples of active neutron interrogation systems pluggable to hot cells developed at CEA in order to assess uranium and plutonium in difficult to transport high level wastes.
    - transportable passive and active neutron system, which are modular to fit a wide range of radioactive waste packages with volume ranging from a few liters to almost 1 m^3.
    - TOMIS, high-energy x-ray tomographic system under development for in situ characterisation of legacy waste.
    - For some difficult to transport high level wastes: requirements before transportation for in situ neutron measurement and gamma or X-ray imaging Systems

➢ H2020 PREDIS (for more details see International Initiatives in Chapter 8)

WP4 of PREDIS project focus on Innovations in metallic material treatment and conditioning

[234] Deliverable (D2.3) of EU- H2020 CHANCE project (GA 755371): "R&D needs for conditioned waste characterization", 21/11/2019

One of the objectives of this Work Package is the developing of innovative and optimised characterisation techniques for metallic wastes.

- Other European initiatives where best practices can be found:
  - - H2020 EMPIR

- H2020 PLEIADES

Demonstrate a modular software ecosystem based on interconnection of front-line support tools by the partners through a decommissioning specific ontology building upon open BIM (see Chapter 2).

A module is related to the collection of inventories

- H2020 Project TRANSAT[235]

This multidisciplinary project will contribute to improving the knowledge on tritium management in fission and fusion facilities. It will aim to address the challenges related to tritium release mitigation strategies and waste management improvement, and refine knowledge in the fields of radiotoxicity, radiobiology, and dosimetry. Modelling tools for tritium inventory and tritium permeation fluxes estimation in fusion and fission devices will be compared and benchmarked to improve the level of confidence in their estimation. In addition, technological solutions for the development of on-request tritium production systems will be evaluated. WP2 is partly dedicated to non-destructive techniques to analyse tritium. Three techniques are currently developed: Autoradiography, LIBS (Laser Induced Breakdown Spectroscopy) and NRA (Nuclear Reaction Analysis). By 2021, all these techniques should be tested for tritium investigation, a radionuclide difficult to analyse in dismantling facilities.

- EU CARBOWASTE

The CARBOWASTE programme focused on the development of integrated guidelines, outlining the best-available and most environmentally acceptable technologies for the retrieval, treatment and disposal of irradiated graphite[236].

- EU CAST

The CAST project (CArbon-14 Source Term) aimed to develop understanding of the potential release mechanisms of carbon-14 from radioactive waste materials under conditions relevant to waste packaging and disposal in underground geological disposal facilities[237].

---

235 http://transat-h2020.eu/

236 Metcalfe, M.P., Banford, A.W., Eccles, H., Norris, S., 2013. EU Carbowaste project: Development of a Toolbox for Graphite Waste Management. J. Nucl. Mater. 436 (1–3), 158–166

237 Neeft, E.A.C., Carbon-14 Source Term CAST: Summary of the Progress achieved through CAST for the General Public and Decision Makers, D7.25 (2018), https://www.projectcast.eu/publications

- ELINDER

ELINDER presents a modular, coherent and commonly qualified training programme in nuclear decommissioning. The target groups for ELINDER are students at the end of their education cycle, young professionals at the start of their career and experienced professionals and managers who change their career orientation towards nuclear decommissioning.

"Metrology for Waste Characterisation and Clearance" is a Specific, topical training module for "specialisation in decommissioning". Experts from JRC explain the current practices and the developments in the following fields:

- Radiological measurement principles
- Destructive assessment techniques
- Non-destructive assessment techniques
- Measurement validation and statistics
- New developments in waste characterisation
- Waste and material clearance approaches
- Metrology networks

People are then able to elaborate and monitor a characterisation plan in a nuclear installation or a clearance/release plan of a nuclear infrastructure.

Other Initiatives

- ISO, within TC 85, several WG are of interest for SHARE:
  - SC2 "Radiological protection" (see Chapter 1)
  - SC5 "Nuclear Fuel Cycle":
    - WG 1: Analytical methodology in the nuclear fuel cycle, Deals mainly with lab analysis techniques and protocols, e.g. for determination of uranium and plutonium by various methods
    - WG 4: Transportation of radioactive material,
    - WG 5: Waste characterisation, Deals mainly with measurement and calculation techniques for characterisation of low- and intermediate-level radioactive wastes. E.g.: – ISO-16966:2013 "Theoretical activation calculation method to evaluate the radioactivity of activated waste generated at nuclear reactors" – ISO-21238:2007 "Scaling factor method to determine the radioactivity of low and intermediate-level radioactive waste packages generated at nuclear power plants". This resulted in a companion IAEA report: "Determination and Use of Scaling Factors for Waste Characterisation in Nuclear Power Plants", Nuclear Energy Series NW-T-1.18 – ISO/CD 19017 "Guide for gamma spectrometry measurement of radioactive waste"
    - WG 13: Decommissioning, a relatively new WG, where a standard is currently under development: – ISO/CD 18557 "Sampling and characterisation of sites, land, buildings and infrastructures contaminated by radionuclides or chemical products for remediation purposes"

List of already available standards:

- Guidance for gamma spectrometry measurement of radioactive waste[238]

[238] ISO 19017:2015

- Measurement of radioactivity — Gamma-ray emitting radionuclides — Generic test method using gamma-ray spectrometry[239]
- Measurement of radioactivity in the environment — Soil — Part 7: In situ measurement of gamma-emitting radionuclides[240]
- Characterisation principles for soils, buildings and infrastructures contaminated by radionuclides for remediation purposes[241]
- Measurement of radioactivity in the environment — Soil — Part 1: General guidelines and definitions[242]

ISO[243] advocates the integration of geostatistical methods for site characterisation.

This ISO standard articulates a set of principles for sampling strategy and characterisation of soils, buildings, and infrastructures during nuclear site decommissioning, taking into account constraints imposed by operations, budgets, and regulations while respecting As Low As Reasonably Achievable (ALARA) principles. The ISO is intended to standardize practices and aid users in the planning and reporting of characterisation activities.

- EPRI[244] reviewed the application of geostatistical methods in the nuclear power industry and in related industrial applications, along with available products for performing geostatistical analysis. Guidance is also provided for using geostatistics in support of nuclear site decommissioning and final status survey.

- ANIMMA CONFERENCES[245]

The objective of this analysis is to provide the nuclear scientific and industrial community with a state-of-the-art review of the whole field of nuclear measurements and instrumentation, mainly but not exclusively based on papers presented at the first four editions of the international conferences ANIMMA[246], i.e. from 2009 to 2015.

---

239 ISO 20042:2019

240 ISO 18589-7:2013

241 ISO 18557:2017

242 ISO 18589-1:2005

243 ISO, 2017. 18557. Characterization principles for soils, buildings and infrastructures contaminated by radionuclides, for remediation purposes

244 Guidance for Using Geostatistics in Developing a Site Final Status Survey Program for Plant Decommissioning. Product ID 3002007554. EPRI, 2016.

245 Michel Giot, Ludo Vermeeren, Abdallah Lyoussi, Christelle Reynard-Carette, Christian Lhuillier, Patrice Mégret, Frank Deconinck, Bruno Soares Gonçalves, Nuclear instrumentation and measurement: a review based on the ANIMMA conferences, EPJ Nuclear Sci. Technol. **3**, 33 (2017), https://www.epj-n.org/articles/epjn/full_html/2017/01/epjn170015/epjn170015.html

246 Advancements in Nuclear Instrumentation Measurement Methods and their Applications, http://www.animma.com, https://www.im2np.fr/fr/node/1113

What has been the progress made during this period of time, in terms of modelling, design, testing and signal interpretation of the various sensor types and measurement methods?

In which context were the new developments achieved, to satisfy which needs and address which challenges?

To answer these questions, the authors have chosen to develop the analysis according to seven major technological areas, some of them listed below being of interest in the field of decommissioning and associated waste management:

The first area, dealt with in Section 2, is that of neutron measurements. Fission chambers and Self-Powered Neutron Detectors (SPNDs) provide instantaneous data on in-core reactor neutron flux measurements. Progress on fission chambers means there is an ability to work within higher neutron and gamma fluxes, higher temperatures, and to select the most appropriate mode of operation (current, pulse or Campbell mode). It also means miniaturization and new developments on Fast Neutron Detection Systems (FNDSs). Thanks to improved simulation tools, there is a growing interest in SPNDs as a valuable and cheaper alternative to fission chambers for high level thermal neutron flux monitoring. They can be implemented as fixed in-core sensors for applications in which mobile in-core systems are not acceptable and in which ex-core sensors cannot ensure all required functions. Reactor activation dosimetry delivers time integrated data often useful for calibration purposes. Other topics of interest are semiconductor-based detectors or scintillator systems. They are partly driven by the need to replace He-3 based neutron detectors.

Section 3, deals with the second area: the photon detection and measurement, a wide topic with different kinds of applications for non-destructive assays and controls of materials and facilities, as well as medical and environmental applications. Two kinds of measurement techniques are considered here: passive photon measurements and active photon measurements whether they measure radiation from spontaneous decay of isotopes/materials or radiation induced by an external interrogating source.

In the case of passive measurements, the signals to be detected are obtained without external stimulation. Gamma spectrometry, X-ray spectrometry, photon emission tomography, self-induced fluorescence are the most frequent techniques. They make use of the radioactive decay and took the spontaneous emissions of particles from the object to be characterised. Challenges here are detection efficiency, energy resolution, qualification of uncertainties, miniaturisation for use on robotic platforms, testing on real systems as for instance burnup measurement of spent fuel assemblies, etc.

In comparison, active measurements are based on identifying the particle emissions induced using an external radiation source. The most widely used techniques are active neutron measurement, straight line photon transmission, X-ray gamma fluorescence, transmission tomography, and, to a lesser extent interrogation by induced photofissions, photon activation and photofission tomography.

Section Chapter 5 reports on a number of interesting examples of research carried out in the field of acoustics.

Optical fiber technology, the subject of session 6 is becoming a very useful technology to use in industrial instrumentation and in the nuclear industry in particular.

Cross-fertilisation is the topic of a last section of the chapter. Indeed, the use of coded apertures for imaging in fields such as decommissioning, safeguards and homeland security builds on experience in the field of medical imaging. Similarly, Compton camera design to detect alpha and beta emitting sources builds on developments in astronomy and medical imaging.

The seventh area reviewed in this analysis (Sect. 8) is that of data acquisition and electronic hardening.

The next section reports on the growing interest to use Field-Programmable Gate Array (FPGA) modules in Nuclear Power Plants (NPP) environments, explaining why they can be used to efficiently monitor and control such environments.

The last section of the chapter is devoted to advances in data communication networks.

The conclusion chapter of this paper (Section 8) tentatively draws some prospects for the future of nuclear measurements and instrumentation.

- PREDEC conference (2016, Lyon)

  Lessons learned from this conference are presented from a regulatory perspective. They highlighted the importance of pre-planning for decommissioning by the regulator and continuous dialogue between regulators and those undertaking the characterisation activities.

  It is also necessary to have a concept of radiological characterisation that includes facility history and the waste management aims.

  The overall characterisation of the plant should be completed at an early stage but results should be verified repeatedly throughout the decommissioning project considering the needs and objectives for the actual phases. Experience shows that characterisation competence is needed to the end of the project.

- Data Quality Objectives (DQO) [247]

  EPA has developed the DQO Process as the Agency's recommended planning process when environmental data are used to select between two alternatives or derive an estimate of contamination.

[247] EPA (2006), Guidance on Systematic Planning Using Data Quality Objectives Process, EPA QA/G-4, https://www.epa.gov/sites/production/files/documents/guidance_systematic_planning_dqo_process.pdf

The DQO Process is used to develop performance and acceptance criteria (or data quality objectives) that clarify study objectives, define the appropriate type of data, and specify tolerable levels of potential decision errors that will be used as the basis for establishing the quality and quantity of data needed to support decisions. This document provides a standard working tool for project managers and planners for determining the type, quantity, and quality of data needed to reach defensible decisions or make credible estimates.

The DQO planning process described in the following figure is used in the nuclear industry in three main ways:

1. During the early stages of decommissioning, to develop data acquisition plans for the initial characterisation of materials, equipment, buildings or land suspected or known to be contaminated.
2. During decommissioning or operations, to determine whether the concentrations of contaminants in waste materials fall above or below specific limits (e.g. as defined by the WAC for a waste disposal or treatment facility) so that wastes can appropriately managed.
3. Towards the end of a decommissioning, to develop data acquisition plans to determine whether decommissioning, remediation and/or decontamination of materials, equipment, buildings or land has achieved specified clean-up targets.

These applications are similar but, in the first case, the purpose is to determine a 'best estimate' of the concentrations of contaminants in the materials or wastes. In the other two cases the purpose is to test whether the concentrations of contaminants are above or below a specific limit, to support a waste management decision.

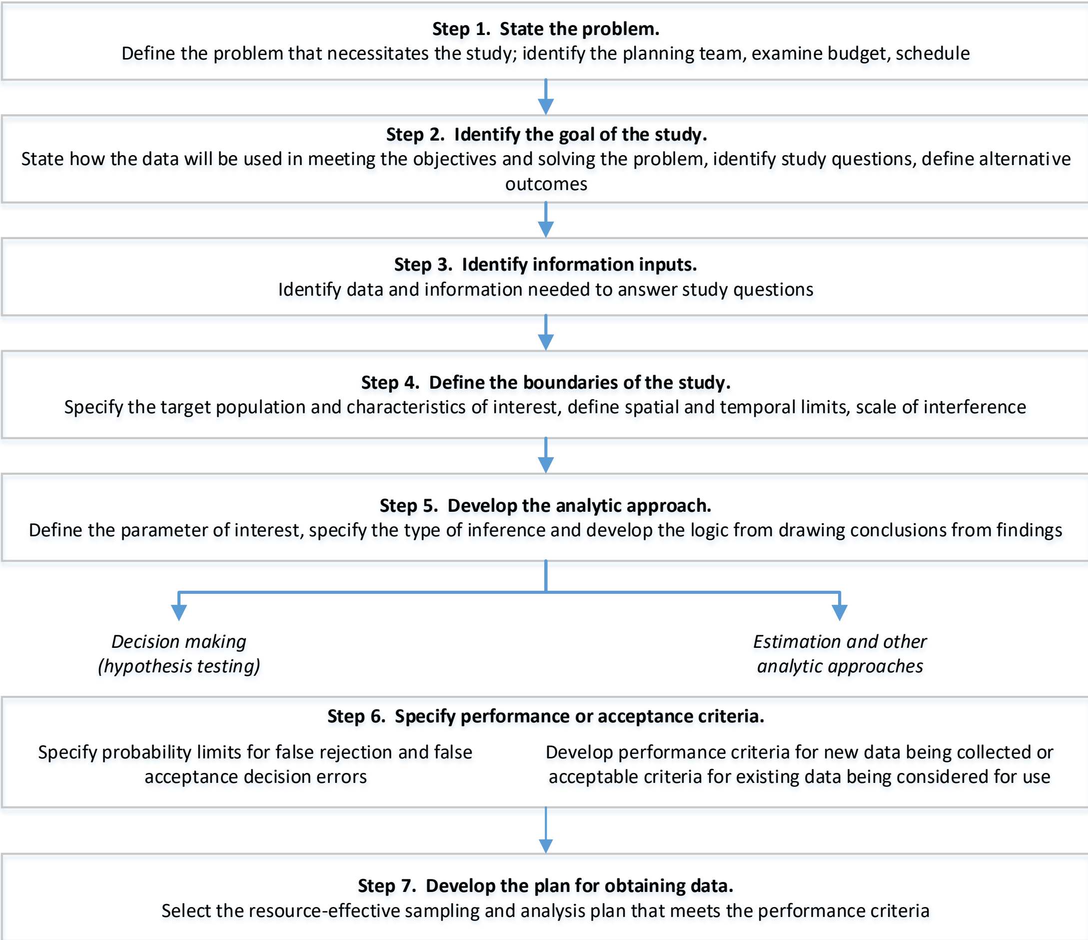


Figure 4-3 US EPA's Data Quality Objectives methodology

- Data Quality Assessment [248]

  US EPA has produced DQA guidance to determine if data obtained from environmental data operations are of the right type, quality, and quantity to support their intended use. Tools are also appropriate to the evaluation of characterisation data obtained from waste.

  The fundamental premise of DQA is that data quality, as a concept, is only meaningful when it relates to the intended use of the data. It is necessary to know what the data are to be used for before judging whether the dataset is adequate.

- "Multi-Agency Radiation Survey and Site Investigation Manual" (MARSSIM)

  Applying DQO and DQA, MARSSIM is focusing on the final status survey of surface soil and building surface, which is undertaken after the completion of decommissioning, remediation and/or decontamination.

[248] US EPA, 2000,

It is applicable to surveys of surface soil and building surfaces, providing information on planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose or risk-based regulations or standards whether the site meets the release criteria for radioactive contaminants.

- Multi-Agency Radiation Survey and Assessment of Materials and Equipment (MARSAME)

  Applying also DQO and DQA, MARSAME is dedicated to surveys contaminated material and equipment, demonstrating if they meet the clearance criteria.

  This involves metals, concrete, tools, equipment, piping, conduit, furniture and dispersible bulk materials such as rubbish, rubble, roofing materials and sludge, liquids, gases and solids stored in containers (e.g. drums of liquid, pressurized gas cylinders and containerised soil).

- Multi-Agency Radiological Laboratory Analytical Protocols (MARLAP)

  Applying also DQO and DQA, MARLAP provides guidance for the planning, implementation and assessment phases of projects that require laboratory analysis of radionuclides. It details the need for a consistent approach to producing radioanalytical laboratory data that meet a project's or programme's data requirements.

4.1 Methodology for historical site assessment

As seen already in Section 2.4.1, the first natural step in decommissioning projects is starting to gather historical information about the targeted environment for later use in the planning and implementation process. This includes identifying, collecting and securing all the technical data necessary for the deconstruction of the facilities: drawings, schemes, material certifications, operational history. Information on paper will be scanned to digital form. A data management system for this historical information will be created including definitions how to select, sort, structure and archive that information for decommissioning purposes.

This information can also come from a variety of sources including measurement, sampling, and modelling data. In addition, one potentially very important component of historical information relevant for decommissioning may exist in a tacit form within the existing or earlier crew of the installation.

Regardless of short or long-term decommissioning plans of nuclear sites, historical site assessments are performed to ensure that the institutional knowledge of the current workforce is captured for future reference.

Historical site assessment, documents a comprehensive investigation that identifies and evaluates historical information pertaining to events and conditions that have resulted in activation or contamination of structures and materials during the operational history of the site.

Contaminants of interest include both radiological and non-radiological materials that may have impacted systems, structures or components of the plant or environmental media within the site boundary.

Historical site assessment (HSA) involves:

- Plant operating history, radiological status inspections, license and technical specifications revision history and site modifications.
- A thorough review of records and reports from spills, incidents, effluent releases, operational surveys, radiological environmental monitoring and other documents related to radioactive material handling and past contamination.
- Interviews – particularly of long-term employees - to capture the historical use of buildings for activities that may have resulted in contamination that may not be documented otherwise.

Information collected during the HSA allows further development of an overall characterisation plan to collect measurements and samples from plant structures, systems and open land areas to cover the areas where contamination existed, remained or had the potential to exist.

As seen in Section 2.4, there is a high risk in losing some part of this knowledge due to downsizing in the transition and decommissioning phases. In addition, such kind of information is hard to capture using classical methods. In current practice, interviews with staff members are performed with relevant information captured in textual form. Capturing such historical information could be greatly enhanced by application of user-friendly interactive visualisation of the environment where connections of the information to systems, structures and components (SSC) in the environment or procedures performed in the environment are easily made. Applicability also extends to improvement of the registration of explicit information (data) relevant for decommissioning, since connection of such data (e.g. radiological contamination) to the environments is also very important.

4.1.1 Experiences/Case studies

4.1.1.1 Historical Site Assessments in the US

A number of commercial nuclear reactors in US have been fully decommissioned, where the facility has been deconstructed and the site returned to greenfield status. Based on MARSSIM manual, historical site assessments[249] [250] [251] [252] have been performed as one of the first steps in this decommissioning process.

4.1.1.2 Example in Korea

HSA implemented in USA served as basis for the more recent decommissioning of Kori Unit 1 in Korea[253]

4.1.1.3 Example in France – EDF /CEA

EDF is implementing in collaboration with CEA the Dismantling Information Model (DIM), in addition to more conventional project DOCADEC (documentation for Decommissioning) , to identify, collect, secure all the technical data necessary for the deconstruction of the facilities: drawings, schemes, certifications, historical, …) and to Define how to select, sort, structure and archive them for decommissioning projects.

The DIM of Fessenheim is under development. It is the first time in the context of a complete decommissioning of two nuclear power reactors, but some functions were already implemented on smaller operations.

It will also be using 3D modelling and virtual reality and has already conducted virtual visits of Fessenheim with helmets to improve their knowledge of the structure and avoid future contamination.

[249] U.S. NRC, YAEC, « Haddam Neck Plant Historical Site Assessment Supplement », ML012420073, 2001
[250] U.S. NRC, YAEC, « Yankee Nuclear Plant Site Historical Site Assessment », Rev.1, ML042510588, 2002
[251] https://www.nrc.gov/docs/ML0229/ML022970071.pdf
[252] U.S. NRC, ComED, « Zion station Historical Site Assessment (HSA) », Version 1, ML15342A281, 1999
[253] Development of HSA Procedure for Decommissioning Nuclear Power Plants, Ji-Hawn Yu, Wook Sohn, KHNP, 2017

4.2 Inventory assessment (Radiological and non-radiological)

Lessons learned from the first decommissioning operations in the 90s were that quantitative and qualitative estimates of equipment / structures to be decontaminated or waste to be removed from the site were often inaccurate, e.g. due to wrong knowledge of radiological, physico--chemical status of equipment or older waste drums, or to wrong inventory calculations only based on site history, etc.

Data were also missing when used in order to define best decommissioning activities before disposal (treatment of waste, conditioning, storage on sites, transportation) in compliance with one of the national waste classification schemes and associated disposal acceptance criteria. Also missing were accurate values of mobilisable source term for safety and environmental impact analysis during decommissioning and for possible prioritization of operations.

Plant or site inventory was thus recognized as key foundation to choose the best detailed decommissioning and waste management strategy, to minimise project risks and not to overestimate future storage or disposal. As a consequence, a lot of initiatives were launched at the international level to give guidance on methodologies in the field of characterisation:

- Site prioritisation - using screening analysis to define characterisation objectives.
- Characterisation objectives - using a “life cycle” and “holistic” approach.
- Historical information - what is likely to be available and how this can be used.
- Site reconnaissance and monitoring
- Sampling plan and strategy
- Geostatistics
- Developing radionuclide vectors (fingerprints)
- Associated modelling
- Decision-making
- importance of the overall waste management framework including waste acceptance criteria and categorisation/classification of waste

4.2.1 Description of methods

The physical and radiological inventory is an essential step in decommissioning:

- The more accurate the radiological inventory, the more accurate the technical-economic study of scenarios can be
- Visiting the areas to be dismantled, if accessible, is often very useful
- An inventory is never complete: we must make hypotheses and/or carry out investigations (samples, maps, etc.),
- The inventory must be optimised as the project progresses (iterative method)

Inventory assessment usually contains:

- A physical inventory: in what shape are the equipment, infrastructure and other to be dismantled, what logistics are available? etc.

- A complete mass balance: mass, volume, nature of materials
- A radiological inventory: cartography and evaluation of the total residual activity by zone, by mass of equipment, etc.
- A waste inventory

Inventory should be prepared carefully, with:

- List of zones with list of equipment concerned and accessibility
- Pictures and comments
- Visit to archives
- Preparation of data sheets
- Estimate time for collection on site and time to extract valid data

Physical inventory:

Physical inventory is the collection of information and data generally already existing:

- drawing, technical sheets
- reference safety documents
- minutes of activities, documents from operators
- pictures, videos
- memories from operators, interviews of retired people

Main constraints and difficulties rely on:

- missing or not updated drawings,
- lack of system for fast retrieval of documents
- bad traceability of modifications done in the facility
- difficulty of to access some zones due to high irradiation levels

Laser telemeters are now widely used to capture existing environments in 3D.

3D scanning technology offer high precision point clouds at 360 °.

Radiological inventory

Radiological inventory is needed to:

- determine “source term” in order to evaluate safety risks in case of accident and to prioritize operations if needed
- classify waste produced by category: VLLW, LLW, etc.
- draw complete inventory of waste and define best waste management strategy: sorting, treatment, routes, recycling, etc.)
- draw operational waste zoning
- establish isotopic reference spectra (footprints)
- allow evaluation of dosimetry for dismantling scenarios (ALARA)
- evaluate efficiency of decontamination techniques

Radiological inventory should be as accurate and as comprehensive as possible, with:

- irradiation dose rate of all intervention areas in the vicinity of the equipment to be dismantled,
- isotopic spectrum of contaminating radioelements for each area or equipment,
- surface, labile and fixed contamination of equipment,
- mass activity of each component, and waste classification,
- total radiological activity.

The establishment of the radiological inventory must first be based on the radiological history of the facility.

The irradiation dose rate of an area or equipment is measured from Geiger-Muller or other devices.

When there are several “hot spots”, it is necessary to identify them, by collimated and directional probes, systems combining image and dose rate measurement (gamma camera), sampling or smears,

The mass activity of an element also determines the dose rate.

For nuclear reactors and particle accelerators, activation calculations can also be implemented which needs powerful computing software, perfect knowledge of the operating history (neutron flux), perfect knowledge of the nature of materials and their geometry and complex physical calculations by specialised computing services.

The activation calculation is an important element in the evaluation of irradiation activity and dose rates, but must be validated by measurements and analyses on representative samples.

The isotopic spectrum consists of all the radioelements present in the facility or area concerned. It is used to better interpret measurements and characterise the radioactive waste produced. It is established in the form of a list of radioelements, with their respective percentages (activity or mass).

Surface contamination is divides into labile and fixed contamination. It is generally expressed in Bq/cm^2. It is used to:

- assess the risk of atmospheric dispersion during dismantling
- define the conditions for intervention by staff involved
- determine the most appropriate and effective decontamination procedures
- calculate mass activities and therefore classify waste by category.

It is measured by smear analysis, direct measurement on the support or after sampling (solid or liquid).

The radiological inventory often faces technical difficulties and intervention constraints:

- measurement of the dose rate of an area where background noise is significant, due to one (or more) "hot" source(s), does not allow the establishment of a representative radiation map,
- determination of the isotopic spectrum of contaminating radioelements for each equipment often requires sampling in the form of coring, smear or material cutting,

- Same problem for determination of the mass activity of each component; in the case of pipes or hollow-body equipment, the difficulty also lies in the knowledge of the internal activity, which not always accessible
- Measuring surface contamination of equipment often requires smears and measurements near or even in contact with the equipment, which is not always possible.

Waste inventory

Waste inventory will allow the decommissioning team:

- to know the quantities of waste for each category,
- to confirm or search waste routes and implement waste studies, with possible treatment (reduction of volume, decontamination, etc.),
- to assess the costs associated with the overall waste management, from retrieval sorting, storage, transportation up to disposal
- to determine the dimensions of the waste generated (scenario),
- provide for approved transport packages, etc.

Classification of radioactive waste is determined from the reference isotopic spectrum and mass activity.

For conventional waste (produced in non-nuclear zones), classification of waste is essentially by nature and according to routes (storage, recycling).

Initially, it’s mostly from auxiliary inactive buildings, offices, and some controlled areas.

At the end of the electro-mechanical dismantling, this also concerns infrastructures (walls, etc.), provided that they have been declassified beforehand. In this case it is mainly concrete rubble or metal structures.

The preliminary definition of waste zoning is therefore important, especially for country’s with regulatory no clearance level.

In addition to the "inventory" of the facility the control of the waste inventory is essentially based on periodic updates of waste flows forecast with search for the most appropriate route.

4.2.2 Experiences/Case studies

4.2.2.1 Examples of inventories at CEA

- Methodology for evaluation of source term in a research reactor, used for Phenix Fast breeder reactor [254], [255], [256]
- Methodology for evaluation of source term in a reprocessing plant with use of gamma camera and implementation in 3D mock up, used for APM Plant , Marcoule[257]
- Example of "Initial characterisation of tanks of radioactive effluents before dismantling" [258]
- CEA began standardisation of methods and techniques for collection and management of data in cartographies identifying singular points.

4.2.2.2 Example in Korea "Radiological Characterisation of a low and intermediate-level Radioactive Waste Samples from Research Reactor[259]

The radioactive solid wastes from research reactor and facilities consist of soft waste, disassembled equipment, laboratory supplies used during research activities, filters and ion exchange resins used in the purification of gas and liquid effluents, and other various type of combustible materials. According to the regulation for low and intermediate-level radioactive wastes in Korea, gross alpha, 3H, 14C, 55Fe, 59Ni, 63Ni, 90Sr, 94Nb, 99Tc, 129I, and gamma emitters (e.g., 58Co, 60Co, 94Nb, 137Cs,and 144Ce) should be quantitatively determined for the disposal treatment. In this study, analytical procedures were developed to quantitatively determine the radio-nuclides for the various type of radioactive solid waste samples. In the case of radioactive soft wastes, it is difficult to obtain the representative in the whole sample volume due to inhomogeneity for the radionuclide's contamination. Therefore, in order to assure the homogeneity of the sample, the whole samples were cut and mixed repeatedly. The process of sample preparation and measurement is composed of four main processes: direct measurement (gamma emitters), alkali digestion (129I), acid digestion (3H and 14C) and sequential separation and purification using extraction chromatography. The validated process were applied to radiological characterisation for the combustible waste samples from HANARO research reactor and facility of KAERI. The validation results used the standard spiked samples revealed that the methods could be applied for rapidly and satisfactorily recovering the specific target nuclides from samples with a high degree of accuracy and precision.

---

[254] "Radiological Characterization of Shut Down Nuclear Reactors for Decommissioning Purposes", 1998, Technical reports series, Number 389.

[255] J. VENARA *and al.,* "Radiological Characterization Methods Specifically Applied to the Preparation of the Dismantling of PHENIX Fast Reactor", ICEM, Brussels, sept. 8-12, 2013.

[256] "Radiological Characterization of Shut Down Nuclear Reactors for Decommissioning Purposes", 1998, Technical reports series, Number 389.

[257] P. GIRONES, L. BOISSET et C. DUCROS, "First report from an advanced radiological inventory for a spent fuel reprocessing plant", Avignon, SFEN, DEM2013

[258] "Initial characterization of tanks of radioactive effluents before dismantling", S. Tillard - CEA – FRA, Avignon, DEM 2018

[259] "Radiological Characterization of a low and intermediate-level Radioactive
Waste Samples from Research Reactor", Jong Myoung Lim1, KAERI, RadChem 2018, May 2018

4.3 Characterisation of activated components and areas

4.3.1 Metal

Metal waste from decommissioning represents one of the largest waste streams within the nuclear industry. In the UK, 16.9%[260] of all radioactive waste is classified as metallic. Metallic waste has a significant potential value if the waste can be decontaminated to a point where it can be released from regulatory control. Stainless steel for example, widely used in large quantities for a range of applications in the nuclear industry, has a residual scrap value outside the nuclear industry of around £140 per tonne[261] rising to over £3000 per tonne for copper. Along with the legal European wide driver of the Waste Hierarchy (introduced in EU Directive 2008/98/EC and also known as the Waste Framework Directive), this commercial value drives the requirement for detailed characterisation to determine if metals can be recovered for release from the nuclear industry and reuse rather than discarded as waste. The characterisation of metals is however, challenging. Being dense in nature, metals present shielding challenges related to characterisation. Sampling, of what can be large plant items and structures, is also difficult and these challenges need to be addressed if a comprehensive characterisation study is to be delivered.

Metallic radioactive waste can be split into two categories, contaminated metals and activated metals (Figure 4.3-1) although it is possible for a metal component to be both. Contaminated metals[262] are defined by activity being attached to the surface of the metal from an external source. It can be subdivided into fixed and loose contamination. Loose contamination can easily be wiped or brushed off using physical means whereas fixed contamination tends to be more difficult to remove, often having penetrated the surface structure of the metal, becoming chemically or physically bound, and therefore requiring more aggressive cleaning techniques to remove. Removal of fixed contamination may necessitate the removal of the metal surface to remove the fixed contamination completely but in general, contaminated metals can be decontaminated.

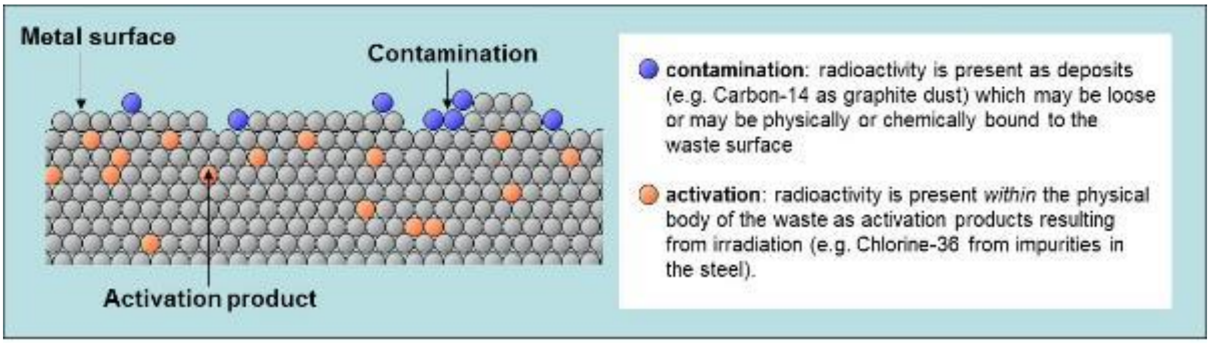


Figure 4.3-1 Illustration of Activation and Contamination associated with metals

[260] UK Radioactive Waste Inventory 2019 https://ukinventory.nda.gov.uk/the-2019-inventory/infographics/
[261] UK Scrap metal prices as of the 7th July 2020 https://www.scrapmetalpricer.co.uk/
[262] International Atomic Energy Agency, 1998 Radiological Characterization of Shut Down Nuclear Reactors for Decommissioning Purposes TECHNICAL REPORTS SERIES No. 389

Activated metals are metal components where the metal itself has become radioactive because of exposure to high energy neutrons. This results in contaminants or minor constituents within the metals becoming activated. A common example of this is the activation of non-radioactive 59Co, present in most carbon and stainless steels at concentrations ranging from 80 to 150, and 230 to 2600 ppm respectively, to 60Co[263]. 60Co decays by β emission and produces two major γ rays: 1.17 MeV and 1.33 MeV.

Activation of constituents within the metal structure results in the radioactivity being present within the structure of the metal itself, often throughout the whole thickness of the metal component. Whereas the activity associated with contamination of metals can potentially be removed, it is almost impossible to decontaminate activated metal by separating the activated constituents from the bulk metal.

Characterisation of metal is principally focused on establishing the nature of the radioactivity; contamination and/or activation, the principle radionuclides that contribute to the radioactivity and the levels of radioactivity present for waste categorisation purposes. Activated components cannot be decontaminated to lower their waste categorisation or to recover the metal and therefore, are generally only characterised to establish the activity categorisation and fingerprint for disposal or, if the half-life of the radioactive species permits, consideration for decay storage prior to either recovery or disposal as a lower category or waste.

There are two main methodologies which can be deployed for the characterisation of metals; Non-destructive and destructive techniques. Non-destructive techniques can be divided into two separate groups:

- Swab analysis, a technique where a paper swab of known area is wiped across an area and then measured for radioactivity and
- Remote direct measurement, for example. Gamma Spectroscopy and dose measurements.

A detailed summary of these techniques, complete with advantages and disadvantages of each, based on practical experience as well as technical performance, is provided in reference [264].

Swab analysis techniques, where an absorbent material (for example paper or cloth) is wiped across a contaminated surface, are used to determine if contamination is 'loose' of 'fixed'. Loose contamination will be wiped from the surface onto the swab so when the swab is measured, the activity will be detected. Conversely if contamination is fixed, no activity will be detected. In practice, both loose and fixed contamination are often found together. The swab sample can be used for infield analysis of metals by placing the swab under hand held probes and determining levels of alpha and

[263] International Atomic Energy Agency, 1998 Radiological Characterization of Shut Down Nuclear Reactors for Decommissioning Purposes TECHNICAL REPORTS SERIES No. 389

[264] Nuclear Decommissioning Agency, Solid Radioactive Waste Characterisation Good Practice Guide (wood) 2019. Document Ref No 207228-TR-01, Issue 1

beta contamination and the presence of gamma emitting radionuclides but if more detailed analysis is necessary, the swab sample can be sent for destructive analysis in a laboratory setting.

Swab sampling should be used with caution and only as an indicative method to determine if contamination is or isn't present. Variability in the pickup of contamination, the size of the area wiped with the swab and the pressure applied to the swab all contribute uncertainty regarding the activity collected. Therefore, the technique should not be used to try and qualitatively assess the levels of contamination on a surface.

Remote direct measurement techniques are numerous and can determine both the radiological and isotopic nature of the activity both on and within metal objects. Gamma based techniques tend to be the most commonly deployed and over recent years, as improvements in software and data processing systems have been introduced, the ability to interpret the gamma signals and generate both two- and three-dimensional images of activity has significantly improved. Again, a detailed summary of these techniques is provided in reference [265].

Other remote direct measurement techniques include, for example, x-ray analysis, neutron-based interrogation techniques and laser-based methods. Portable hand-held X-ray fluorescence detectors are widely used to identify elemental components so will detect surface contamination of a metal object elementally but do not provide activity or dose data. Neutron based systems, for example, passive neutron coincident counting (PNCC) are more usually used screen plutonium bearing wastes and are also highly complex and relatively expensive to deploy. In addition, given the attenuation of neutrons by metals, examples of the application of this technique tends to be less common in the area of decommissioning characterisation. Laser based methods, for example Laser Induced breakdown spectroscopy (LIBS) and Raman Spectroscopy can also be used and significant research and development in the use of these techniques, particularly at a distance techniques, has been undertaken and is documented in the literature. However, these techniques again measure elements and compounds and whilst are of use in metal characterisation, do not detect radioactivity.

Destructive analysis encompasses a wider range of laboratory-based analytical techniques however, the challenge when characterising metals is the collection of representative samples. Sampling metals can be challenging especially as often, metal items are large and complex shapes (boilers, pipework, tanks etc.) and may be difficult to access fully. In addition, activity in metal components is often on the internal surfaces (of pipe work, valves, tanks or similar) and therefore very difficult to directly access. Cutting of metal components will be required to undertake destructive analysis. The analysis can either be undertaken on the section of metal recovered, by direct laboratory analysis of the metal sample can be dissolved and an aliquot of the liquid analysed to characterise the contamination present. With all analysis, the purpose of the characterisation will drive the development of the analytical schedule.

[265] Nuclear Decommissioning Agency, Solid Radioactive Waste Characterisation Good Practice Guide (wood) 2019. Document Ref No 207228-TR-01, Issue 1

Table 4.3-1 Summary of the characterisation techniques

Characterisation requirement	Data gathering Technique	Advantages/disadvantages
Large area / large metal components.	Swabs	Cheap, rapid but only pick up loose contamination and prone to variability depending on technique used. – Qualitative only.
	Remote direct measurement	Well-developed techniques and can be used to map activity on large items. Will detect both loose and fixed contamination but not differentiate. Techniques may be challenged by large complex metal items dur to shielding issues etc.
	Physical sampling / analysis	Difficult to obtain representative samples for analysis. High quality analysis can be challenging but laboratory methodologies can used to differentiate between contamination and activation
Small components / internal surfaces	Swabs	Unable to deploy inside small components (small diameter pipes, sealed tanks etc).
	Remote direct measurement	Can detect gamma activity inside pipework etc but of alpha and beta activity will be shielded.
	Physical sampling / analysis	Likely to be the most effective method for characterisation but sampling may be challenging

4.3.2 Concrete

Large volumes of concrete are present in nuclear facilities across the world. In addition to forming structural components of buildings, concrete is also used in large volumes for its shielding properties, being relatively dense and compared to other shielding materials, for example lead and steel, cheap. Concrete is the main component in lower activity wastes. Figures published in the 2019 UK National waste inventory[266] indicate that low and very low level radioactive wastes account for 94.5% of the total radioactive waste volume. Concrete and rubble account for 31% of all low level and 88% of all very low level UK radioactive waste.

With the majority of concrete waste being only slightly contaminated or less, detailed characterisation has the potential to yield significant benefits by enabling disposal as lower category waste or even release from regulatory control if it can be demonstrated that the levels of activity with the concrete fall below (for example, in UK legislation) Out of Scope levels. In the UK, guidance issued in 2018 by the Environment Agency, Scottish Environment Protection Agency and Natural Resources Wales[267] provides a framework for the potential reuse of slightly contaminated concrete for site restoration

[266] Nuclear Decommissioning Agency. 2019 UK Radioactive Waste and Material Inventory. Radioactive waste inventory (https://ukinventory.nda.gov.uk/the-2019-inventory/2019-inventory-reports/)
[267] Scottish Environment Protection Agency. Management of radioactive waste from decommissioning of nuclear sites : Guidance on Requirements for release from radioactive substances regulations. Version 1.0. SEPA 2018

purposes. The legislation provides a risk-based framework to enable slightly contaminated concrete to be reused for void filling, road sub-base materials and reprofiling as examples, if it can be demonstrated this can be achieved with only a minimal residual risk to future site users. This approach aligns with the requirements of the Waste Hierarchy by enabling a potential waste material to be recovered and reused.

Concrete, like metal, can become radioactive by both contamination and activation. Concrete differs from metal in that concrete can be porous and contamination can penetrate through a significant thickness of concrete over time, especially if more mobile contaminants, for example tritium are present (Figure 4.3-2).

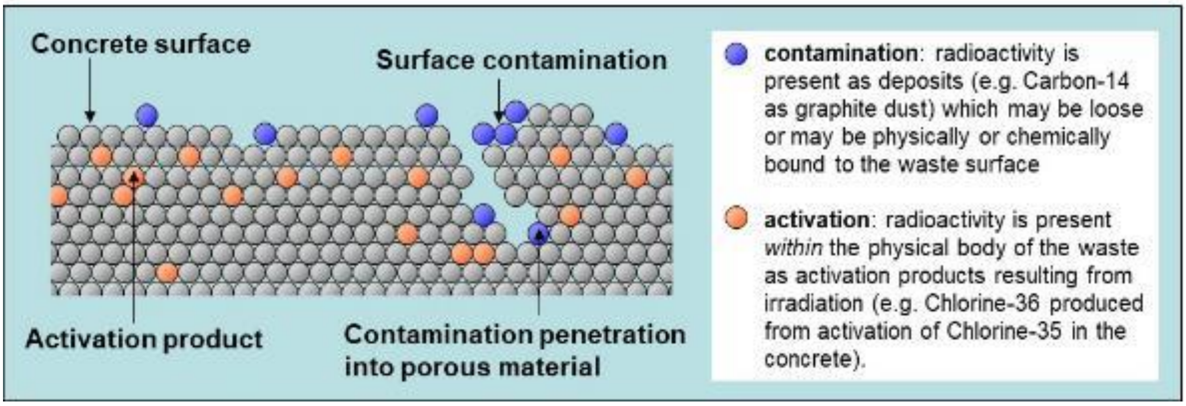


Figure 4.3-2 Illustration of Activation and Contamination associated with concrete

Characterisation of concrete can be undertaken can be undertaken in three ways;

- Whilst the concrete is insitu as part of the original structure.
- As a solid block or section cut from an existing structure using (for example) diamond wire cutting techniques.
- As crushed material or rubble, broken up by mechanical methods into a range of different sized fragments from powdered materials to large lumps.

The characterisation approach will be dependent on the decommissioning approach to be taken. Insitu characterisation of a concrete structure has the advantage of allowing the contaminated areas of the structure to be located and removed, potentially decontaminating the structure and allowing demolition to be undertaken using standard demolition techniques. Where the contamination/activation of the whole structural element has occurred, preventing decontamination, deconstruction by cutting the concrete into blocks allows the individual blocks to be assayed using both remote detection systems (gamma spectroscopy for example) and destructive sampling and analytical techniques which can then be modelled to provide a detailed assessment of the radioactivity present.

When concrete has been crushed into a range of sizes, several possible characterisation approaches can be undertaken. Dependent on the level of characterisation required, a number of small discreet samples of concrete can be taken for destructive analytical analysis. The number of samples can be determined statistically, depending on the degree of heterogeneity in the spread of contamination in the concrete or determined through a structured decision-making process, for example, the Data Quality Objective approach[268]. Alternatively, the crushed concrete can be placed into a container of a known geometry; 200l drum, a skip or 3m3 box would be examples, and assayed using remote detection systems and modelling, the same methodology used for a solid block.

A range of sampling and analytical techniques are available for the characterisation of radioactive concrete. They can be divided into remote detection and destructive sampling and analytical techniques but again, as with metals, the principle challenge is ensuring the samples taken are representative of the waste and the sampling approach chosen does not bias the results obtained. One of the challenges when characterisation concrete is undertaking depth profiling to demonstrate how deep contamination and/or activation has penetrated. The approach taken to sampling concrete is again driven by the characterisation requirements and the required output of the characterisation program needs to be defined if the correct sampling technique is to be used.

Sampling techniques for concrete include;

- Surface ablation of the concrete by mechanical scabbling. This can also be achieved by Laser ablation, heating/spalling etc.
- Diamond core drilling to remove intact cores of concrete from large structures or concrete sections.
- Vacuum drilling, where the concrete is drilled to produce a powder which is collected within a filter for analysis. Again, for large structures and concrete sections.
- Discrete sampling for laboratory analysis from rubble or crushed concrete resulting from demolition of structures, slabs etc.

Analytical techniques are similar to those used to analysis metal but also include depth profiling;

- Swab analysis of concrete surfaces for the presence of loose contamination.
- Large area scanning using remote detection techniques, gamma spectroscopy etc. This can be undertaken by deploying hand-held units but also vehicle or drone mounted units for very large areas.
- Core scanning for depth profiling using gamma spectroscopy.
- Gamma spectroscopy combined with bespoke modelling software for known geometries; discreet blocks, drums or skips of crushed concrete waste.
- Laboratory analysis of individual samples for a wide range of parameters, both radiological and non-radiological.

[268] EPA. Guidance on Systematic Planning using Data Quality Objective Process. EPA QA/G-4, 2006.

Table 4.3-2 Summary of the characterisation techniques for concrete

Characterisation requirement	Data gathering Technique	Advantages/disadvantages
Large area / large concrete structures and concrete cores.	Swabs	Cheap, rapid but only pick up loose contamination and prone to variability depending on technique used. – Qualitative only.
	Remote direct measurement	Well-developed techniques and can be used to map activity on large structures. Will detect both loose and fixed contamination but not differentiate. Modelling can be used to infer depth profile but this will not be by direct measurement. Alpha and beta activity may be shielded if within the structure (within voids, pores, construction joints etc.). Very useful technique if robust fingerprint for the material has been established.
	Physical sampling / analysis	Relatively straight forward to obtain representative samples for analysis. Core samples enable ex-situ dose profiling but cross contamination during coring is a possibility. High quality analysis can be achieved by laboratory methodologies and can differentiate between contamination and activation.
Rubble and crushed concrete	Swabs	Unsuitable on crushed material unless large fragments are present, but application is limited and data quality likely to be poor.
	Remote direct measurement	Advantage is rubble and crushed concrete can be placed in a known geometry (drum or skip etc) but alpha and beta activity will be shielded. Very useful technique if robust fingerprint for the material has been established.
	Physical sampling / analysis	Likely to be the most effective method for characterisation but care required to obtain representative sampling. Heterogeneity of the concrete is a significant challenge when sampling.

4.3.3 Graphite

Irradiated graphite waste around the world is estimated to be approximately 250,000 tonnes[269][270]. This material will arise from several of the early materials test reactors and plutonium production reactors as well as the commercial Magnox, UNGG, RBMK and AGR fleets. In the UK, irradiated graphite is currently estimated to constitute approximately 23% of the UK's Intermediate Level Waste (ILW) inventory[271]. It is also recognised that the issue of graphite waste management must be

[269] International Atomic Energy Agency, 2010. Progress in Radioactive Graphite Waste Management. IAEA-TECDOC-1647.
[270] International Atomic Energy Agency, 2006. Characterisation, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. IAEA-TECDOC-1521.
[271] International Atomic Energy Agency, 2016. Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal. IAEA-TECDOC-1790

addressed in the context of Generation-IV reactors, such as the (Very) High Temperature Reactor ((V)HTR) and Molten Salts Reactors (MSR)[272].

Characterisation requirements vary greatly depending on the plant, previous operating and/or storage conditions, decommissioning strategy and regulatory framework. For example, characterisation requirements for the decommissioning of core graphite from a power generation reactor that has remained in Safestore for a number of years will differ from the decommissioning of graphite components stored with mixed waste in ponds and silos for decades. Similarly, characterisation requirements for disposal of irradiated graphite in a Geological Disposal Facility (GDF) will differ from those for shallow or near-surface disposal. Although the IAEA issues guidelines on the categorisation of waste and the various disposal options, the regulatory framework and hence, the waste acceptance criteria for irradiated graphite differ in each member state. In a sense, characterisation needs to be an iterative approach initially to guide and optimise strategic decisions and later to confirm limitations and requirements of more detailed decommissioning plans.

It must also be emphasised that production, power generation and test reactors were not built nor operated with decommissioning in mind. Available data from the operation of the reactor will typically be of limited use to decommissioning processes. In addition, some reactors and storage facilities were built in the fifties when quality assurance and document control arrangements were not comparable to modern day standards. Any characterisation programme would strongly benefit from a knowledge domain review.

This report outlines the graphite data required for all decommissioning steps. These include as-manufactured (virgin) graphite information, irradiation and storage history, physical and mechanical properties, Wigner energy, chemical properties and radiological data. The information in this section is a summary of the findings from the EU framework projects (Carbowaste and CAST) [273][274]and the IAEA projects[275][276][277][278] :

- Characterisation, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors
- Progress in Radioactive Graphite Waste Management
- Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal

---

[272] Wickham, A., Steinmetz, H.-J., O'Sullivan, P., Ojovan, M.I., 2017. Updating irradiated graphite disposal: project 'GRAPA' and the international decommissioning network. J. Environ. Radioact. 171, 34–40

[273] Metcalfe, M.P., Banford, A.W., Eccles, H., Norris, S., 2013. EU Carbowaste project: Development of a Toolbox for Graphite Waste Management. J. Nucl. Mater. 436 (1–3), 158–166

[274] Neeft, E.A.C., Carbon-14 Source Term CAST: Summary of the Progress achieved through CAST for the General Public and Decision Makers, D7.25 (2018), https://www.projectcast.eu/publications

[275] International Atomic Energy Agency, 2006. Characterisation, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. IAEA-TECDOC-1521

[276] International Atomic Energy Agency, 2010. Progress in Radioactive Graphite Waste Management. IAEA-TECDOC-1647

[277] International Atomic Energy Agency, 2016. Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal. IAEA-TECDOC-1790

[278] Wickham, A., Steinmetz, H.-J., O'Sullivan, P., Ojovan, M.I., 2017. Updating irradiated graphite disposal: project 'GRAPA' and the international decommissioning network. J. Environ. Radioact. 171, 34–40

- Irradiated Graphite Processing Approaches (GraPA).

4.3.3.1 Pre-irradiation graphite data

Graphite properties after irradiation vary significantly for a number of reasons discussed in the next paragraph. The post-irradiation properties also depend on the as-manufactured graphite properties. Collation of all available information on the graphite type and grade, manufacturing method, graphite qualification data, as well as drawings of reactor cores, positioning of channels, braces, thermocouples and other auxiliary devices is an essential exercise.

4.3.3.2 Irradiation and storage history

All graphite properties change after exposure in a reactor environment. In addition to the virgin graphite properties, the magnitude of the change depends on the irradiation temperature, dose and gas coolant chemistry. Since every part of the graphite component has experienced different irradiation conditions, graphite properties will vary throughout the component. For example, the graphite bore, being next to the fuel, has experienced higher fluence, temperature and different coolant chemistry compared to graphite in the periphery of the graphite component. Detailed records of irradiation temperature, fluence, gas coolant changes, accidents, burst fuel, oil ingress and other incidents are essential in providing a better understanding of the graphite properties for the entirety of the core and minimising further characterisation requirements.

4.3.3.3 Physical and mechanical properties

A. Density

Depending on the irradiation temperature, dose and coolant gas, the graphite density in the reactor core can be reduced due to radiolytic oxidation. This is particularly the case for Magnox and AGR cores towards the end of life where peak graphite weight loss is ca. 40 %. Without the presence of a radiation field, graphite does not oxidise at temperatures lower than 500 – 600 °C, so thermal oxidation of the graphite in the reactor is not an issue. In a pure inert atmosphere, graphite does not oxidise.

It has been shown that there is a correlation between graphite mechanical properties and density. Hence, density and its variation throughout the core are important at the dismantling stage, when it is important to assess the integrity of the components.

The density measurement will also provide a measure of the porosity in the graphite, which may be an important aspect to consider when selecting component retrieval tools. In addition to conventional density measurements (ASTM C559) on trepanned samples, an assessment of the near-surface density of the graphite components can be achieve by eddy current inspection tools. More recently, muon tomography has been proposed as an innovative, non-intrusive technology that can be used to give density measurements over the entire core.

B. Dimensional change

Fast neutron irradiation causes the graphite components to shrink or grow and change shape (e.g. bowing or barrelling) depending on the irradiation temperature and fluence and type of graphite as well as the geometry of the component. These are important considerations during the dismantling of the reactor core. In the UK, Magnox and AGR licensees have developed inspection tools used to measure the graphite brick bore diameter, tilt and length. Specifically for the AGRs, these tools incorporate a camera for visual inspection and an eddy current tool to provide information on the graphite density near the brick bore.

C. Thermal conductivity and specific heat

The specific heat of graphite does not change with irradiation. Thermal conductivity reduces by almost an order of magnitude at the onset of fast neutron irradiation and remains constant for a large range of fluence after that. Materials Test Reactor (MTR) experiments at very high doses have shown a further decrease when the graphite starts to degenerate. In addition to irradiation damage, thermal conductivity decreases exponentially with weight loss due to radiolytic (or thermal) oxidation.

Thermal conductivity is important in assessing fault scenarios and heat dissipation. It is usually calculated from thermal diffusivity measurements on graphite samples (ASTM C714).

D. Stored (Wigner) energy

Wigner (or stored) energy is the accumulation of energy in the graphite crystal lattice as a result of defects caused by fast neutron irradiation. At sufficiently high temperatures, thermal vibrations in the lattice will re-arrange the atoms towards the perfect lattice state, releasing this stored energy as heat. After the fire at the Windscale Pile 1, extensive work was undertaken to understand the conditions of accumulation and release of Wigner energy. It is now known that there is a potential for release of Wigner energy only if the graphite temperature is raised at least 50 °C above the irradiation temperature.

Stored energy is measured on irradiated graphite samples using the differential scanning calorimetry (DSC) method.

E. Strength

Graphite strength is important in determining the block retrieval method at the dismantling stage. At the onset of irradiation, graphite strength increases significantly and then remains constant until a relatively high dose, when it increases again. The magnitude of the increase depends on the graphite type, irradiation temperature and fluence. However, strength is reduced exponentially with weight loss caused by radiolytic (or thermal) oxidation.

The type of strength (compressive, flexural or tensile) required will be determined by the retrieval method considered. Graphite strength is measured on graphite samples using the ASTM standards C695, D8289, C651, C565.

F. Elastic modulus

Similarly, elastic modulus is important in determining the strain on the graphite components during dismantling. At the onset of irradiation, the elastic modulus increases significantly and then remains constant until a relatively high dose, when it increases again. The magnitude of the increase depends on the graphite type, irradiation temperature and fluence. However, modulus is reduced exponentially with weight loss caused by radiolytic (or thermal) oxidation.

There are three main methods of measuring elastic modulus on graphite samples: i. using strain gauges during the tensile strength testing (ASTM C565), ii. using the ultrasonic time-of-flight method (ASTM C769) or, iii. using the sonic resonance method (ASTM C747).

4.3.3.4 Chemical properties

A. Reactivity

Graphite is a material of low chemical reactivity and is used safely in extreme temperatures and pressures. The use of air as coolant for the production and test reactors operating up to 200 °C shows that there is little adverse effect on as-manufactured graphite at these temperatures.

There are numerous reports on irradiated graphite reactivity with air. Deposit concentration and reactivity and graphite activation energy were part of the periodic Post Irradiation Examination (PIE) programmes for the life of the Magnox reactors. The work was motivated by the Long Term Graphite Transient (LTGT) fault safety case study for Magnox reactors; this is referred to as breach of pressure circuit in the earlier reports. The LTGT is an assumed fault scenario leading to depressurisation of the reactor pressure vessel and subsequent air ingress to the core.

The vast majority of the studies, if not all, measure the reactivity of graphite and deposits in air at a temperature of 400 – 550 °C. The Differential Thermal Oxidation method uses a thermogravimetric analyser (TGA) to remove the carbonaceous deposits at 450 °C first, followed by determination of activation energy at isothermals of 450, 500 and 550 °C.

It is noteworthy that graphite oxidation rates can be increased by catalytic impurities and therefore care must be taken in deciding the dismantling methodology for the graphite and other ancillary equipment in the core. Most importantly, care must be taken during treatment when graphite has been stored with other materials in ponds and silos.

B. Explosibility

There is no requirement for further investigations of the explosibility of graphite and graphite dust. The possibility of a 'graphite fire' has occupied many scientists and regulators when considering nuclear graphite storage, treatment and final disposal. A substantial body of evidence on this topic has been produced in the last decade and it is now accepted[279] that, in order for graphite to 'burn', the following conditions must be met simultaneously:

- A minimum temperature of 900°C
- Maintenance of this temperature either by heat of combustion or by some outside energy source
- An adequate supply of air or oxygen
- The gaseous oxidant source must flow at a rate capable of removing gaseous products but without excessive cooling of the graphite surface
- A suitable configuration of graphite and oxidant.

The same IAEA report also provides evidence on the low likelihood of graphite dust explosibility280 presents the criteria, all of which must be satisfied before a dust explosion can be initiated:

- The dust must be combustible
- The dust must be airborne, implying a need for a turbulent gas flow
- The particle size must be optimised for flame propagation
- The dust concentration must fall within an explosible range (i.e. neither too high nor too low)
- An ignition source of sufficient energy to initiate flame propagation must be in contact with the dust suspension (i.e. the use of thermal cutting devices should be avoided)
- The atmosphere in which the dust is suspended must contain sufficient oxygen to support combustion.

C. Galvanic corrosion

Galvanic corrosion occurs when two dissimilar metallic materials are in contact through a corrosive medium and is exhibited by the accelerated corrosion of the least noble of the two materials. For example, it has been shown that stainless steel corrosion is accelerated when in contact with graphite. This needs to be taken into account when considering the different packaging options and the requirements for container integrity for hundreds of years. For the case of ILW, this issue is addressed by using the concrete lining in the 4m box. It appears that this is not used for LLW graphite waste and this decision may require reviewing.

There are currently no investigations into galvanic corrosion. The three methods of measurement are galvanic series, polarisation curves and galvanic current measurements.

4.3.3.5 Radiological data and leaching behaviour

There are two main routes leading to radioactivity of irradiated graphite. The first route is the activation of the impurities present in the material from the manufacturing and machining processes. The second process is the activation of material carried by the gas coolant and deposited on the

[279] International Atomic Energy Agency, 2006. Characterisation, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. IAEA-TECDOC-1521
[280] International Atomic Energy Agency, 2006. Characterisation, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. IAEA-TECDOC-1521

graphite. In terms of graphite waste management after the initial storage period (10 years), the most important nuclides to consider are Co-60, H-3, C-14 and Cl-36. A number of work programmes have advanced understanding on C-14 formation and leaching behaviour[281][282][283][284][285][286] but further work is required

for H-3 and Cl-36. Co-60 is measured by gamma spectrometry while the beta emitters are obtained by pyrolysis (or oxidation) and liquid scintillation.

4.3.3.6 Characterisation methods

With the exception of the possibility of in-situ density measurements by eddy current technology (near the component surface only) and muon tomography, all characterisation requirements can be met by conventional laboratory measurement methods. The ASTM standard methods for the measurement of physical and mechanical properties typically require large samples that are often difficult to obtain from the reactor. ASTM STP1578[287] captures state of the art measurement methods and analysis and the ASTM Standard Guide D7775 provides guidance for measurements on small graphite specimens.

Due to the variability of the graphite properties, a statistically significant number of samples is essential to inform the decommissioning steps. A full irradiation and storage history can assist in reducing the characterisation requirements.

The characterisation requirements and standard methods currently used are shown in the following Table 4.3-3 characterisation requirements and standard methods currently used..

[281] International Atomic Energy Agency, 2006. Characterisation, Treatment and Conditioning of Radioactive Graphite from Decommissioning of Nuclear Reactors. IAEA-TECDOC-1521

[282] International Atomic Energy Agency, 2010. Progress in Radioactive Graphite Waste Management. IAEA-TECDOC-1647

[283] International Atomic Energy Agency, 2016. Processing of Irradiated Graphite to Meet Acceptance Criteria for Waste Disposal. IAEA-TECDOC-1790.

[284] Wickham, A., Steinmetz, H.-J., O'Sullivan, P., Ojovan, M.I., 2017. Updating irradiated graphite disposal: project 'GRAPA' and the international decommissioning network. J. Environ. Radioact. 171, 34–40.

[285] Metcalfe, M.P., Banford, A.W., Eccles, H., Norris, S., 2013. EU Carbowaste project: Development of a Toolbox for Graphite Waste Management. J. Nucl. Mater. 436 (1–3), 158–166.

[286] Neeft, E.A.C., Carbon-14 Source Term CAST: Summary of the Progress achieved through CAST for the General Public and Decision Makers, D7.25 (2018), https://www.projectcast.eu/publications.

[287] Tzelepi, N. and Carroll, M. eds., Graphite Testing for Nuclear Applications: The Significance of Test Specimen Volume and Geometry and the Statistical Significance of Test Specimen Population. (West Conshohocken, PA: ASTM International, 2014), https://doi.org/10.1520/STP1578-EB.

Table 4.3-3 characterisation requirements and standard methods currently used

Characterisation requirement	**Method**	**Available standard**	**Alternative technology**
Density	Weights and mensuration or immersion	ASTM C559	In-situ inspection by eddy currents Muon tomography
Dimensional change	In-situ visual inspections	N/A	
Thermal conductivity	Thermal diffusivity	ASTM C714	
Stored (Wigner) energy	Differential Scanning Calorimetry	N/A	
Strength	Compressive, flexural, tensile strength tests	ASTM C695, D8289, C651, C565	
Elastic modulus	Ultrasonic time-of-flight, sonic resonance, strain gauges	ASTM C747, C769	
Reactivity in air	Differential Thermal Oxidation	N/A	
Galvanic corrosion	Galvanic series, polarisation curves, galvanic current measurements	N/A	
Radiological characterisation	Gamma spectrometry, pyrolysis and liquid scintillation		

4.4 Technologies for hard to access areas (high walls, embedded components, harsh environment…)

Characterisation is one of the key activities in decommissioning preparation but also throughout the entire decommissioning project. It plays an essential role in providing the necessary confidence and understanding about the initial/current state of the facility and offering an important input for both the dismantling and waste management planning.

Having, as soon as possible, radiological characterisation data is critical for dismantling planning and for adapting the license documentation to the new risk profile. A detailed knowledge of the initial radiological status of the nuclear plant must be complemented by a detailed physical and radiological Inventory of the site, as it is a key element to integrate the dismantling and waste management activities. This information will be used to define the selection of dismantling techniques, the design of auxiliary systems and facilities supporting waste management or the estimation of radiological impacts to workers and the public.

A well-performed characterisation reduces uncertainties, associated with the execution of the decommissioning tasks, and offers different outcomes:

- understand conditions of the facility – radiometric, chemo-toxic, biological, physical and structural;
- define amount, location and composition of contaminants (radiological and non radiological) and the associated physical parameters;
- support a categorisation of site areas in contaminated, potentially contaminated and non-contaminated areas as a basis for zoning or implementation of a graded approach for clearance.

Characterisation activities must be planned and delivered in a structured way at each phase of decommissioning assuring coordination with the dismantling and decontamination works. Characterisations occur throughout the decommissioning process and are refined to provide increasing detail and information to support the project as it progresses.

To achieve a complete characterisation it is necessary to carry out several types of measurements in hard to access areas due to different constrains (lack of space, working at height, radiological conditions, etc.)

Characterisation of systems, structures, components and land areas is typically performed using manually delivered systems. Characterisation surveys intentionally follow a prescribed protocol with respect to parameters such as:

- Areas covered
- Scan speeds
- Required detector offsets

However, the characterisation of specific components which are highly contaminated or the accessibility to particular locations in terms of spaces in nuclear facilities, represents a challenge. The use of remote control vehicles / robots for implementing an automated inspection reduce risks for workers from industrial safety and health physics point of view.

Substantial cost savings may result from developing the means, preferably real-time, minimally invasive and field usable, to locate, identify and quantify contaminants.

The challenge under this topic is to optimise the use of remote equipment and in situ characterisation technologies to ensure more complete and cost-effective characterisation of the facility. Another challenge is to increase the reliability and quality of characterisation data collection and measurement data analysis and interpretation.

Robots are being used at decommissioning projects in diverse ways. Although robotics have been used in nuclear power industry for over 30 years, their mainstreaming into the performance of D&D tasks lags far behind that of other robotics industrial and service sectors. The high cost of development of robotics technology as an obstacle to obtaining a suite of robotic and/or remote technologies (platforms and tools) for efficient operations in high radiation or contaminated areas.

Robotic capabilities have been used in several decommissioning projects for inspection, sampling and surveying in hard to access area due the severe radiological conditions (high dose rate, alpha contaminants, etc.). The use of robots has been usual in several nuclear facilities after accidents (Chernobyl, Fukushima, Three Mile Island) due to the specific conditions.

The implementation of the methodology of surface clearance involves the characterisation of extensive areas of the buildings in order to allow later demolitions. Characterisation of some of these areas require to measure in hard to access areas because of the structural design and configuration of the plants. For example, it is necessary to reach high elevation walls or ceilings of the different radiological buildings. The application of this methodology involves different types of measurements in an iterative process implying an important effort from the logistic point of view. For instance it can involve a massive use of scaffolding or lifting platforms during long periods with the subsequent costs for the project. Considering the repetitive effort of performing surveys over large areas, automation of radiological characterisation has a high potential to be successful. Therefore, there is an opportunity to optimise this process by using new technologies based on remote control equipment. The use of these systems could improve cost effectiveness and safety linked to these operations.

Figure 4.4-1 Surface characterisation (Jose Cabrera Reactor Building)

In the recent years several decommissioning projects have employed drones for inspection or characterisation tasks trying to adapt technology which is usually deployed in other sectors or industries. Drones, which usually carry cameras, can deploy a huge range of equipment (sensors, detectors, etc.) to analyse and interact with the surroundings around them. This technology can be used for different purposes in decommissioning projects: remote sensing of soil contamination, radiation measurements, 3D mapping, emergency response, visual images for system inspection, etc. The capability of collecting data is also needed. From the experience in different projects there are several areas to improve: capability of moving detectors required for accurate measurements due to their weight, autonomous navigation and flight stability in the interior of buildings, etc

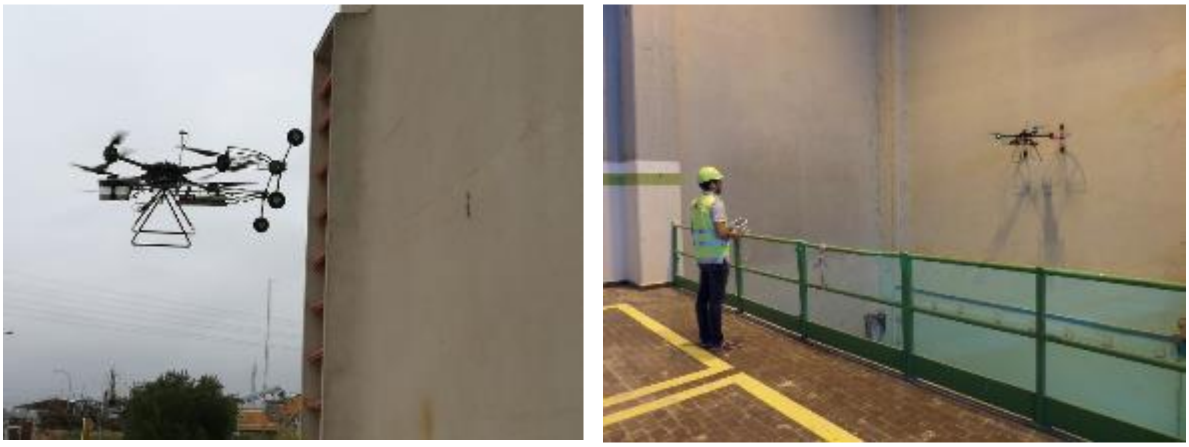


Figure 4.4-2 Use of drones for characterisation purposes (Jose Cabrera NPP)

As mentioned previously characterisation of systems, structures and components (SSC), as well as potentially impacted land areas of the site, must be conducted for all decommissioning facilities regardless of their size and use. This process includes the measurement of special components as embedded pipes or singular structures. The specificity of these elements requires to adapt the existing equipment in a case by case basis. Nevertheless, these elements can be similar in the different nuclear facilities representing an opportunity for development and standardisation of technologies and processes.

Figure 4.4-3 Characterisation of the stack (Jose Cabrera NPP)

4.4.1 Experiences/Case studies

4.4.1.1 EPRI 2020: Design and Demonstration of an Autonomous System for Radiological Characterisation (large land areas and floors)

This report[288] documents the development and demonstration of a system that integrated an autonomous vehicle owned by the Electric Power Research Institute and industry-available radiation detection and analysis equipment to perform characterisation and final status surveys.

4.4.1.2 Jose Cabrera NPP, Spain GUALI (Gamma Unit Advanced Location Imager)

A versatile, compact and portable gamma-ray imaging system allowing operators to ma radioactive sources in contaminated environments, as well as precisely determine the radioactive contamination distribution and activity (see also Section 4.8).

[288] EPRI report 000000003002018420, 2020

4.5 Modelling and simulation software for characterisation of activated or contaminated irradiated components

The predictive capability of numerical simulation is used ahead of decommissioning projects to complement in situ measurements.

Numerical simulation is used in the first instance to determine the source term, which characterises the facility's radiological conditions, i.e. the inventory of radionuclides present and the dose rate.

For instance, for nuclear reactors, this allows the type, concentration, activity and radiotoxicity of any type of nuclide to be determined for each structure in the reactor block for a given cooling time, i.e. at any moment after reactor shutdown.

Once this information about the source term is known, it can be used to determine other useful physical quantities, such as dose rates, categories of waste to be produced, etc. in order to define the best decommissioning approach.

4.5.1 Description of available solutions[289]

4.5.1.1 Neutron activation and dose rate calculations

Following Figure 4.5-1 shows the method for predictive source term and associated dose rate calculations (for gamma rays in particular), which involves three separate steps:

- Calculation of the neutron flux in the different reactor structures; this flux is proportional to the reactor power. This is then used to calculate the relative reaction rates for neutron activation of the structures, required for the next stage of the process,
- If there have been any significant modifications in the reactor structures that can influence the neutron fluxes, all the historical periods have to be modelled separately.
- Calculation of the concentrations of radionuclides formed under this neutron flux and after final shutdown; this calculation sequence uses historical operating data relating to reactor power during the irradiation period and cooling time
- Original data on activating impurities is construction materials can be insufficient for activation calculations. This data should be updated with accurate composition measurements using samples. Collecting these samples (or even having identical reference materials) well before starting the decommissioning project improves the calculations significantly.
- Calculation of the radiation transfer, which maps the equivalent gamma dose rates in the volumes of matter and the gamma and beta spectra at the measurement points of interest.

[289] Monograph on "Decommissioning of nuclear facilities", E-DEN, © CEA Paris-Saclay, Éditions du Moniteur, Paris, 2017, *ISSN1950-2672*

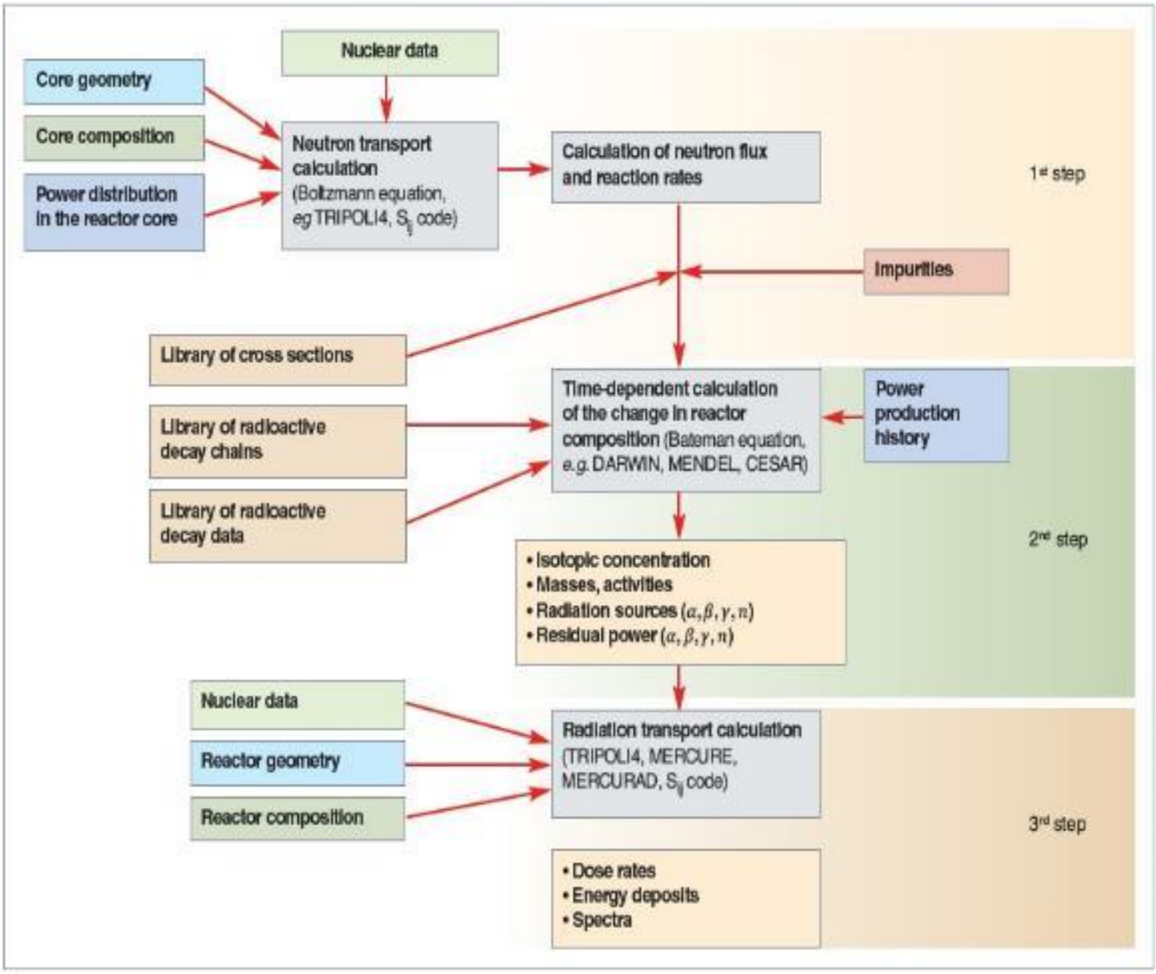


Figure 4.5-1 The calculation process used in engineering studies for nuclear decommissioning and waste management

Existing codes usually share the straight line 'attenuation' method, a semi-empirical 'simplified' method for gamma photon propagation in matter:

- MARMER (Netherlands)
- PANTHERE (EDF, France),
- MERCURE, MERCURAD and NARMERC (CEA, France)
- NARVEOS (CEA/ORANO, France)
- MICROSHIELD (USA)
- QAD (USA)
- MCNP (USA)
- SERPENT (Finland)

NARVEOS, PANTHERE and MARMER were developed using the MERCURE calculation

Main characteristics of these codes:

- Type of code: radiation propagation in matter
- Particle type/energy range: 10 MeV to 15 keV photons

- Method: straight line attenuation with build-up factor
- Geometry: 3D surface (1st and 2nd degree) for MERCURE, same as TRIPOLI-4® for NARMER
- Calculated quantities: dose equivalent rate, heating, uncollided flux, air exposure
- Type of calculations: fixed source problem
- Study type: radiation protection, instrumentation, project
- Cross-section data library: 195 energy groups from10 MeV to 15 keV
- Build-up factor data library: 195 energy groups from 10 keV to 10 MeV; single-layer buildup factors for all Z elements 1 to 99, 21 relaxation lengths between 0.5 and 50; buildup factors for 42 double-layer configurations of 7 materials (N, H2O, Al, Fe Ba, Pb and U)
- Implementation: Fortran language for MERCURE; C++ and Python languages for NARMER; dedicated HMI for MERCURAD
- Type of machine: PC with Windows and Linux
- Operating mode: single processor
- Typical execution time: ranging from a few seconds to several hours
- Validation: comparisons with measurements; benchmarks with 'exact' Monte-Carlo codes

4.5.1.2 Prediction of radionuclide migration

Some operations need to take account the occurrence of radionuclide migration. This phenomenon, initiated and governed by the physico-chemical conditions, can have a major impact on the location and level of radioactivity: hence, it is a source of contamination of exposed structures in nuclear facilities.

This is the case for tritium, a particularly mobile radionuclide produced in reactors, or because of the corrosion of metal structures in the primary system and the formation of deposits of activated corrosion products containing activation products like cobalt-58 or 60.

Ruptured fuel rods represent another source of contamination in the primary circuit from fission products (such as xenon, krypton, iodine and strontium) and actinides (uranium, neptunium, plutonium, americium and curium).

In fuel cycle facilities, the contamination migration phenomena can even be dominant: this is certainly the case for reprocessing plants, in which fuels are dissolved and their component parts separated using hydrometallurgical processes which 'break down' the structures. Waste vitrification, however, can cause volatilisation of some elements, such as caesium and ruthenium. Fuel fabrication plants are also worth mentioning, given that they process powdered materials which migrate easily. Chemical transport modelling is thus essential in these facilities to determine the source term.

Lastly, containment barriers can sometimes leak which causes contamination of such barriers in nuclear facilities and subsequent soil contamination. This contamination is itself likely to migrate according to the laws of chemistry transport coupling.

Identifying the activity from contamination and activity from activation requires measured data. However, the key-nuclide for these processes is often different (e.g. Co-60 for activation and Cs-137 from fuel contamination), therefore the other nuclides in the nuclide vector can be estimated from using the key nuclides and finally summing up the two nuclide vectors. Nevertheless, forming the nuclide vector for contamination requires understanding the chemical behaviour of contamination products.

4.5.1.3 Modelling of migration phenomena in the reactor primary system

Analysis of the radiation emitted by surfaces shows that it is produced by radionuclides such as Co-58, Co-60, Mn-54, Cr-51, Sb-124 and Ag-110m. In addition, in the case of cladding failure, fission products and sometimes fuel can be observed in the primary system. In the context of decommissioning, surface contamination is one of the key issues. It is associated primarily with neutron activation of corrosion products transported by the primary fluid. This system contamination process exists because chemical and mechanical interactions occur between the different solid and liquid

phases present in the primary system and transport phenomena (convection and diffusion) are present within these phases.

The following modelling methods are therefore used to predict radionuclide migration in reactors:

- Multiphysics modelling to describe:
    - The physico-chemical behavior of the species in the different solid and aqueous phases,
    - Activation of corrosion products as unwanted radionuclides
    - Radioactive decay,
    - The thermo-physical conditions for convection and diffusion transport of radionuclides in the solid and aqueous phases in the primary system,
    - Mechanical action of the flow on surfaces,
- Multi-scale, spatio-temporal modelling to describe:
    - The vastly differently-sized sub-systems in the primary system: for instance, the mass of oxide present between the metal and the primary fluid is several orders of magnitude less than the mass of water in the primary system,
    - The considerably different characteristic times, such as nominal operating time, transient physico-chemical reactor shutdown/restart conditions, or water circulation time in the primary system.

The complexity of the system to be described is reflected by the implementation of computational codes like OSCAR[290] and EKINOX from the multi-scale MATIX_P simulation platform, to determine the prevailing dose rate in the maintenance areas of the primary system.

Among the numerous mechanisms taken into account by the OSCAR code, the physico-chemical aspects of the chemical reactions between components can be expressed by chemical reactions: in this case, another chemistry code (PHREEQCEA) is used to solve the law of mass action equations in the physico-chemical conditions prevalent in reactors (high temperatures and pressures).

All these codes (OSCAR, PHREEQCEA, etc.) are solvers for a relatively complex set of physico-chemical equations which need to be solved either simultaneously when there is a strong coupling between the physico-chemical phenomena or separately when couplings are negligible. Realistic input data is also required for these codes to work. Such data includes the compositions of the materials used in the primary system of a particular reactor, the operating parameters imposed over the plant's service life,

---

[290] J. B. GÉNIN et al., "The OSCAR code package : a unique tool for simulating pwr contamination", NPC 2010, Québec City, October 3-7, 2010.

and the thermodynamic and kinetic data for the chemical species present in the primary system. These data volumes are managed in databases like the CEA's BDCEA- OSCV1 database, which supplies data for the computational codes, or BAMCO, which is used to validate codes based on operating experience gained from reactors. Ultimately, these calculations help to guide actions taken by operators to significantly reduce dose rates and thereby reduce personnel radiation exposure.

4.5.1.4 Modelling of migration phenomena in fuel cycle facilities

Modelling of migration phenomena in the environment

Migration of a contaminant into the environment involves numerous processes which will govern the contaminant's behaviour: the physical form in which it is released (soluble or particulate), its chemical form interacting with other chemical, organic or biological compounds in its environment (free form, complexed form, solubility, bioavailability, etc.), the physical properties (porosity and texture) and transport parameters of the porous medium into which the contaminant is migrating, as well as its mineral composition which conditions its capacity to retain contaminants (including sorption and coprecipitation of phases), and finally the prevailing hydrological conditions associated with climatic factors (rainfall, evaporation, etc.). All these processes must be incorporated into migration models coupling the reactive properties of the medium with the transport properties, with the aim of assessing, in the case of contamination, the contaminant transfer time from the nuclear facility to potential outlets.

The most commonly used transport models incorporate an empirical description of the retention properties of the porous medium through which contaminants are likely to migrate.

This representation of retention can be expressed using a solid–solution partition coefficient (Kd (l/kg) = [RN] adsorbed /[RN] in equilibrium solution).

However, these models only offer a limited predictive capability, since the partition coefficients are contextual values which cannot be transposed in conditions other than those for which they have been determined. To be able to develop models that are truly predictive, it is essential to couple the transport models with a more detailed description of the chemical reactivity of the porous media. A thermodynamic database specifically describing the sorption properties of the main mineral phases of natural media (including clay, oxide and carbonate minerals) in relation to the main elements of interest (Cs, Sr, U, Pu. etc.) is currently being developed. It will then be incorporated directly into the chemistry-transport coupled computational codes.

Models for radionuclide retention on complex natural organic matter such as soil and sediment are already applicable in static and dynamic laboratory conditions and provide input data for transport codes used on site.

4.5.2 Experiences/Case studies

4.5.2.1 Calculating activation structures

The afore-mentioned computational tools have been used for decommissioning of CEA and EDF nuclear facilities[291],[292],[293],[294],[295],[296],[297],[298],[299],[300]

Examples of such studies include dismantling studies conducted on the G2 and G3 reactors at CEA Marcoule, the UNGG fleet (graphite-moderated gas-cooled reactors fuelled with natural uranium) Chinon A1, A2 and A3, Bugey 1 and Saint-Laurent A1 and A2, the EL4 heavy water reactor at Brennilis, the Chooz A PWR, and the CEA's TRITON and NÉRÉIDE experimental reactors at Fontenay-aux-Roses.

The majority of these studies are characterised by complex, highly heterogeneous 3D geometries and high neutron flux attenuation over several decades. This explains the need for a computational code such as TRIPOLI[301],[302] which simulates propagation of gamma neutrons and photons in matter using the Monte-Carlo method in 3D geometries. Following Figure 4.5-2 represents one of the "TRIPOLI" models for the Bugey 1 UNGG reactor developed as part of its dismantling study. These models break down the facility into several hundreds, even thousands, of homogeneous volumes, the dimensions and shape of which are dictated not only by the physical and geometrical characteristics of the structures, but also by the compromises made between IT resources, calculation time and the accuracy required for the desired results.

---

[291] I. BRÉSARD, F. MARCEL, M. MESSAOUDI, G. IMBARD, G. BETSCH, J.-M.PARIZE and J.-C.NIMAL, "Radiological Characterization of Nuclear Reactors Structures, Calculations and Measurements Comparisons", *Proceedings of the 1998 ANS Radiation Protection and Shielding Division Topical Conference, Technologies for the New Century*, April 19-23, Knoxville, Tennessee, USA, 1998.

[292] M. EID, J.-C. NIMAL,G. CARUDEL and L. M. GREAT, "Activation calculations for dismantling– The feedback of a seven years' experience in activation calculations for graphite gas cooled reactors in France", *Proceedings of the 8th International Conference on Radiation Shielding*, Arlington, Texas, USA, April 24-28, 1994, pp. 512-517.

[293] State of the art of Monte Carlo technics for reliable activated waste evaluations, M Culioli, Orano, and S. Jenski, EDF, France, Lyon , PREDEC 2016

[294] A. TSILANIZARA, C. M. DIOP, B. NIMAL, M. DETOC, L. LUNÉVILLE, M. CHIRON, T. D. HUYNH, I. BRÉSARD, M. EID, J. C. KLEIN, B. ROQUE, P. MARIMBEAU, C. GARZENNE, J. M. PARIZE and C. VERGNE, "DARWIN: An Evolution code system for a large range of applications", *Journal of Nuclear Science and Technology*, Suppl.1, pp. 845-849, March 2000. (*Proceedings of the 9th International Conference on Radiation Shielding*, Tsukuba, Japan, 1999

[295] A. TSILANIZARA, T.D. HUYNH, L. LUNÉVILLE, C.M. DIOP et M. EID, « Les fonctionnalités du formulaire de calcul de la radioactivité DARWIN et les études de radioprotection », Journées scientifiques de la SFRP, *Codes de Calcul en Dosimétrie Radiophysique et Radioprotection*, Sochaux, 2-3 octobre 2003.

[296] A. TSILANIZARA, N. GILARDI, T. D. HUYNH, C. JOUANNE, S. LAHAYE, J. M. MARTINEZ and C. M. DIOP, "Probabilistic approach for decay heat uncertainty estimation using URANIE platform and MENDEL depletion code", *Joint International Conference on Supercomputing in Nuclear Applications and Monte-Carlo 2013* (SNA + MC 2013), Paris, France, October 27-31, 2013.

[297] J.-M. VIDAL, R. ESCHBACH, A. LAUNAY, C. BINET and J.-F. THRO, "CESAR5.3: An Industrial Tool For Nuclear Fuel And Waste Characterization With Associated Qualification",*WM2012 Conference*, February 26-March 1, 2012, Phoenix, Arizona, USA.

[298] F. LAYE and M.-C. PERRIN, "Comparison of Activation Calculations with Measurements For The Biological Shield of The Bugey1 Graphite-Gas Reactor", *5th EPRI International Decommissioning and Radioactive Waste Workshop*, October 31 – November 2, 2006.

[299] F. LAYE et M.-C. PERRIN, « Mise en oeuvre d'un schéma de calculTRIPOLI-4-DARWIN/PEPIN2 pour les études de démantèlement », *Journées SFRP: Codes de calcul en Radioprotection, Radio physique et Dosimétrie*, INSTN/Saclay, 28-29 novembre 2006.

[300] I. BRÉSARD, C. M. DIOP, J.-C. NIMAL, J.M. PALUT and J.-M. POTIER, "Determination of the neutron and gamma dose rates and heating induced by radioactive wastes in repository, geological salt formations", *Proceedings of the 8th International Conference on Radiation Shielding*, Arlington, Texas, USA, April 24-28, 1994, pp. 505-511.

[301] Benchmark study of TRIPOLI-4® for Decommissioning purposes, C. Loirec, CEA, France, DEM2018

[302] E. BRUN, F. DAMIAN, C. M. DIOP, E. DUMONTEIL, F. X. HUGOT, C. JOUANNE, Y. K. LEE, F. MALVAGI, A. MAZZOLO, O. PETIT, J. C. TRAMA, T. VISONNEAU, A.ZOIA, "TRIPOLI-4®, CEA, EDF and AREVA Reference Monte Carlo Code", SNA + MC 2013, DOI: 10.1051, published by EDP Sciences, 2014 ; *Annals of Nuclear Energy, 82C*, May 2015.

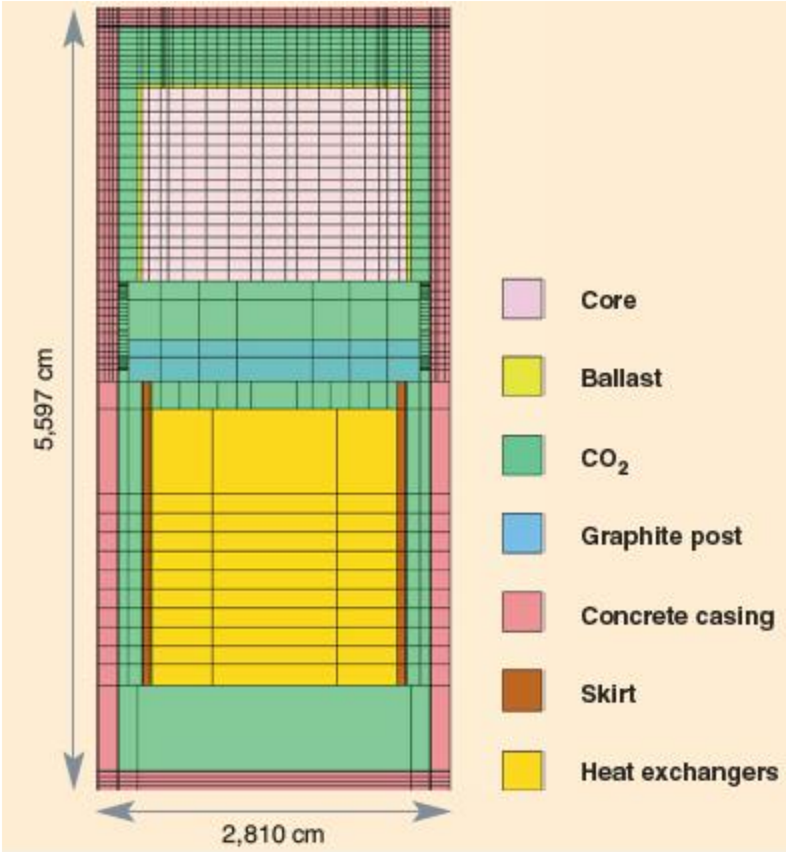


Figure 4.5-2 'TRIPOLI' model for the entire Bugey 1 reactor block

Corrosion products are generated in the primary circuit during normal operation and are activated in the core. Those activated corrosion products, mainly ^{58}Co and ^{60}Co (coming respectively from the activation of ^{58}Ni and ^{59}Co), are then transported by the primary fluid and deposited on the out-of-flux surfaces (steam generators, primary coolant pipes…).

To minimise this radioactive contamination, one needs to understand the behavior of corrosion products: ENGIE and CEA carried out measurements in PWRs and test loops combined with reactor contamination assessment code OSCAR and calculated the influence of the change in the Dissolved Hydrogen (DH) concentration on the contamination of the primary loops of Belgian unit DOEL-4 PWR. Results were compared to autoclave experiments called DUPLEX with thermodynamic and chemical conditions closed to those observed in PWRs[303]. OSCAR V1.3 calculations showed that an increase in the DH concentration results in a decrease in ^{58}Co surface activities. These results are consistent with those from the DUPLEX experiments. Finally, an increase of the DH concentration is then recommended in operating PWRs to reduce the ^{58}Co surface contamination.

4.5.2.2 Other applications for fuel cycle facilities

MERCURAD® software was deployed to determine the activity and spectra of deposits on dedicated dismantling equipment e.g. for ORANO's UP2-400 fuel treatment plant at La Hague, by coupling it with

[303] Influence of the dissolved hydrogen concentration on the radioactive contamination of the primary loops of DOEL-4 PWR using the OSCAR code, Mehdi Gherrab, Frédéric Dacquait, Dominique You, Etienne Tevissen, Raphaël Lecocq and Kim Schildermans, https://doi.org/10.1051/epjn/2020005

gamma dose rate and spectrometry measurements or to estimate the residual plutonium mass in glove boxes.

DARWIN and TRIPOLI software are used at CEA to study the phenomenon of radiolysis in cemented waste packages (see Figure 4.5-2). The calculated beta and gamma dose rates are used as input data to evaluate the formation of hydrogen in the cement matrix.

MERCURE software has been associated with the CEA's waste management platform CARAÏBES. MERCURE and SN1D codes have been used within the scope of studies conducted with ANDRA on the temporary storage/final disposal conditions for vitrified high-level waste packages in the deep geological repository in the former Asse salt mine in Germany to obtain a map of the thermal power released into the rock.

4.5.2.3 Other applications for contaminated soils

In the context of Decommissioning of INB 56 facility at CEA Cadarache, several operations were conducted to recover and condition waste both in the storage area commissioned in 1963 and in the trenches where radioactive waste was buried between 1969 and 1974. A simulation was carried out to predict the migration of various radionuclides of interest (Sr-90, Cs-137 and Pu-239+240) downstream of the storage area, assuming a constant source term over time (300 Bq/L in the case of Sr-90). Figure 4.5-3 below illustrates the simulated migration amplitude of Sr-90 in the Miocene layer directly below the facility. After 50 years of migration, the final source term activity in Sr-90 at 50 m is less than the detection threshold (0.1 Bq/L).

The following elements were required for this simulation:

- The development of a 3D hydrogeological model at different local and regional scales. The hydrodynamic parameters are matched to piezometer time lines obtained over a period of 15 years, from 1998 to 2012. Transport simulations were carried out on a local scale in continuous high-water hydrodynamic conditions to ensure a conservative characteristic,
- Verification of the retention model for the radionuclides of interest on the core sample specimens taken from the underlying aquifer formations. Definition of the retention model required a detailed mineral characterisation of aquifer sediments and acquisition of the physico-chemical parameters of the aquifer water which was monitored regularly over time.

However, these models only offer a limited predictive capability, since the partition coefficients are contextual values which cannot be transposed in conditions other than those for which they have been determined.

To be able to develop models that are truly predictive, it is essential to couple the transport models with a more detailed description of the chemical reactivity of the porous media. A thermodynamic database specifically describing the sorption properties of the main mineral phases of natural media (including clay, oxide and carbonate minerals) in relation to the main elements of interest (Cs, Sr, U, Pu. etc.) is currently being developed. It will then be incorporated directly into the chemistry-transport coupled computational codes. Within the scope of CEA facility monitoring, models for radionuclide retention on complex natural organic matter such as soil and sediment are already applicable in static and dynamic laboratory conditions and provide input data for transport codes used on site.

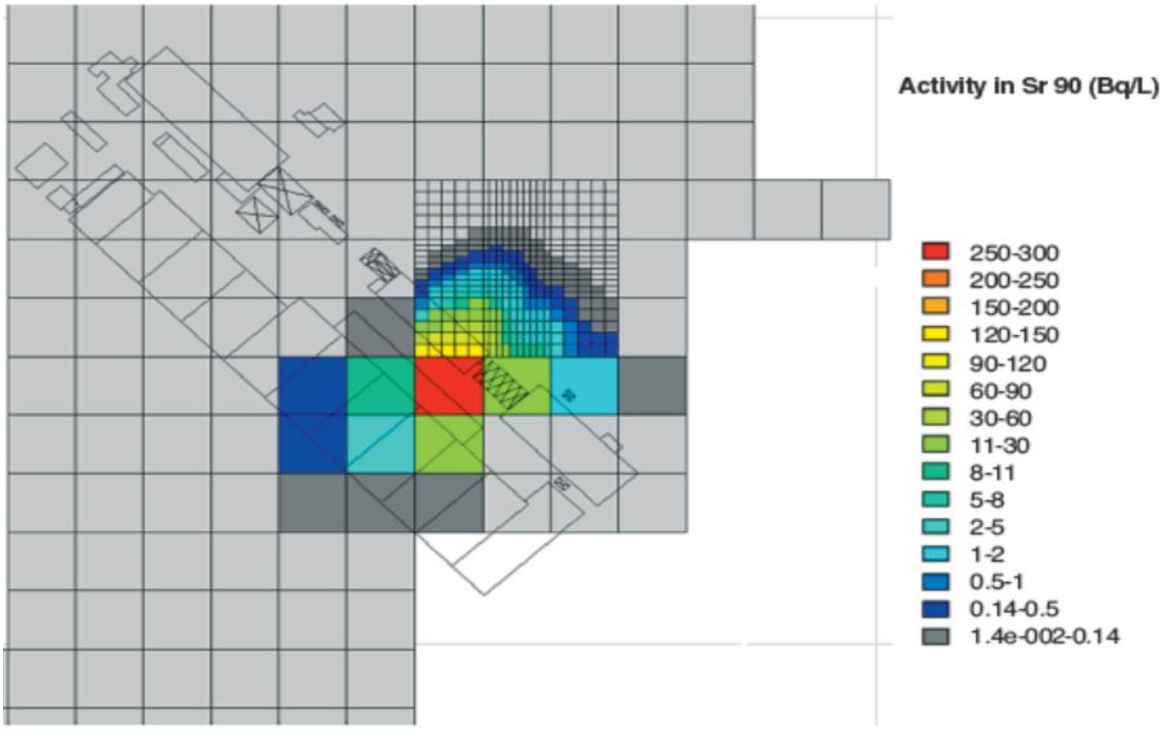


Figure 4.5-3 Simulation of the migration of Sr-90 downstream of the storage area in the INB 56 facility using the Marthe transport code (BRGM ©) with a Kd value of 0.009 m3/kg and a permeability between 1.2.10-6 and 5.10-5 m/s

4.6 Standards for statistical sampling

Drawing up a radiological inventory based on a small number of measurements is a particularly difficult statistical problem[304]. The shortage of data can lead either to a coarse over-estimation, which has large impact on economic cost, or to a coarse under-estimation, which has an unacceptable impact in terms of public health and environment protection. In the past, several attempts have been made to deal with such problems. For instance, Perot and Iooss [305]focused on the problem of defining a sampling strategy and assessing the representativeness of the small samples at hand. In the context of irradiated graphite waste, Poncet and Petit [306] developed a method to assess the radionuclide inventory as precisely as possible with a 2.5% risk of under-assessment. In a recent work, Zaffora et al. [307]described several sampling methods to estimate the concentration of radionuclides in radioactive waste, by using correlations between different radionuclide's activities. When the contamination characterised exhibits a certain spatial continuity and when the spatial localisation of measurements can be chosen, geostatistical tools can be used, as shown in ref.[308],[309].

4.6.1 Description of methods

The sampling design specifies the number, type, and location (spatial and/or temporal) of sampling units to be selected for measurement. It is based on a conceptual site model, the investigation objectives, the media to be sampled and types of data to be obtained.

An important part of the sampling design process is defining the geographical boundaries, the population of interest and dividing the site into strata based on distinct characteristics. The sampling design may also depend on whether there is surface or sub-surface contamination. Depending on the sampling objectives, a probabilistic or non-probabilistic approach to soil sampling may be adopted. Several common probabilistic sampling designs that can be employed are shown in next Figure 4.6-1. Knowledge of site history, visual inspections, and professional judgement is recommended for all sampling design strategies.

[304] Probabilistic risk bounds for the characterization of radiological contamination, Géraud Blatman, Thibault Delage, Bertrand Iooss and Nadia Pérot, https://www.epj-n.org/articles/epjn/full_html/2017/01/epjn160031/epjn160031.html

[305] N. Pérot, B. Iooss, Quelques problématiques d'échantillonnage statistique pour le démantèlement d'installations nucléaires, in *Conférence λμ16, Avignon, France, October 2008* (2008) [Google Scholar]

[306] B. Poncet, L. Petit, Method to assess the radionuclide inventory of irradiated graphite waste from gas-cooled reactors, J. Radioanal. Nucl. Chem. **298**, 941 (2013) [CrossRef] [Google Scholar]

[307] B. Zaffora, M. Magistris, G. Saporta, F. La Torre, Statistical sampling applied to the radiological characterization of historical waste, EPJ Nuclear Sci. Technol. **2**, 11 (2016) [CrossRef] [EDP Sciences] [Google Scholar]

[308] N. Jeannée, Y. Desnoyers, F. Lamadie, B. Iooss, Geostatistical sampling optimisation of contaminated premises, in *DEM – Decommissioning challenges: an industrial reality? Avignon, France, 2008* (2008) [Google Scholar]

[309] Y. Desnoyers, J.-P. Chilès, D. Dubot, N. Jeannée, J.-M. Idasiak, Geostatistics for radiological evaluation: study of structuring of extreme values, Stoch. Environ. Res. Risk Assess. **25**, 1031 (2011) [CrossRef] [Google Scholar]

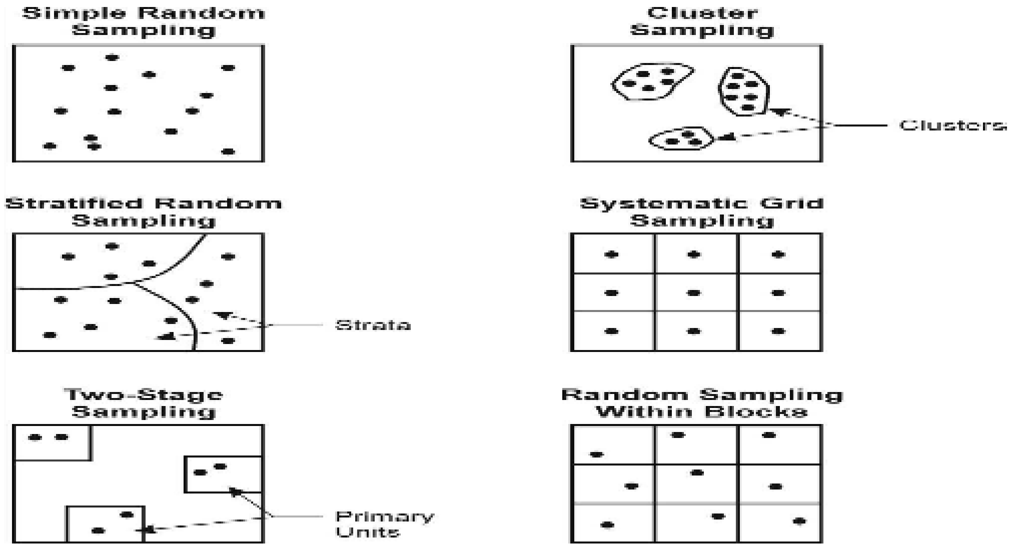


Figure 4.6-1 Some Common Two-Dimensional Sampling Designs[310]

As already mentioned to address 3D contamination, sampling designs are generally combined to improve the overall characterisation. For initial characterisation, geostatistics proves to be relevant as regards sampling definition and optimisation to tackle spatially structured contamination (see next section). For final survey, MARSSIM distinguishes three area classes according to the contamination expectation and then mixing from 10 to 100 % scan survey with statistical determination of appropriate number of destructive samples.

Along the vertical direction, contamination variations present a different spatial continuity in comparison to what can be observed on the horizontal plane. That is directly linked to the physical behavior of the contamination and the impacted matrix (sub horizontal geological layers versus gravity migration). Consequently, sampling resolution along vertical cores must be chosen carefully and adequately. In addition, remediation unit as a volume can significantly impact the vertical resolution as well.

Similarly, frequency considerations for time variations with groundwater has to be considered.

Geostatistics aims to describe structured phenomena in geographic space, possibly in time, and quantify the estimation uncertainties, whether global or local. Estimates are calculated from a partial sampling and result in different representations of the contamination, including interpolation mapping (by a kriging algorithm). But the added value of geostatistics goes much beyond this first result. Its key feature lies in its ability to quantify estimation uncertainty and provide risk analysis for decision making.

The spatial variation of a contaminant within a domain can be quantified by the variogram. The variogram is a function that shows how the variation between observations of a variable at two sites depends on the distance in space between the sites. The variogram is half the mean squared difference between two observations plotted against the distance between them for all the results in a data set. Typically, the variogram increases with distance until a plateau in the plot is reached at a

310 Guidance manual for environmental site characterization in support of environmental and human health risk assessment - volume 1 guidance manual. Canadian Council of Ministers of the Environment, 2016

value called the sill variance, which it reaches at a distance called the range. Variography is a very powerful and visual tool for the identification of outliers as well as boundaries between different spatial populations.

To illustrate differences between statistics and geostatistics, following Figure 4.6-2 shows three phenomena with the same statistical characteristics (in terms of a histogram):

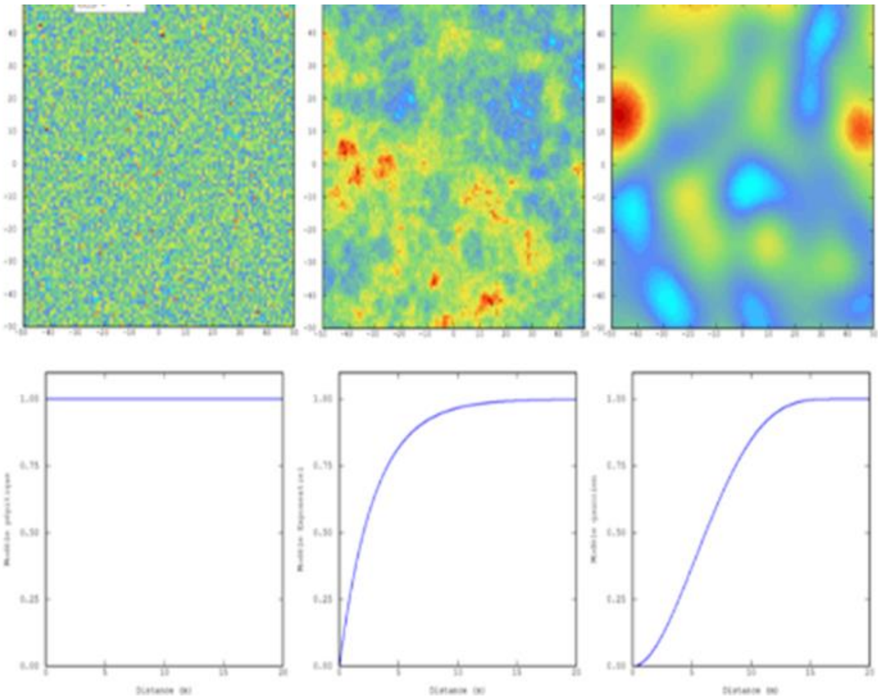


Figure 4.6-2 Example of different phenomena with the same statistical characteristics

From left to right, these are the spatial structure that can be found in nuclear decommissioning and site remediation projects for:

- No spatial structure: random background variation or heterogeneous waste in trenches/tanks/silos for instance
- Spatial continuity: contamination of soils outdoor and concrete slab indoor
- Spatial differentiability: Activation phenomenon around reactors, etc.

They have very different spatial organisations (variograms). On the left, a spatial random phenomenon with a pure nugget model as a variogram, in which the variability equals the experimental variance whatever the distance; in the centre figure, a largely continuous phenomenon with a linear increase in variability at small scale, then a plateau (variability asymptote) at the 15m range (characteristic distance); on the right, a continuous phenomenon with a progressive increase in variability at small scale, then a plateau at the same 15m range.

Initial sampling design may advantageously benefit from the knowledge of the variogram as the range is linked to the size of the contamination. Therefore, the sampling mesh can be optimised to suit to this typical spatial distance. It also can take other characterisation objectives into account, identifying hot spots for instance.

Thanks to input data and the spatial structure identified through the variogram, geostatistical techniques enable an estimate to be made of the variable being studied by kriging (best linear unbiased estimator). This interpolation always comes together with a quantification of the associated uncertainty.

More advanced geostatistical methods, such as conditional expectation or geostatistical simulations, are used to provide other quantifications of uncertainties: risk of exceeding the threshold, for instance. These estimates are thus powerful decision making-aid tools for the waste classification of surfaces and/or volumes prior to decontaminating works (based on different thresholds as well as considering the remediation support impact) and for sampling optimisation as well.

Finally, multivariate geostatistics allows different kinds of information to be combined to improve the estimation, through the spatial correlations between variables. Therefore physical/historical data (matrix, incident) and non-destructive measurement values (dose rate, in situ gamma spectrometry) are advantageously integrated to improve understanding and prediction of the main variable (laboratory analysis results, for example) while reducing the estimation uncertainty (still linked with overall sampling optimisation).

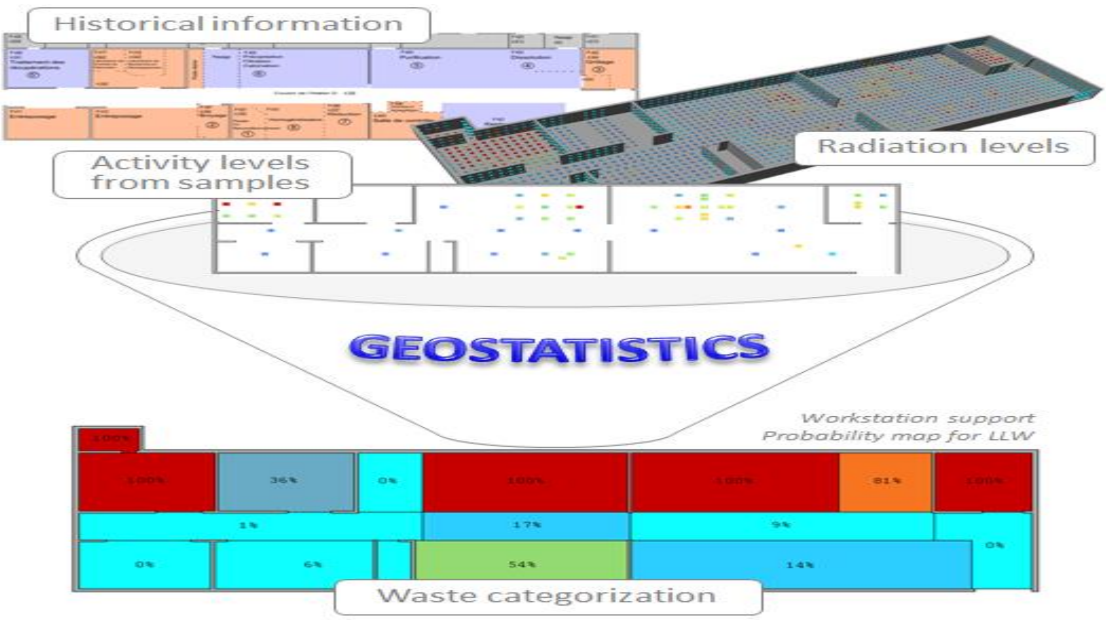


Figure 4.6-3 Waste classification map from geostatistical processing of all sources of information

In some cases geostatistics does not apply[311]. The main challenge is related to the small number of data which are usually available in real-world situations. In this context, the normality assumption is generally unfounded, especially in the case of strongly asymmetrical data distributions, which are common in real-world characterisation studies. Moreover, these are distribution-free tools and no strong assumptions are needed, e.g., with respect to the normality of the distribution of the variable under consideration. These tools are distribution statistics aids which can provide practical confidence bounds for radiological probabilistic risk assessment.

[311] Probabilistic risk bounds for the characterization of radiological contamination, Géraud Blatman, Thibault Delage, Bertrand Iooss and Nadia Pérot, https://www.epj-n.org/articles/epjn/full_html/2017/01/epjn160031/epjn160031.html

Certain concentration inequalities, used in a conservative, have shown to be strongly robust. However, the prediction and tolerance bounds given by the standard Bienaymé-Chebychev inequality are very loose. Thus, their use in risk assessment leads to unnecessarily high conservatism. If their assumptions (unimodality and tail convexity of the pdf) can be justified, the Camp-Meidell and Van Dantzig inequalities should be considered first. In the absence of any assumptions, Wilks' formula offers the advantage of directly giving an upper bound of the risk of being non-conservative, but is not of great advantage when dealing with very small-sized samples or low risk bounds. Indeed, in such cases, the excessive conservatism can be greater than when using the concentration inequalities. Moreover, Wilks' formula can suffer from a lack of flexibility in practical situations.

In terms of future directions, more recent concentration inequalities[312],[313] could be studied and may potentially give much narrower intervals. As an aside, it has also been shown in[314] how to use probabilistic inequalities to determine the precision in the estimation of the mean of a random variable from a measurement sample. With these kinds of inequalities, we can find the minimal number of measurements required in order to reach a given confidence level in estimating the mean. In conclusion, possible applications of these tools are numerous across all safety considerations based on expensive experimental processes. Further research and applied case studies could lead to the development of useful guides for practitioners, in particular in the nuclear dismantling context.

Reference documents

Most of the reference documents are based on a general workflow for radiological characterisation. Based on different data types (historical knowledge and records, in situ measurements, laboratory analyses on destructive samples, numerical models), they try to develop the methodology on 2D mapping and 3D characterisation of contaminated volumes. As evaluation objectives can cover a large variety of issues (removal of doubt, identification of hot spots, spatial distribution and/or time variation of contaminants, demonstration of compliance with clearance levels…), sampling strategy are diverse (judgmental, probabilistic, iterative, real time). And to combine all available collected values, dedicated data analysis and data processing are involved (statistics, geostatistics, numerical models).

Following Table 4.6-1 intents to present the relevance of the different reference documents according to these specific issues ("+" means introduction/summary and "+ +" means a large/detailed description).

Table 4.6-1 Relevance of the different reference documents

[312] S. Boucheron, G. Lugosi, S. Massart, *Concentration inequalities: a nonasymptotic theory of independence* (OUP, Oxford, 2013) [CrossRef] [Google Scholar]

[313] W. Hoeffding, Probability inequalities for sums of bounded random variables, J. Am. Stat. Assoc. **58**, 13 (1963) [CrossRef] [MathSciNet] [Google Scholar]

[314] G. Blatman, B. Iooss, Confidence bounds on risk assessments – application to radiological contamination, in *Proceedings of the PSAM11 ESREL 2012 Conference, Helsinki, Finland, June 2012* (2012), pp. 1223–1232 [Google Scholar]

Reference document	General workflow	Characterisation objectives	Sampling design	Surface mapping	Destructive samples and lab analysis	Data analysis	Geostatistical data processing	Final survey
IAEA, 2017				+ +		+	+	
ISO-18557	+ +	+ +	+	+	+	+	+	+
NEA, 2013	+ +	+ +	+					
NEA, 2014		+		+	+	+	+	+
INSIDER, 2018			+ +			+ +	+ +	+
CETAMA, 2014	+ +	+		+ +	+	+	+ +	+
EPRI, 2016			+			+ +	+ +	+
ITRC, 2007	+	+	+ +			+		
MARSSIM, 2000	+ +	+ +	+	+		+ +		+ +
EPA, 2002		+	+ +					
CCME, 2016	+ +	+ +	+ +	+	++	+	+	

ITRC[315]

Triad is a best management practice developed from experience in the environmental field to provide the tools for making better clean-up decisions at contamination sites. The Triad approach is built on an accurate conceptual site model (CSM) that supports project decisions about exposure to contaminants, site clean-up and reuse, and long-term monitoring. The Triad approach also incorporates application of successful work strategies and the use of technology options that can lower project costs while ensuring that the desired levels of environmental protection are achieved.

Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (2000).

The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) is a technical document for providing radiological survey approaches to United States federal agencies, states, site owners, contractors, and other private entities on how to demonstrate that their site is in compliance with a radiation dose or risk-based regulation, otherwise known as a release criterion. The MARSSIM radiological survey approach is the industry standard for radiological surveys in the United States. The MARSSIM approach is applicable to “real property” as defined in US legal practice. “Real property” consists of land, buildings and other permanent improvements fixed to the land (walls, utility piping, sidewalks and roads).

[315] Triad Implementation Guide, Overview Document ITRC, 2007.

EPA[316]

Provided technical guidance on specific sampling designs that can be used to improve the quality of environmental data collected. Based in statistical theory, each chapter explains the benefits and drawbacks of each design and describes relevant examples of environmental measurement applications. To choose a sampling design that adequately addresses the estimation or decision at hand, it is important to understand what relevant factors should be considered and how these factors affect the choice of an appropriate sampling design.

CCME[317]

A group of four documents is dedicated to the description of the general approach for environmental site characterisation. It details contaminated site management and investigation process, quality assurance / quality control, conceptual site mode for contaminated sites, soil characterisation guidance, groundwater characterisation guidance, and other guidance for soil vapour, indoor air, surface water, sediment and biological characterisation that are less in the scope of this work.

4.6.2 Experiences/Case studies

4.6.2.1 Geostatistics used for characterisation of contaminated soils at CEA[318]

The methodologies recommended in the IRSN guidelines called 'Management of sites potentially polluted by radioactive substances' and Council Directive 96/29/Euratom provided CEA with experience feedback which helped to establish a methodology in 1999, leading to the inter-operators guidelines (CEA/EDF/ORANO).

It is much easier to ascertain the radiological state of a nuclear site or facility if a direct beta or gamma radiation measurement is combined with a position (X, Y, Z) determined by a GPS with differential correction, to obtain a map on a georeferenced medium. Location by differential GPS is accurate to approximately one meter.

The areas to be characterised can vary from a few dozen square metres to several hectares, or even thousands of hectares. In most cases, the pollutants are beta-gamma emitters for which the flux can be measured with conventional detectors (NaI, gamma spectrometry or plastic scintillators).

Using this geostatistical approach, drill holes are made in areas where there is a high degree of uncertainty and variability, unlike previous practices, where almost all drill holes were in areas with the highest activity levels.

Furthermore, in addition to the measurements taken on the surface soil, it is now also possible to calculate the number of drill holes needed for a relevant radiological assessment of the deep soils.

Once the drill holes have been made, in most cases using techniques that do not involve water, so as to minimise leaching, representative soil samples should be taken in the form of core samples (or

[316] Guidance for Choosing a Sampling Design for Environmental Data Collection (EPA QA/G-5S), 2002.

[317]. Guidance manual for environmental site characterization in support of environmental and human health risk assessment. Volume 1. PN 1551, CCME, 2016

[318] Monograph on "Decommissioning of nuclear facilities", E-DEN, © CEA Paris-Saclay, Éditions du Moniteur, Paris, 2017 *ISSN1950-2672*

sections). This operation is generally preceded by gamma scanning measurement of the core sample every 10 cm, to identify the presence of any hot spots, and then sampling is carried out on the core sample.

Depending on the sampling interval, the size of the core samples can reach twenty or so metres (20 cm interval for a 2 meters core sample, 100 cm interval for a 20 meters core sample).

Each sample is measured in the laboratory using gamma spectrometry and/or radiochemical measurements of the pure alpha and beta emitters. The results are used to plot the profiles of the various radionuclides for each core sample. The study of the vertical migration mechanisms of each radionuclide* can therefore start, taking into account the nature of the soil.

As for 2D mapping, the creation of a 3D map uses geostatistics as the data analysis technique and for estimating the levels of activity. This mapping provides an assessment of the probabilities of the expected activity level being exceeded. These results are used to compare the various rehabilitation scenarios for the areas from a technical and financial viewpoint, taking planned re-use into account.

4.6.2.2 Experience at CETAMA

CETAMA produced a report [319] in 2014 to present the general methodology and best practice approaches which combine proven existing techniques for sampling and characterisation to assess the contamination of soils prior to remediation. It is based on feedback of projects conducted by main French nuclear stakeholders involved in the field of remediation and dismantling (EDF, CEA, AREVA and IRSN). The main part describes the applied geostatistical methodology with the exploratory analysis and variogram data, identification of singular points and 2D/3D mapping of the contamination.

[319] CETAMA, 2014. Soil Radiological Characterisation Methodology. Strategies for sampling and statistical and geostatistical data processing, from initial characterization through to final clean-up inspections.

4.7 Geostatistical software applications

The use of geostatistics software in the field of Decommissioning helps providing the best knowledge of the initial radiological state, prior to starting projects, defining the various contamination levels, their areas and the associated 3D volumes.

By associating an uncertainty and probabilities with the contamination map, they become an indispensable tool for the rational management of contaminated sites.

Methodologies were addressed in Section 4.6.

4.7.1 Experiences/Case studies

4.7.1.1 Visual Sampling Plan (VSP)

VSP is a software tool developed by Pacific Northwest National Laboratory with support from, amongst others, US EPA. It supports the design of data acquisition plans (sampling and/or surveying) that collect the right type, quality and quantity of data to support decision-making at the required level of confidence. Several statistical sampling designs can be selected, including random, systematic, stratified and combined judgment/probabilistic. Locations of samples are determined in VSP based on the sampling design and the required number of samples. In addition to the sampling design, VSP also provides data quality assessment and statistical analysis functions to support evaluation of data and make decision recommendations.

4.7.1.2 Kartotrak used for characterisation of contaminated soils at CEA[320]

In the context of post-incidental remediation of a site with contaminated soils, the constraint environment comes from the difficulty of collecting samples beneath a building on the one hand and the fact that samples were collected in the past with no possibility for additional samples. This task has been initiated by gathering prior knowledge for the contaminated site and analysing the available dataset (historical assessment and available data from non-destructive and destructive analyses).
Then the approach used to establish the map of the gamma flux emerging from the soil on the site was based on Real-time measuring devices developed by CEA (VEgAS®, KRP® and KRT®), and on geostatistical methods. The KARTOTRAK software platform used to collect measurements from the various detectors every second is a first all-in-one software solution designed for characterisation of soil contamination for all those in charge of environmental site assessment or remediation who need to locate and estimate contaminated soil volumes confidently. It has been conceived by CEA on the basis of software developed in mining exploration, hydrogeology and the oil industry and is now commercialised by Geovariance[321]. Kartotrak offers an integrated workflow that streamlines the characterisation process and can be used at any step of a remediation project: during the scoping and the characterisation phase or after remediation to check site compliance with remediation regulatory rules.

[320] Monograph on "Decommissioning of nuclear facilities", E-DEN, © CEA Paris-Saclay, Éditions du Moniteur, Paris, 2017, *ISSN1950-2672*
[321] https://www.geovariances.com/en/software/kartotrak-software-contamination-characterization/

It is used to construct the best possible resource map based on a limited number of field measurements[322]. As well as providing a map, geostatistical processing of the data reveals areas of interest where it is felt additional measurements are needed.

For more detail on methodology, see Sections 4.1 and 4.6.1.

[322] Post accidental site remediation – CEA, Yvon Desnoyers, Claire Faucheux and Nadia Pérot, https://doi.org/10.1051/epjn/2019060

4.8 Sample analysis technologies

Characterisation is recognised as an essential, but often costly and time-consuming step in the decommissioning process. A large number of measurement techniques are available for successful application of radiological characterisation, allowing rapid and comprehensive determination of the activities of most relevant radionuclides. For other radionuclides that are hard to detect, scaling factors can be established that relate their activities to key nuclides[323].

Major developments in the use of in situ methods of site characterisation coupled with the use of GIS and geostatistical analysis and software demonstrate the advances realised over the past decade. The merits of using geostatistics to manage the data and present it in terms that are more readily understood and which also express the level of uncertainty are discussed in previous sections. In situ, real time measurements to collect data have increased the representativeness of the data and reduced the costs and time spent. The use of mobile laboratories and drones also represent an opportunity to reduce costs and expedite the process.

The OECD NEA Report 2014[324] R&D and Innovation needs for Decommissioning Nuclear Facilities identified the following priorities for future R&D in the area of characterisation and survey:

- Develop and integrate imaging technologies with imaging software applications to characterise contaminant distributions in concrete cracks and at depth in solid materials;
- Develop and test technologies and methodologies/approaches to enable qualitative and quantitative determination of hard-to-detect radionuclide levels in solid samples without sample dissolution.
- Develop/refine equipment and instrumentation capable of identifying hard-to-detect levels in solid samples using primary or secondary particle or photon emissions. Deploy and test or develop mass spectroscopy-based systems and applications capable of supporting decommissioning characterisation efforts.

The H2020 Euratom funded INSIDER project[325] analysed the needs in terms of the developments of new techniques for sampling and measurement through a survey including a wide population of end-users having ongoing decommissioning programmes. The main results, concerning the identification of specific needs and tools for improving the characterisation process, revealed the following needs:

- Development of in-situ methods for alpha/beta emitters
- NDA metrological tools with lower detection limits
- Statistical or geostatistical software for spatial distribution of the activity and for the Scaling Factors analysis
- Optimisation of sampling and in particular determination of the representativeness of samples
- Mobile high-sensitive measurement equipment

323 https://www.oecd-nea.org/rwm/docs/2013/rwm-wpdd2013-2.pdf
324 https://www.oecd-nea.org/rwm/pubs/2014/7191-rd-innovation-needs.pdf
325 http://insider-h2020.eu/

The EMRP (EURAMET) Metrology for decommissioning nuclear facilities (MetroDecom) project addressed the needs of the decommissioning process by the development and implementation of new radioactivity measurement techniques, instruments, calibration standards and reference materials. The final report of MetroDecom project as well as some publications are public available[326]. The follow-on EMPIR project In-situ metrology for decommissioning nuclear facilities (MetroDECOM II[327]) will use results from EMRP projects ENV09 MetroRWM[328] and ENV54 MetroDecom to enable nuclear site operators to characterise waste material rapidly and accurately, throughout all stages of the disposal process, by providing validated techniques for measuring radioactivity on site, and segregating and monitoring waste.

In a recent report released by EPRI[329], the development and demonstration of an autonomous system for radiological characterisation of large land areas and floors is discussed. The project involved combining an existing autonomous robot with an existing radiation detection system. The report provides specifications, results and lessons learned, as well as a state-of-the-art review of the available technologies for site characterisation.

The H2020 Euratom funded INSIDER deliverable D5.1[330,331] contains a description of the main non-destructive techniques used for in-situ radiological characterisation of nuclear facilities subject to a decommissioning programme. The document focused on constrained environments in terms of radioactivity (medium or high radioactivity), under difficult accessibility conditions and/or in underwater interventions. It hence describes instruments usually used for environmental measurements, surface contamination measurements, gamma spectrometry, neutron coincidence measurements, and radiation cameras.

IFIC (CSIC, Spain) in collaboration with ENRESA has developed GUALI (Gamma Unit Advanced Location Imager)[332] a versatile, compact and portable gamma-ray imaging system allowing operators to map radioactive sources in contaminated environments, as well as precisely determine the radioactive contamination distribution and activity. A fundamental distinctive feature of GUALI when compared with other systems commercially available resides on its capability to geometrically recognise the environment by means of a coupled optical system.

---

326 https://www.euramet.org/research-innovation/search-research-projects/details/project/metrology-for-decommissioning-nuclear-facilities/?L=0&tx_eurametctcp_project%5Baction%5D=show&tx_eurametctcp_project%5Bcontroller%5D=Project&cHash=2ff2dd4536bc89c3988063414b8babb1

327 https://www.euramet.org/research-innovation/search-research-projects/details/project/in-situ-metrology-for-decommissioning-nuclear-facilities/?tx_eurametctcp_project%5Baction%5D=show&tx_eurametctcp_project%5Bcontroller%5D=Project&L=0&cHash=91d12fa4ed8d1bfee0ce729ae1593b09

328 https://www.euramet.org/research-innovation/search-research-projects/details/project/metrology-for-radioactive-waste-management/?L=0&tx_eurametctcp_project%5Baction%5D=show&tx_eurametctcp_project%5Bcontroller%5D=Project&cHash=31533f8774a0bc6b299635ee4f2519f6

329 https://www.epri.com/research/products/000000003002018420

330 http://insider-h2020.eu/wp-content/uploads/2018/05/INSIDER_D5_1_Inventory_of_existing_methodologies_for_constrained_environ___V1-1.pdf

331 https://www.epj-n.org/articles/epjn/pdf/2020/01/epjn190054.pdf

332 https://arxiv.org/ftp/arxiv/papers/1801/1801.04108.pdf

The French "Investing for the future" funding enables ANDRA[333] to support the following projects related to characterisation of decommissioning sites and facilities:

- TEMPORAL[334]: Gamma-ray imaging spectrometer based on a temporal imaging method for nuclear decommissioning. The TEMPORAL project is an industrial research project aimed at developing a camera that can detect gamma rays and visualise their location and intensity on an image of the mapped area (Compton camera). This camera is based on a new concept, "temporal imaging";
- MAUD[335] : Measurement by Digital Autoradiography. The MAUD project's main aim is to improve the detection of radionuclides that are difficult to measure and make it possible to provide in situ measurements via the development of a transportable system that combines activity level measurement and characterisation of radionuclides. It seeks to adapt reliable autoradiography analysis methods developed for research in biology (monitoring of biomolecules with very weak tritium content) and geology (detection of uranium in mining exploration) to the constraints of decommissioning;
- CAMRAD[336]: High-performance radiation-hardened imaging system for in-situ characterisation of nuclear waste. A radiation-resistant (hardened) imaging systems that are more versatile, compact and effective, using CMOS imaging technologies.
- ComptonCAM[337] : Development of an ultra-sensitive portable gamma camera to locate and characterise post-dismantling radioactive waste. ComptonCAM is an experimental development project that aims to produce a pre-industrial prototype of an ultra-sensitive portable gamma camera based on innovative detection technologies developed for gamma-ray astronomy instruments in space.
- TOMIS[338]: In Situ Low Dosimetric Impact Multi-Energy Tomograph. TOMIS project proposes to develop a powerful tomography tool that can be implemented in situ, for the physical characterisation of old waste, decommissioning waste, as well as possible parts of structures and equipment.

Gamma imaging techniques enable the superimposition of a colour map display, indicating the amount of emitted X- or gamma-rays, on a given optical image of the scene under study. It provides an optimal solution to track most radioactive sources from greater distances than conventional rate meters, thus significantly reducing the radiation dose received by operators. Gamma-cameras for industrial applications have recently undergone impressive upgrades in terms of lightness, compactness, usability, response sensitivity, angular resolution and spectrometric capabilities[339]. In this regard, perhaps the main technological breakthrough has so far been the development of a stereo

333 https://international.andra.fr/innovative-pre-disposal-projects-characterization-decommissioning-sites-and-facilities
334 https://international.andra.fr/sites/international/files/2019-08/Fiche%20projet%20TEMPORAL%20VF-UK.pdf
335 https://international.andra.fr/sites/international/files/2019-08/Fiche%20projet%20MAUD%20VF-UK.pdf
336 https://international.andra.fr/sites/international/files/2019-08/Fiche%20projet%20CAMRAD%20VF-UK.pdf
337 https://international.andra.fr/sites/international/files/2019-08/Fiche%20projet%20COMPTON-CAM%20UK.pdf
338 https://international.andra.fr/sites/international/files/2019-08/Fiche%20projet%20TOMIS%20VF-UK_0.pdf
339 https://iopscience.iop.org/article/10.1088/1748-0221/11/08/P08012/meta

gamma-camera[340], which is able to automatically retrieve the 3-D location of any radioactive source, regardless of its shape and volume, even when this source is behind or within an occluding object.

The H2020 Euratom funded MICADO[341] project proposes a cost-effective solution for non-destructing characterisation of nuclear waste, implementing a digitization process that could become a referenced standard facilitating and harmonising the methodology used for the in-field waste management and dismantling & decommissioning operations. As part of MICADO project, CEA, IAEP CTU and X-Ray imaging Europe carry on the developments toward an improved version of Nanopix gamma camera[342]. This second version of Nanopix embed a computational capacity close to the sensor in order to provide the camera with some intelligence. A specific Power-over-Ethernet module is designed so that the system could easy manage its connectivity and power supply and several automation steps were included along with advanced processing capabilities, making the camera easily controllable from remote location.

Dual particle imaging systems detect gamma-rays and neutrons simultaneously and can differentiate between the two radiations. This method of imaging has an advantage over single particle imaging methods because it allows the passive detection and identification of a wide range of nuclear materials and other radioactive sources. There are two main groups of systems in the field of dual particle imaging. The first group is comprised of single materials that are sensitive to both gamma-rays and neutrons. The second group uses multiple detection materials systems with detectors not necessarily sensitive to both particles. The latter imaging technique offers a reduction in system complexity, as additional discrimination techniques are not necessarily required. In addition, this category offers higher design flexibility, as the parameters employed to enhance system response to one radiation field are usually independent of the other. Gamma hybrid cameras can simultaneously achieve X-ray and gamma-ray imaging by combining features of “Compton” and “pinhole” cameras in a single detector system. Similar to conventional Compton cameras, the detector consists of two layers of scintillator arrays with the forward layer acting as a scattered for high-energy photons (> 200 keV) and an active pinhole for low-energy photons (< 200 keV). A prototype of gamma hybrid Camera for the nuclear industry[343] has been recently developed by CEA.

Real-Time Simultaneous 3D Volumetric Imaging and Mapping of Gamma-ray and Neutron Sources[344,345,346], developed at Lawrence Berkeley National Laboratory (LBNL) and University of California Berkeley, combines radiation detection instrumentation, data processing algorithms, and visualisation software to enable, for the first time ever, simultaneous, real-time imaging of both gamma ray and neutron sources (fast and slow). Radiation detectors sensitive to gamma rays and neutrons are integrated with readout electronics that allow discriminated particles to be incorporated into real-time 3D volumetric reconstructions as two separate data streams. Using the Scene Data Fusion (SDF) conceptual framework and contextual sensors from the Localisation and Mapping Platform (LAMP) developed at LBNL, gamma-ray and neutron data are fused onto contextual

---

340 https://ui.adsabs.harvard.edu/abs/2018NIMPA.910..194P/abstract
341 https://www.micado-project.eu/
342 http://www-list.cea.fr/en/media/news/2018/378-may-17-2018-nanopix-the-world-s-smallest-gamma-camera
343 https://tel.archives-ouvertes.fr/tel-02522908v2
344 https://www.osti.gov/biblio/1577135
345 https://www.mdpi.com/1424-8220/19/11/2541/htm
346 https://www.sbir.gov/node/1606129

information from the environment around the detection system. The invention is compatible with radiation detection media capable of detecting and discriminating between gamma rays and neutrons or the pairing of multiple detectors separately capable of detecting gamma rays and/or neutrons. It can be integrated into a compact, lightweight imaging platform for operation as a hand-carry device, on unmanned aerial systems, and with unmanned vehicles. The ability to measure neutron enables the detection and 3D localisation of shielded nuclear materials such as Pu-239, which is not possible with means of gamma-ray detection and spectroscopy overcoming of major limitation in the detection of such materials to-date.

The Alpha camera is quite a promising technique. It has been widely tested in realistic fields, with encouraging results and it has the potential to evolve into an industry-standard procedure in the near future. But, right now, is not widely used by the industry. CEA developed a prototype camera capable of displaying alpha radioactivity[347]. The system is based on the detection of ultraviolet radiation emitted by nitrogen when irradiated by alpha particles. The alpha contamination is localised by superimposing it on a visible-light image. All measurements must be carried out in complete darkness. Laboratory tests showed that the system is capable of detecting point sources and extended sources at levels as low as 430 $Bq.cm^{-2}$. Profile measurements of point sources revealed a scintillation bubble with dimensions corresponding to the range of alpha particles in air. The device is also capable of detecting the phenomenon through translucent materials such as glove box panels and under strong beta and gamma environments, which are not able to generate as much localised fluorescence as in the case of a-particles. The camera has been implemented for in situ examination of various fuel cycle facilities, and under these conditions has revealed alpha contamination without any breach of containment through several millimeters of Plexiglas.

Main research in the detection of alpha contamination through nitrogen radioluminescence has concentrated on the main peaks of the radioluminescence spectrum, which occur in the 300 to 400 nm range. This leads to background UV radiation from the sun or artificial lighting interfering with the detection of the alpha induced radioluminescence by masking its much weaker signal. Filtering of the wavelength of photons detected allowed for the imaging of alpha sources in dark or special background lighting conditions, but not yet in daylight. By moving away from the UVA and UVB range into the UVC range a possible route to overcoming this limitation becomes apparent. Although the peaks of intensity in this band appear to be lower, there is not the competition from sunlight and artificial light, improving the signal to noise ratio. This would potentially make detection possible on site in nuclear installations to provide characterisation for decommissioning and other purposes. A detailed analysis of the spectrum of UVC is required, including identification of any significant peaks, which may provide the best chance of detection. Other gasses may provide a better scintillation atmosphere, including in the UVC wavelength range and should be investigated. An evaluation of the alpha induced air radioluminescence detectors developed to date and their potential to develop a stand-off, alpha radiation detector which can be used in the nuclear decommissioning field in daylight conditions to detect alpha contaminated materials has been reported[348]. A Single Dual Alpha-Gamma Camera for radiological characterisation[349] and specifically for in-situ alpha/gamma measurements

[347] https://ieeexplore.ieee.org/document/1589317?denied=
[348] https://pubmed.ncbi.nlm.nih.gov/29597340/
[349] https://www.researchgate.net/publication/329495375_Development_of_a_Dual_Alpha-Gamma_Camera_for_Radiological_Characterization/link/5d23379ba6fdcc2462cae777/download

has been developed by CEA (patented in 2016). The dual camera systems using either a customised aperture and scintillators or a UV lens depending on the mode of operation. The first prototype of this dual camera consists of a pinhole or coded mask collimator and a scintillator for gamma detection, and UV optics for alpha detection, with both paths leading to an intensifier tube. The intensifier tube and UV optics are the same as those used in the S20-type alpha camera. The pinhole design was adapted and different types of scintillators (CsI, BGO, CdWO4, BaF2) were investigated. A 25 mm diameter photocathode was chosen to keep the device as compact as possible. The feasibility of gamma detection with this device was first confirmed using a pinhole collimator taken from a gamma camera with a 50 mm diameter intensifier tube. The pinhole design was then adapted to improve the spatial resolution of the images. Studies in view of producing commercial versions of the device are still ongoing but initial results and investigations of its components (scintillator, intensifier tubes, optics, shielding) are encouraging as to the viability of dual detection with a single camera. Future tests will focus on integrating the new aperture with different intensifier tubes to optimise the sensitivity and spatial resolution of the images. Peripheral shielding is being considered to minimise the effect of light pollution from sources outside the field of view and the feasibility of automatic switching between alpha and gamma optics will be investigated.

Several neutron imaging prototypes have been recently developed by independent research units mainly to detect, characterise, track and localise special nuclear material (SNM) efficiently and unambiguously are needed for nuclear non-proliferation efforts and nuclear safeguard activities that focus on nuclear material accountancy[350,351,352]. However, the challenge for the initial characterisation of nuclear facilities subject to a decommissioning programme, remains to design neutron cameras that are as compact and robust as possible, so they can be used in constrained environments while remaining sufficiently sensitive to neutrons and optimising the angular resolution. Potentially good compromises in this aspect is the coded-aperture fast-neutron imaging based on Timepix detector recently published[353]. The first prototype is a highly compact (19X15X15 cm^3, 2.2 kg) fast-neutron imager based on a MURA coded-aperture and a Timepix detector enhanced with a paraffin layer has demonstrated the feasibility of coded-aperture fast-neutron imaging based on those technologies. In addition, by adding the coded-aperture in tungsten alloy of Gampix[354], the prototype can also be used as a dual particle imager.

Although most nuclear materials emit either or both neutron and gamma-rays, heavy shielding of gamma-rays can greatly lower the efficiency of gamma-ray imaging systems, negatively impacting their efficacy in nuclear materials' detection applications. Neutrons are highly penetrating and nuclear materials emitting neutrons require bulky shielding to completely conceal neutrons. Therefore, neutron imaging systems are extensively used in nuclear materials imaging and they offer an excellent alternative. Neutron detection systems are often based on He-3 filled gas proportional counters, but the He-3 reserve is nearly depleted.

[350] https://www.ncbi.nlm.nih.gov/pmc/articles/PMC7002589/
[351] https://www.mdpi.com/2313-433X/3/4/60
[352] https://www.sciencedirect.com/science/article/pii/S0168900217312238
[353] https://www.sciencedirect.com/science/article/pii/S0168900218313664?via%3Dihub
[354] https://ieeexplore.ieee.org/abstract/document/6154706

Given the increasing need for reliable neutron detection alternatives for ^{3}He detectors the most viable options available among the crystal scintillators have been recently reviewed[355]. Given the complexity of neutron detection, various methods are required to target specific neutron energy range. Inorganic crystals utilising isotopes with high thermal neutron cross-section (lithium, boron and gadolinium) provide a very good alternative for low energy neutron detectors. However, the manufacturing cost is still high, and the growing process is long. Fast neutron region, on the other hand, has been targeted by organic scintillators for a long time, due to ^{1}H content, which allows elastic scattering of neutrons with a proton. Stilbene crystal is arguably the best available scintillator detector capable of n/g separation. Nonetheless, growing large size detectors using stilbene crystals is expensive in comparison to organic plastics and liquids. There have been attempts to develop a neutron detector targeting a larger energy spectrum. However, due to different mechanisms governing neutron interactions with matter at various energy levels, this is not possible with a single material detector. Up to date literature reports on multidetector systems, where different detectors are used independently to detect specific group of neutrons. Readout electronics attached to such system can combine the results into one system. Another method, stemming from the multidetector approach described, is based on composite detectors, where a detector such as CLYC is incorporated into plastic scintillator to detect gammas, and thermal and fast neutrons. Regardless of the target energy range, it is clear that scintillating crystals will continue to play a key role in neutron detectors.

Sellafield Ltd has recently granted Arktis Radiation Detectors UK Ltd to develop the next generation systems that detect and identify radioactive and nuclear materials. The WANDS[356] (Waste Assay Neutron Detection System) project will make use of innovative helium-4 (^{4}He) neutron detector technology to design a prototype for a mobile active neutron assay system. 4He is the most abundant of the two naturally occurring isotopes of helium. ^{4}He systems perform well with dense metallic packages and have inherent cost, weight, and performance advantages over the helium-3 based systems currently used to measure fissile material. They can achieve much lower limits of detection and it is hoped that a mobile system will support the classification of waste at Sellafield, helping to determine the most appropriate and cost effective waste disposal routes.

Muon Imaging System (MIS) technology developed by University of Glasgow and the UK National Nuclear Laboratory (NNL) with significant investment from the UK Nuclear Decommissioning Authority and Sellafield Ltd and commercialised by Lynkeos Technology Ltd have pioneered this technique for the characterisation of shielded nuclear waste containers[357] and for the analysis of thermally treated nuclear waste surrogates[358]. The muographic system uses the Coulomb-scattering property of cosmic-ray muons to passively image the contents of shielded nuclear waste containers.

The H2020 Euratom funded CHANCE[359] project address the specific and complex issue of the characterisation of conditioned radioactive waste (CRW) by means of non-destructive analytical (NDA) techniques and new methodologies. The CHANCE project proposes the development of a mobile muon tomography instrumentation to address the imaging of large volume and heterogeneous nuclear waste package. A mobile muon tomography system is being developed to address the as-yet

355 https://www.mdpi.com/2073-4352/9/9/480/htm
356 https://www.gamechangers.technology/funding-awarded-to-develop-fast-neutron-technology/
357 https://www.ncbi.nlm.nih.gov/pmc/articles/PMC6335305/
358 https://www.sciencedirect.com/science/article/pii/S0969804319306463
359 https://www.chance-h2020.eu/

unsolved problem of the non-destructive assay of large volume nuclear waste packages, such as large spent fuel casks and large concrete waste packages with heterogeneous waste[360]. A large-area demonstrator system is developed which will utilise two different technologies, namely plastic scintillator (providing timing resolution) and resistive plate chambers (providing position resolution). The system will initially be operated at a dedicated test facility using test volumes comprising materials of different Z (e.g. metal pieces, U rods, cellulose, air enclosures), encased in concrete or bitumen (simulated inactive waste drums). The performance of the system in identifying the composition and placement of the different materials will be evaluated. Other sample analysis technologies[361] being developed within CHANCE project are the calorimetry as an innovative non-destructive technique to reduce uncertainties on the inventory of radionuclides and the Cavity Ring-Down Spectroscopy (CRDS) as an innovative technique to characterise outgassing of radioactive waste.

The integration of gamma detectors and ground-penetrating radar (GPR) for non-intrusive characterisation of buried radioactive objects has been recently described[362]. The method makes use of the density relationship between soil permittivity models and the flux measured by gamma ray detectors to estimate the soil density, depth and radius of a disk-shaped buried radioactive object simultaneously. The results showed that this integrated approach is able to retrieve the key parameters of soil density, depth and radius of disk-shaped radioactive objects buried in soil of varying conditions simultaneously. It also showed that by using two horizontally-separated gamma detectors, all the measurements required for the estimation process can be acquired simultaneously, thereby reducing the time associated with sequential data acquisition. However, the method is currently limited to objects having surface radioactive contamination that can be approximated by a disk. Therefore, there is a need to develop the method further to account for objects of different shapes.

JRC Karlsruhe in collaboration with the Italian National Research Council, the European Laboratory for Nonlinear Spectroscopy in Florence (Italy) and the start-up ppqSense is developing a SCAR (Saturated-Absorption Cavity Ring-Down) system to assess the amount of ^{14}C in materials and waste produced by decommissioning. The goal of RADCAS4DEC project is to design a compact device for on-site, fast, reliable and cost effective detection of radiocarbon with sensitivity comparable to that of accelerator mass spectrometry.

Laser-induced breakdown spectroscopy or LIBS is considered a minimally destructive assay method based on the principle of ablation of a small amount of sample (10^{-12} to 10^{-9} g) by focusing a highly energetic laser pulse onto a given surface point. The ablated material then forms a micro-plasma, which almost immediately emits light photons at characteristic wavelengths, depending on the elemental composition of the sample. It is a very fast and versatile technique that can detect, in principle, all kind of materials, including impurities, and limited only by the power of the laser and the detection performances of the spectrograph sensor. In addition, its wide range of applications is largely driven by its capability with virtually no sample preparation and extremely low detection limit.

[360] https://www.chance-h2020.eu/Document.ashx?dt=web&file=/Lists/Publications/Attachments/7/EURADWASTE%202019%20Proceedings%20KI0219004ENN.en.pdf&guid=1a161ee6-a19e-4d2a-9e0f-aea5fdefc6c2
[361] https://www.chance-h2020.eu/en/Deliverables
[362] https://www.mdpi.com/1424-8220/19/12/2743

Another LIBS advantage is its ability to depth profile the sample by repeatedly discharging the laser beam on the same position, by effectively going into more and more depth with each shot. Being exclusively an elemental analysis technique it has also demonstrated its ability to provide a positive identification of fission products, actinides, and activated corrosion products has in many nuclear materials[363,364,365,366]. However, at least up till now, this technique is not in common use in the nuclear industry.

[363]https://www.researchgate.net/publication/249653245_Determination_of_Impurities_in_Uranium_and_Plutonium_Dioxides_by_Laser-Induced_Breakdown_Spectroscopy
[364]https://researchportal.helsinki.fi/en/publications/analysis-of-contaminated-nuclear-plant-steel-by-laser-induced-bre
[365]https://www.researchgate.net/publication/236594182_Exploring_laser-induced_breakdown_spectroscopy_for_nuclear_materials_analysis_and_in-situ_application
[366] https://www.osapublishing.org/as/abstract.cfm?uri=as-71-4-744

Table 4.8-1 Summary of Sample analysis technologies

Technology / Methodology	Description	Field of application/type of application	Advantages	Disadvantages/what is missing
GUALI compact and portable gamma-ray imaging system	GUALI map radioactive sources in contaminated environments, as well as precisely determine the radioactive contamination distribution and activity. Distinctive feature of GUALI when compared with other systems commercially available resides on its capability to geometrically recognise the environment by means of a coupled optical system.	In-field measurements	GUALI continuously displays a superposition of the image for the measured gamma-activity spatial distribution together with the optical one, thus aiding the quick identification and location of the radioactive sources in the measurement scenario. GUALI is capable of automatically identifying a movement or a change in the image-plane, thus triggering its own (image and gamma) acquisition systems accordingly, saving data and RGB images consistently, hereby minimising human mistakes during the decommissioning works.	
TEMPORAL: Gamma-ray imaging spectrometer based on a temporal imaging	The TEMPORAL project is an industrial research project aimed at developing a camera that can detect gamma rays and visualise their location and intensity on an image of the mapped area (Compton camera). This camera is based on a new concept, "temporal imaging", which significantly improves its performances compared to existing cameras.	• taking images of radioactive waste drums in order to check their contents (inventory, location of "hot points"); • taking images of a site or large pieces of equipment in the field of decommissioning in order to identify potential contamination zones; • being installed on an automatic sorting line in order to identify contaminated elements.	• speed: the TEMPORAL camera will provide wide angle images of gamma rays. Contaminated equipment or a decommissioning site can therefore be observed with a short exposure time; • location, identification and quantification of radioactive elements in one step: the image obtained with the camera will enable radioactivity to be located with precision, identify the radio-element detected on the image and quantify the associated contamination level	• Development phase

Technology / Methodology	Description	Field of application/type of application	Advantages	Disadvantages/what is missing
			• sensitivity: the camera will have outstanding sensitivity to low contamination levels (< 1nSv/h); • cost: the cost of the system should mean that it can be widely used in the nuclear industry.	
MAUD: Measurement by Digital Autoradiography	The MAUD project's main aim is to improve the detection of radionuclides that are difficult to measure and make it possible to provide in situ measurements via the development of a transportable system that combines activity level measurement and characterisation of radionuclides.	• Difficult-to-measure radionuclides such as alpha and beta emitters (tritium, chlorine-36) are harder to map. MAUD project seeks to achieve this more easily.	• The development of a beta and/or alpha activity measurement method that can easily be installed on site will be an important complement of current techniques, thereby increasing the flexibility of characterisation and improving the availability of information on the location and intensity of radioactivity on decommissioning sites. • to obtain images of the location of difficult-to-measure radionuclides on solid materials found in nuclear waste generated by decommissioning (mainly concrete, plastic and metal).	• The MAUD system has been recently patented. • Development phase
CAMRAD: High-performance radiation-hardened imaging system for in-situ characterisation of nuclear waste.	CAMRAD is a radiation-resistant (hardened) imaging systems that are more versatile, compact and effective, using CMOS imaging technologies. A camera with a much greater resistance to ionising radiation than existing products (cumulative dose of 1-10 MGy) with performance levels not generally found on this market (colour image, high resolution, compact design, etc.).	• visual inspection of conditioned or unconditioned radioactive waste is a significant safety issue for all waste management processes from production site characterisation to disposal.	• The inspection and monitoring of nuclear plants (particularly areas that are too radioactive to use existing cameras or rely on human intervention), disposal of radioactive waste and development of radiation-resistant emergency response robots; • the maintenance and instrumentation of nuclear physics facilities (particle accelerators) and experimental reactors; • some space exploration missions (e.g. future missions to Europa, Jupiter's moon).	• Development phase

Technology / Methodology	Description	Field of application/type of application	Advantages	Disadvantages/what is missing
ComptonCAM: Development of an ultra-sensitive portable gamma camera to locate and characterise post-dismantling radioactive waste.	ComptonCAM is an Experimental Development project that aims to produce a pre-industrial prototype of an ultra-sensitive portable gamma camera based on innovative detection technologies developed for gamma-ray astronomy instruments in space.	• The ComptonCAM camera will generally be used in a nuclear facility room during decontamination, the first dismantling phase, or for inspecting radioactive waste packages when they are moved, for example from surface storage to the disposal facility. • Its high sensitivity will enable better control of the contamination level of facilities, thereby increasing the safety of staff and minimising waste volumes • Due to its extreme sensitivity, the ComptonCAM camera will require much shorter acquisition times (minutes, rather than several hours currently) than those required for other cameras on the market.	• development of ultra-low-noise and highly compact electronic systems for picking up gamma detector signals; • use of an artificial neural network (algorithms) to optimise the response to gamma photon detectors; • production of an optimal data acquisition and processing system to generate a real-time image of the gamma-emitters in the broad field of view observed.	• Development phase
TOMIS: In Situ Low Dosimetric Impact Multi-Energy Tomograph	Tomography tool that can be implemented in situ, for the physical characterisation of old waste, decommissioning waste, as well as possible parts of structures and equipment;	• Better evaluation of the contents of radioactive waste containers, and thus make managing them more efficient. • For the recovery of old waste, introducing high-energy imaging characterisation could greatly improve its characterisation and make it possible to better determine the final solution for the waste, or lead to re-categorisation of certain	• Its transportability and adaptability make TOMIS innovative. There is currently no transportable high energy tomography system in Europe. With TOMIS, it will be possible to carry out non-destructive testing of waste containers as close as possible to their storage location without having to transport them over long distances to a dedicated facility.	• Development phase

Technology / Methodology	Description	Field of application/type of application	Advantages	Disadvantages/what is missing
		waste to lower activities (ILW-LL to LILW-SL or LLW-LL, or LILW-SL to VLLW)		
Gamma hybrid Camera	Gamma hybrid camera can simultaneously achieve X-ray and gamma-ray imaging by combining features of "Compton" and "pinhole" cameras in a single detector system. Similar to conventional Compton cameras, the detector consists of two layers of scintillator arrays with the forward layer acting as a scatterer for high-energy photons (> 200 keV) and an active pinhole for low-energy photons (< 200 keV).	• X-ray and gamma-ray imaging	• To compensate for the limitations of Compton imagery by code-mask imagery, and vice versa. • The multiplication of sources of information for the localisation of radioactive sources.	• The sensitivity and angular resolution capabilities of the hybrid single-sensor configuration must be evaluated. • The development of the new ADVACAM MiniPIX TPX3 detector paves the way for a hybrid miniature imager system whose dimensions could be similar to those of the Nanopix gamma camera.
Dual Alpha-Gamma Camera	The first prototype of this dual camera consists of a pinhole or coded mask collimator and a scintillator for gamma detection, and UV optics for alpha detection, with both paths leading to an intensifier tube.	• Radiological characterisation state of a facility prior decommissioning. • In-situ alpha/gamma measurements	• The localisation and visualisation of hot spots in the dismantling plant by combining the technology of alpha and gamma contamination detection.	• Studies in view of producing commercial versions of the device are still ongoing. No new development since 2017.
Alpha cameras	The system is based on the detection of ultraviolet radiation emitted by nitrogen when irradiated by alpha particles.	• In-situ alpha measurements	• Detection, localisation, and visualisation of alpha contamination	• The air inside the glovebox is previously enriched with nitrogen to enhance the measurement. • Accurate quantitative measurements require exact knowledge of the background lighting level, and therefore, they are best suited for applications where complete darkness is ensured. • the self-absorption of the emitter affects the radioluminescence yield and has to be accounted for when quantitative

Technology / Methodology	Description	Field of application/type of application	Advantages	Disadvantages/what is missing
				measurements are conducted
Neutron imaging systems	Ability of neutrons to penetrate objects that are opaque to gamma radiation, the corresponding imaging may be a valuable asset as a non-destructive technique during the dismantling and clean-up of nuclear facilities	• In situ measurements • Nuclear waste characterisation/classification	• Ability of neutrons to penetrate objects that are opaque to gamma radiation.	• neutron detection alternatives for ^{3}He detectors • design neutron cameras more compacts and robust so they can be used in constrained environments while remaining sufficiently sensitive to neutrons and optimising the angular resolution.
Real-Time Simultaneous 3D Volumetric Imaging and Mapping of Gamma-ray and Neutron Sources (US DOE, NLL)	Radiation detectors sensitive to gamma rays and neutrons are integrated with readout electronics that allow discriminated particles to be incorporated into real-time 3D volumetric reconstructions as two separate data streams. Using the Scene Data Fusion (SDF) conceptual framework and contextual sensors from the Localisation and Mapping Platform (LAMP) developed at LBNL, gamma-ray and neutron data are fused onto contextual information from the environment around the detection system.	• Potential applications in nuclear security, safeguards, and other source search or gamma-ray and neutron source mapping scenarios.	• Detection, localisation, and visualisation of gamma ray and neutron sources in real-time.	• Development phase
Lynkeos Muon Imaging System (MIS)	The muographic system uses the Coulomb-scattering property of cosmic-ray muons to passively image the contents of shielded nuclear waste containers. The Lynkeos MIS uses scintillating fibre and MAPMT technology to track the position of the muons as they pass through the waste	• characterisation of legacy waste, both concreted and vitrified forms • analysis of thermally treated nuclear waste surrogates	• improve waste classification and significantly reduce storage costs	• Commercially available

Technology / Methodology	Description	Field of application/type of application	Advantages	Disadvantages/what is missing
	drum, primarily 500-litre ILW containers.			
CHANCE Muon Scattering Tomography Detector	A mobile muon tomography instrumentation using two different technologies, namely plastic scintillator (providing timing resolution) and resistive plate chambers (providing position resolution).	• Non-destructive assay (NDA) of large volume nuclear waste packages, such as large spent fuel casks and large concrete waste packages with heterogeneous waste.	Muon tomography is fully passive and works for heavily shielded volumes and particularly useful to detect heavy elements like lanthanides and actinides, but can also be applied to detect density gradients or differences within a matrix.	• Looking for industry partners to guide the activities
Integrated gamma detector and GPR system	Non-intrusive characterisation of buried radioactive objects. The method makes use of the density relationship between soil permittivity models and the flux measured by gamma ray detectors to estimate the soil density, depth and radius of a disk-shaped buried radioactive object simultaneously	• Characterisation of buried wastes	Rapid characterisation of buried radioactive objects encountered during monitoring and decontamination of nuclear sites and facilities.	• integrated gamma detector and GPR system is in development phase

4.9 Alpha and beta non-destructive measurements

Alpha and beta emitters are known to be difficult to Measure Radionuclides because the mean free path of alpha particles and electrons in dense matter is very short. For the characterisation of nuclear waste, they are generally obtained via destructive laboratory analysis or via scaling factors using easy to measure radionuclides, but in this case, the uncertainties remain quite large.

4.9.1 Description of techniques

4.9.1.1 Autoradiography

Autoradiography technique has been developed at CEA for non-destructive in situ measurements of alpha and beta contamination[367].

Mainly dedicated commercially to biological researches, it refers to a radiation detection technique, where the radiosensitive material is exposed to radiations of an unknown source, in order to evaluate its activity, and locate it. It exists several technologies, but the phosphor screen technique has been developed for pure beta emitters (H-3 and C-14) and alpha (U-238).

However, the industrial diffusion of the screen technique was difficult, and it seemed important to develop new industrial technologies for alpha and beta measurements by Autoradiography. MAUD (Digital AUtoradiography Measurement) developed throughout a state-funded project[368] and H2020 project TRANSAT[369], is an innovative industrial camera for in situ and real time localisation of alpha & beta emitters. The device is currently supplied by a SME (Laumonier Company, France) which has great expertise in Autoradiography developments and industrialisation capacity.

MAUD (Digital AUtoradiography Measurement) is a cutting-edge camera for in situ alpha & beta radiation detection[370]. The technology is currently industrial and original in terms of detection technique (SiPM, Silicon Photo Multipliers), efficiency, robustness and software. The proposed technology is and will be of a great help to investigate Difficult to Measure radionuclides.

The portable camera is placed in direct contact with a raw surface to investigate. After an acquisition of few minutes the device provides the level of contamination and the image (obtained with 64 individual SiPM, Figure 4.9-1below) of the emerging radioactivity coming from the surface.

The mapping in Bq/cm2, obtained on a solid surface, is the main measurement of MAUD camera. In addition, the same sensor can be flipped and used as screening for a potentially contaminated samples or wipes. In 3 minutes, rapid screening system (contaminated or not) and α β emitters discrimination are available. MAUD (TRL 7) represents a cost-effective gain in characterisation without time consuming destructive analysis.

---

367 A non-destructive and on-site digital autoradiography-based tool to identify contaminating radionuclide in nuclear wastes and facilities to be dismantled, R Haudebourg, P Fichet, Journal of Radioanalytical and Nuclear Chemistry, 309, pages551–561(2016)
368 https://www.andra.fr/sites/default/files/2019-03/Fiche%20projet%20MAUD%20VF-FR.pdf
369 www.transat-h2020.eu
370 « MAUD Project - development of a new portable detector for alpha and beta surface contamination imaging», S. Leblond, P. Fichet, R. Laumonier, S. Billon, P. Sardini, https://hal.archives-ouvertes.fr/hal-02415476/, 2020

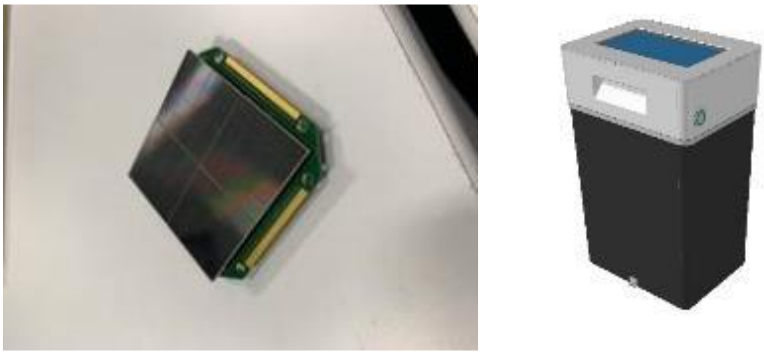


Figure 4.9-1 MAUD Camera. Left : MAUD Camera with 64 SiPM, Right : Current complete system supplied by Company ARL (Ateliers Laumonier), France

4.9.1.2 Spectrometry Beta in situ

Building on existing methods focused on the measurement of 90Sr activity in natural soil contaminated by the Chernobyl accident, this technique[371] was developed for radiological characterisation of different types of contaminated matrices, in particular the contaminated concrete structures. A measurement device equipped with an EJ200 plastic scintillator was designed using Monte Carlo simulations with the MCNP6 and PENELOPE calculation codes. The energy calibration and the response of the detector were determined using experimental measurements and MCNP simulations of laboratory configurations of standard β-sources. These data were used to validate the model of the detector, as well as to determine calibration coefficients by numerical simulation for various on-site measurement configurations.

4.9.1.3 Spectrometry Alpha

The nature of alpha particles makes them difficult to detect by spectrometry and alpha spectrometry is usually carried out in a vacuum to avoid interactions with the surrounding air.

Developments are going on[372] to work in ambient in situ conditions using a collimation method to allow the isotropic emissions to be controlled by only selecting the particles that reach the detector at close to perpendicular incidence. This reduces differences between the path lengths of each alpha particle and improves the energy resolution.

The purpose of this approach is to determine the relative proportions of alpha emitting radionuclides or groups of radionuclides

---

[371] J. Venara, M. Ben Mosbah, C. Mahé, J. Astier, S. Adera, M. Cuozzo, V. Goudea , Design and development of a portable β-spectrometer for 90Sr activity measurements in contaminated matrices, Nuclear Inst. and Methods in Physics Research, A 953 (2020) 163081

[372] D.Degrelle, J. Venara, M. Ben Mosbah, M. Cuozzo, C. Mahé, R. Serrano, "Design by numerical simulation of an *in situ* alpha spectrometer operating at ambient air pressure", *ANIMMA 2019,* https://doi.org/10.1051/epjconf/202022506004

4.9.1.4 Alpha imaging

The aim is to locate alpha contamination in situ and remotely. The phenomenon observed for this is the radioluminescence of dinitrogen in the air under the influence of alpha particles. It is accompanied by emission of photons in the near-visible UV range. The main UV emissions associated with alpha particles are concentrated close to the radioactive source, given the short distance travelled by alpha particles in air. A UV signal can be detected and integrated remotely and through translucent materials (e.g. Plexiglas). The proximity of the visible spectrum means that the UV images must be acquired in the dark or using special filters. The first camera developed at CEA [373]combines a cooled Charge Coupled Device (CCD) sensor (cooled by liquid nitrogen) which is suitable for detecting UV with a wide-angle lens for collecting UV photons. The signal is then processed on a remote standard PC. The initial tests carried out in the laboratory demonstrated the feasibility of the measurement process, creating alpha images which, when processed, are superimposed over the black and white image of the area in question provided by the camera. On this image, the sealed Am-241 source is located 60 cm away from the sensor and the acquisition is carried out in a totally light-proof box. Initial on-site test campaigns have also been carried out on this measurement system. The alpha imaging device now consists of a high-performance intensified CCD sensor using an intensifier with double micro-channel plate providing the possibility of luminous gain and high dynamic sensitivity for observing phenomena with weak photon signals. To obtain maximum quantum efficiency at the main wavelengths of the phenomenon, no 'solar blind' technology was added to the measurement device. The CCD sensor is combined with a multi alkali photocathode optimised for UV radiation. Its spectral response ranges from 180 to 800 nm. The quantum efficiency of the assembly is around 20% for a wavelength of between 200 nm and 440 nm. A standard UV lens is used to collect the UV photons with greater than 60% transmission, starting from 230 nm. The image is digitised in 16 bits. In view of its high sensitivity, the image can be used from a luminous environment of 10-6 lux.

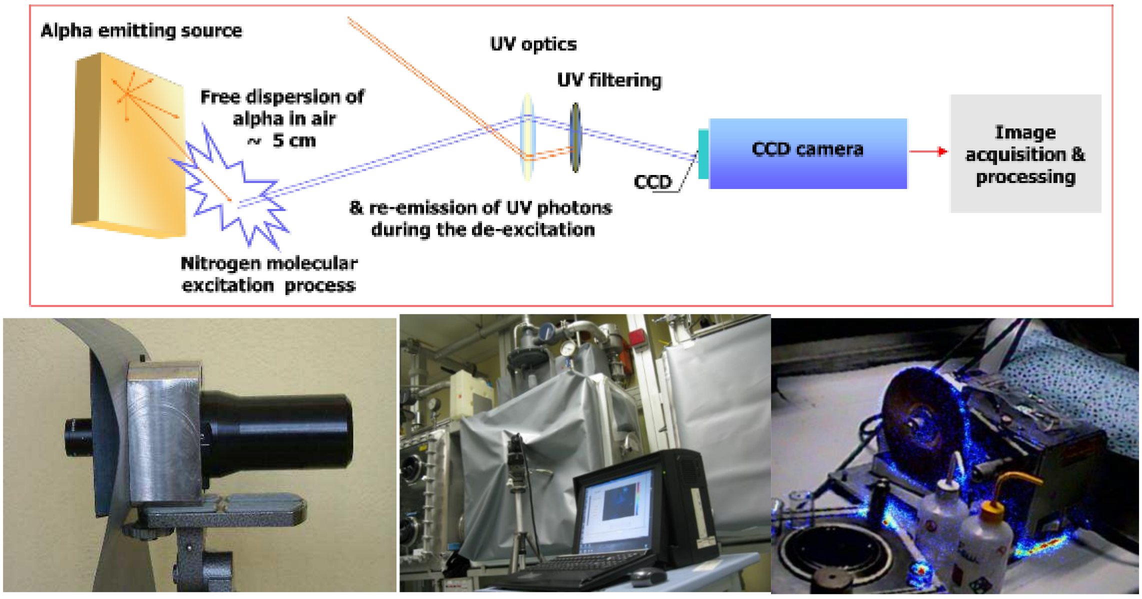


Figure 4.9-2 Camera, developed at CEA, to combine a cooled CCD sensor with a wide-angle lens for collecting UV photons

[373] F. LAMADIE *et al.*, "Alpha imaging: first results and prospects", IEEE, 2004, Nuclear Science Symposium.

4.9.1.5 Other measurement devices

Many standard beta and alpha measurements are currently obtained with wipe tests followed by standard LSC (Liquid Scintillation Counting) or alpha spectrometers. But the wipe test technique strongly depends on the type of radioactivity. The Wipe test is effective for labile radioactivity but absolutely not for non labile radioactivity.

Table 4.9-1 Summary of alpha and beta non-destructive technologies

Technology	What is working	What is missing
Autoradiography	SiPM detectors associated with scintillators are very sensitive for alpha and beta emitters on solid surfaces.	An automatic device to move and localise the MAUD device (i.e. : robotic arm)
Spectrometry Beta in situ	Measurements on 90Sr and 14C within experimental uncertainties compared to destructive measurements	The system (TRL6) needs to be improved to be able to quantify several radionuclides present altogether and to be industrialised.
Alpha imaging	TRL 8-9.	Major drawback is operating conditions as it needs complete darkness.
Wipe tests followed by standard LSC or alpha spectrometers.	Effective for labile radioactivity for many standard beta and alpha measurements.	Wipe test technique strongly depends on the type of radioactivity and is not effective for non labile radioactivity.

4.9.2 Experiences/Case studies

4.9.2.1 Autoradiography at LHA Laboratory- CEA Saclay

After the development of the technique, the high potential of the Autoradiography by screens was demonstrated to provide spots of tritium traces on the surface of a CEA laboratory (LHA in Saclay, laboratories dedicated to high activities in different domains). 20% of the whole surface was measured and after calculations by kriging process, as described above Section 4.6.2[374], the complete mapping was obtained for the spots of tritium activity (see Figure 4.9-3 below).

[374] www.geovariances.com

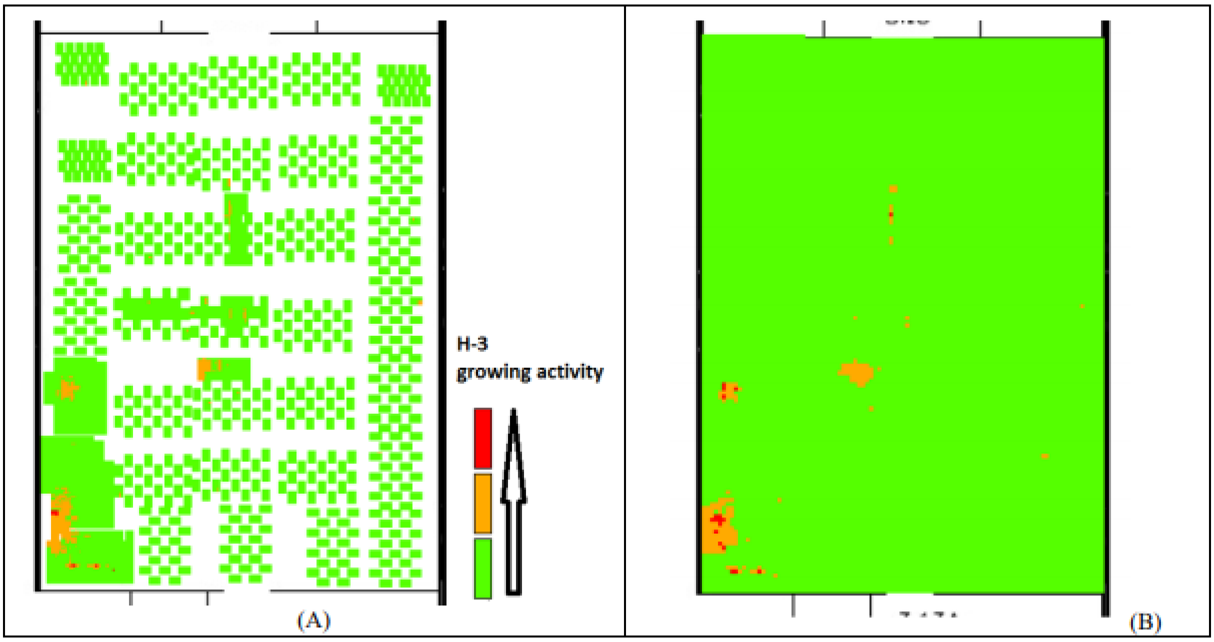


Figure 4.9-3 Autoradiography result: Mapping of tritium spots on the surface of a laboratory (250 m2) in dismantlement (real measurement (left)), measurement after kriging process (right).

A MAUD camera was recently tested to measure very low concentrations of uranium (less than 1 Bq / cm2) on the concrete surface in a facility undergoing final dismantling. The results are very promising.

4.9.2.2 Beta Spectrometry at CEA Fontenay aux Roses and at gas cooled reactor G1 in Marcoule

Before applying Spectrometry Beta in situ to various types of soil encountered in decommissioning sites, measurements were performed on samples of Fontainebleau sand from the CEA site at Fontenay-aux-Roses. Furthermore, to assess the possibility of analyzing multiple samples using an autosampler, measurements were also taken on sand samples conditioned in standardized 500 cm^3 polyethylene containers, which are mainly used to analyze liquid samples by gamma spectrometry.

The resulting 90Sr activities were within experimental uncertainties of those obtained using destructive measurements. These results are promising for wider applications on nuclear decommissioning sites.

The device was also used by CEA in 2016 to measure ^{14}C in some historical samples from gas cooled reactor G1 at Marcoule.

In each case results were validated within experimental uncertainties of those obtained using destructive measurements.

These results are promising for wider applications on nuclear decommissioning sites. The system (TRL6) needs to be improved to be able to quantify several radionuclides present all together and to be industrialised.

4.9.2.3 Alpha imaging CEA Experience

The device was used at CEA since 2011and worked quite well:

- From 2011 to 2013 in Atalante facility (research on processes for back end)
- In 2011 at Pu MOX facility in Cadarache
- From 2015 to 2016 in facility LEFCA (CEA Cadarache)
- In 2016 it was tested in Sellafield site (UK)

TRL level is 8-9. A major drawback is operation conditions as it needs complete darkness.

5. Site preparatory activities

Preparatory activities may be organised in various ways depending on considered decommissioning strategies and physical and radiological status of the nuclear facility after its routine operation is over[375].

It is very desirable to take timely action to place a nuclear facility in a safe, stable and known condition as soon as possible after final shutdown. It is important that stabilisation and other activities for facilities systems and materials be planned and initiated prior to the end of operations. Carrying out these activities during the final stages of a facility's operational phase and during the transition period will be beneficial in that the operational capabilities of the facility and the knowledge of personnel will be utilised before they are lost. Actions taken at this time will pave the way to efficient and cost-effective decommissioning by eliminating, reducing or mitigating hazards, minimising uncertainty and maintaining steady progress.

Typically, these activities include defuelling of reactors, retirement of equipment and systems, radiological and waste characterisation, operational waste treatment and removal of minor components. Generally, removal or dismantling of major components and, where applicable, safe enclosure (SE) are excluded. However, activities carried out during the transition period will depend upon the type of facility and the regulatory regime. The objective of the transition period is to plan and implement these activities in a timely manner. A cultural change is also needed to reflect different management and working practices. It is essential that planning for the transition and decommissioning begin during operation and that activities be implemented as soon as possible after permanent shutdown to ensure a controlled transition and the best use of resources [376].

International initiatives

IAEA Initiatives

- The IAEA have published numerous safety and technical reports providing guidance, recommendations, experiences, good practices and lessons learned, fully or to some extent covering the preparatory phase for decommissioning. Many training courses, workshops, seminars etc. have been organised to support sharing of good practices among specialists and organisations involved. The reference document [377] provides an overview of relevant activities and perspectives of the IAEA in this area and draws some general conclusions and identifies lessons learned on the basis of the initiatives implemented so far. Other relevant IAEA documents are [378], [379] and [380].

[375] "PREDEC 2016: IAEA Perspectives on Preparation for Decommissioning". 2016 February 16-18, Lyon, France.
[376] "Transition from Operation to Decommissioning of Nuclear Installations". IAEA. 2004.
[377] "PREDEC 2016: IAEA Perspectives on Preparation for Decommissioning". 2016 February 16-18, Lyon, France.
[378] "Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities". IAEA SAFETY STANDARDS SERIES No. SSG-47. 2018
[379] "Planning, Management and Organisational Aspects of the Decommissioning of Nuclear Facilities". IAEA TECDOC-1702. 2013
[380] "Transition from Operation to Decommissioning of Nuclear Installations". IAEA Technical Reports Series No. 420. 2004

- Many group events involving participants from different Member States, such as training courses, workshops and seminars, were organised by the IAEA to support sharing of good practices on preparation for decommissioning among specialists and organisations involved. Representatives of operators, technical support organisations and regulators are typical attendees of the IAEA events. Such events are mainly organised as part of the Technical Cooperation (TC) Programme of the IAEA, often with support from the International Decommissioning Network (IDN), and are mainly hosted by the Member States' organisations responsible for decommissioning. They involve lectures by international experts or by the host organisation, discussion sessions, facility visits, practical demonstrations and exercises, and observations of ongoing site activities. Nuclear power plants and research reactors have been the main target facilities for a longer period of time, but recently more events have started to be organised; addressing preparations and implementation of decommissioning of other types of nuclear fuel cycle and radioactive waste management facilities. Focused support to address specific needs of particular Member States is provided within the national TC projects or through national projects financed by extra-budgetary contributions to the IAEA. Assistance in planning for decommissioning, including safety assessment and estimation of decommissioning costs, or support to characterisation activities are typical examples of aspects covered within such kind of projects. Some topics related to preparation for decommissioning are of common interest for many Member States, especially in the area of research reactor decommissioning. In such cases the IAEA organises topical international projects to provide longer-term platforms for cooperation, training, exchange of knowledge and experience, and for promotion of good practices. Examples of such projects are the Research Reactor Decommissioning Demonstration Project [381] and the International Project on Data Analysis and Collection for Costing of Research Reactor Decommissioning (DACCORD) [382].

NEA Initiatives

- With a growing number of nuclear facilities reaching the decommissioning stage the Working Party on Decommissioning and Dismantling (WPDD), within the NEA, formed the Task Group on Preparing for Decommissioning during Operation and after Final Shutdown (TGPFD), which involves regulators, nuclear operators and independent experts who review strategic aspects to optimise preparations for decommissioning from the last years of operation onwards.

  The reference document [383] summarises work carried out by TGPFD between March 2015 and December 2017.

[381] The R2D2 Project, 9 December 2014, http://www-ns.iaea.org/projects/r2d2project/.

[382] “Data Analysis and Collection for Costing of Research Reactor Decommissioning. Report of the DACCORD Collaborative Project”. IAEA TECDOC-1832. 2017.

[383] “Considerations on safety in the transition period from operation to decommissioning of nuclear facilities”. Transactions of the Korean Nuclear Society Spring Meeting. 2019.

5.1 Adaptation of auxiliary systems for decommissioning (ventilation, electrical, monitoring, etc.)

The transition from operational stage of nuclear facility to its decommissioning phase is a critical life cycle of each facility. To meet new highlighted requirements and objectives not only a number of technical but also organisational changes are required, as well as some pre-activities must be initiated that facilitate the transition and preparation for dismantling of the facility.

During the transition period, activities are planned and implemented that facilitate simplification of decommissioning activities, lead to reduced surveillance and maintenance requirements and, of course, pare down operating costs. This can be achieved by identifying those facility's auxiliary systems that after final shutdown will become redundant.

Systems that are not required anymore to maintain the safety of the facility can be de-energised or shut down after cessation of operation, i.e. corresponding systems will be switched off and remain unpressurised and cold.

This also includes the drainage of circuits which may reduce the fire load within the facility or reduce the hazards from spills and internal flooding.[384]

Systems that are required after shutdown but are expensive to operate and maintain should be further considered, e.g. the capacity of ventilation systems needed to control contamination at decommissioning period can be reduced considerably.

The main general requirements and considerations regarding status of facility systems during transition stage to ensure or restore the required level of safety are presented in IAEA Specific Safety Guide SSG-47 [385]. Hence, prior to shutting down a facility or at the latest during the transition from operation to decommissioning, the licensee should initiate studies to support development of the **final decommissioning plan**. These studies should identify the systems, equipment and infrastructure from the operational stage that will need to be maintained for use during decommissioning, and should specify, and if necessary research, any new systems, equipment and infrastructure that will need to be installed to support decommissioning[386].

The selection of the decommissioning strategy should be based on an analysis of various options, which may lead to selecting a combined strategy that consists of some degree of immediate dismantling actions,

[384] NEA No. 7374, Preparing for Decommissioning During Operation and After Final Shutdown, OECD 2018 Working Party on Decommissioning and Dismantling. (WPDD).

[385] IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and other Nuclear Fuel Cycle Facilities, Safety Standards Series No. SSG-47, Vienna (2018).

[386] IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and other Nuclear Fuel Cycle Facilities, Safety Standards Series No. SSG-47, Vienna (2018).

followed by a preservation of the remaining parts of the facility, which are then dismantled after a period of safe enclosure [387].

To protect workers, the public and the environment from exposure due to the spread of radioactive substances, active safety systems such as ventilation systems and fire protection systems might need to be retained for some period during decommissioning or might need to be adapted to the risks present during decommissioning actions [388].

Decommissioning actions might involve the deliberate removal of SSCs that fulfilled specific safety functions during operation of the facility (e.g. confinement, shielding, ventilation and cooling). Such actions should be recorded and aligned with the ongoing decommissioning phases, work packages and tasks identified in the final decommissioning plan[389].

Modifications of the existing infrastructure of the facility that are necessary to facilitate immediate dismantling or, in some cases, to prepare the facility for a period of safe enclosure might involve a set of modification or substitution of SSCs that are important for ensuring safety during decommissioning, such as ventilation systems and containment systems [390].

If these safety functions are still required, they should be provided by suitable alternative means or SSCs (e.g. tents, temporary systems or structures, fire systems, electrical systems and/or administrative procedures) for as long as is required on the basis of the safety assessment. The fulfilling of safety functions of these alternative means should be demonstrated. Procedures for changing the means by which safety functions are provided during decommissioning should be justified and demonstrated in advance of their implementation[391].

Decisions on which facility systems must remain functional should be made during the planning of the transition period and [392] are based on:

- An evaluation to ensure that safety requirements will continue to be met,
- Support of human entry or occupancy for surveillance and maintenance,
- Possible use during future phases of decommissioning,
- Restrictions posed by the current operating licence.

---

[387] IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and other Nuclear Fuel Cycle Facilities, Safety Standards Series No. SSG-47, Vienna (2018).
[388] IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and other Nuclear Fuel Cycle Facilities, Safety Standards Series No. SSG-47, Vienna (2018).
[389] IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and other Nuclear Fuel Cycle Facilities, Safety Standards Series No. SSG-47, Vienna (2018).
[390] IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and other Nuclear Fuel Cycle Facilities, Safety Standards Series No. SSG-47, Vienna (2018).
[391] IAEA, Decommissioning of Nuclear Power Plants, Research Reactors and other Nuclear Fuel Cycle Facilities, Safety Standards Series No. SSG-47, Vienna (2018).
[392] IAEA, Transition from Operation to Decommissioning of Nuclear Installations, Technical Reports Series No. TRS-420, Vienna (2004).

During planning of the transition period, decisions regarding systems and major equipment within a facility may need to consider the following options[393]:

- Operable as is: Systems that must remain operable and do not require modification. For example, lighting where surveillance and maintenance is to be done.
- Modified: Some systems will need to remain operable but modifications are required. For example, building ventilation is needed to maintain control of remaining contaminated areas but its design capacity is excessive, or redundancy of systems and components is no longer required because the consequences of temporary failure are acceptable until repairs can be made.
- Preserved for future use: A limited number of systems and equipment may be preserved for the future. For example, installed manipulators and cranes can be of use during dismantling, or radioactive waste treatment systems may be valuable for processing decontamination solutions. Decisions of this type will depend on the length of time until such use is expected as some ageing will occur even in systems that are not in operation.
- New: In some cases system functions will be needed, but use of the installed system may not be feasible because it may be overly complex, be over capacity, have high levels of contamination, or entail difficulty of access for operation or maintenance. In such cases, total replacement with new systems and/or equipment is the prudent course of action. Another option includes replacement of instrumentation because of obsolescence or the need for monitoring from a different or a remote location, and the installation of limited lighting for infrequent inspections where isolation of other unused circuits is not practical. A new electrical distribution system could be installed as well, to repower that equipment necessary to support the decommissioning work.
- Retired: In many cases, a large number of systems will no longer be needed. In such situations, they are generally left in place and suitably isolated using standard safety practices, especially where there is internal radiological or hazardous chemical contamination or, in the case of electrical systems, the potential for short circuits or high voltage shocks. In some cases, complete removal of a system may be chosen, for example when the assets can be used at other facilities, or systems such as installed ventilation may be isolated where it is beneficial to use temporary or portable equipment when needed.

The end point specifications and requirements for system surveillance and maintenance during the transition period and safe enclosure shall be determined after a decision on the system is made. In any case the licensee shall implement these modifications of the auxiliary systems according the final decommissioning plan in compliance with national regulations.

[393] IAEA, Transition from Operation to Decommissioning of Nuclear Installations, Technical Reports Series No. TRS-420, Vienna (2004).

5.1.1 Experiences/Case Studies

5.1.1.1 Lithuanian experience adopted for Ignalina NPP decommissioning

The immediate dismantling option was offered by IAEA experts and adopted by Lithuanian Government as an Ignalina NPP decommissioning strategy which determined the dismantling of the equipment practically immediately after the closure of reactor's operation. The end of the *brown-field* stage is expected to be reached by 2038, meanwhile the dismantling of the 1st Unit reactor will start in 2027, and the 2nd Unit reactor in 2029. Even before the dismantling of Ignalina NPP reactors the decontamination and the dismantling of auxiliary technological equipment placed in auxiliary buildings or areas (turbine hall, boiler house, ECCS and helium facilities, etc.) were/are implemented. Based on this collected experience the main findings are reported below.

Ventilation systems

For radiological containment, removal of toxic gases and excess heat the supply and exhaust ventilation systems are installed in Ignalina NPP buildings were employed during D&D operations. Some modifications to existing ventilation systems and installation of new temporary ductworks, as well as usage of mobile filtration unit(s) were required in order to effectively discharge air flow from decontamination and/or size reduction workshop(s) covered by a containment tent (Speed-frame type design) in order to confine radioactivity inside during D&D operations.

The external make up supply air was supplied, in most cases, directly from the main plant **external ventilation system** with existing ductwork and diffusers placed inside the rooms where D&D activities were performed. In some cases the rooms were only provided with extraction inlets to guarantee the correct air direction. The supply air comes from the neighbouring *clean* rooms/corridors through build openings (i.e. doorways, excess pressure valve openings, in-bleeds) via the *clean* premises where supply air was also provided to obtain a positive pressure within these *clean* areas. Whenever the existing extract ventilation system was demonstrated to be insufficient, Mobile Filtration Unit(s) (MFU) for radiological containment and general toxic gas dilution was installed in the nearest practical location in the adjoining corridor to assure the correct air extraction. The galvanised spiral wound ductwork from the MFU outlet to an existing duct of extract ventilation system was used for connection. Whenever D&D operations were generating significant quantities of potentially contaminated smoke and fumes, a dedicated and portable Fume Extract Unit(s) (FEU) was employed for local (at-source) toxic gas extraction with a mobile extract canopy which was installed at the cutting location within the room being deplanted. For contaminated rooms the outlets of MFU and FEU were connected to existing extract ventilation system, and this option was proved as safe and effective method.

The existing **supply air ventilation system** was used to provide heating to the building(s)/room(s) in order to maintain a satisfactory working conditions for D&D operators, namely it was retained a minimum internal room temperature of 16^{o}C during the cold period of the year.

The combination of existing plant ventilation systems and additional mobile MFUs/FEUs ensured that sufficient fresh air was provided in each room for its occupants during D&D implementation. The required cascaded air movement, negative pressure and velocities through engineered and non-engineered openings were maintained, and it constituted the necessary containment by making air pass from areas/rooms of lower to areas/rooms of higher contamination. All active ventilation systems operated in a mode that lead to no re-circulation of air between the supply and extraction ventilation systems.

Electrical systems

For execution of scheduled D&D activities the essential electrical services to supply portable tooling, portable ventilation units, task lighting and etc. were provided prior and used during D&D activities. The plant buildings has already installed a network of power supply outlets. It was decided to utilise the existing **power distribution network** and to supplement and upgrade it where necessary for providing maximum flexibility for the arrangement of D&D equipment, and to allow simple installation of new equipment. Due to the temporary nature of this installation (until site/building demolition) it was agreed that any new fixed installation work should be minimised and limited to provision of a small number of higher power/voltage supply sockets from which the required lower voltage power sockets and additional lighting were derived. Cables were retained in-situ whether or not they were redundant due to difficulties in identifying those which are redundant. As it was frequently impossible to identify whether each cable was redundant or not, localised protection measures were implemented.

As the most economical option free-standing portable electrical distribution units fitted with multiple three and single phase socket outlets for the portable tools and cutting equipment at the workface, and containing all necessary switching and protection equipment were employed. This approach allowed the workplace supplies to be easily and quickly redeployed and reused to accommodate changing site requirements as the D&D activities progresses. They were simply removed from the buildings/rooms on completion of the work and reused elsewhere.

Therefore the existing plant building services electrical installation was retained and maintained in an operational state throughout D&D operations, and once all other systems were removed the general power and lighting systems were then completely isolated at the points where their main incoming supply cables enter the buildings resulting in a completely electrically isolated building/area.

The existing illumination from the general lighting already installed in buildings was considered to be adequate for at least general building access. The **normal lighting system** for plant buildings and rooms were D&D activities took place were continued to be maintained throughout the deplanting life cycle. In D&D tented working areas additional lighting was provided by the deployment of free-standing task lighting.

Plant buildings already are equipped with an **emergency lighting system** to provide lighting within the buildings/areas in case of emergency which is powered from design supplies. This installation was routinely maintained throughout the D&D operations. Additional emergency lighting was provided locally inside any tented areas by use of portable standby lighting system with battery integral back-up power supply to allow for safe egress (access) in the event that the main normal supply is lost.

Monitoring and alarm systems

Entry to any areas with the potential for high radiation doses during D&D operations was under site approved systems of work (**radiological access control system**) according to Ignalina NPP procedures. The suitability of all radiological monitoring equipment was agreed by plant health physics department prior it was purchased and/or used and was in-line with existing Ignalina NPP procedures.

Portable real-time **beta/gamma activity in-air monitoring** units were located at each working area, workshop and at main access routes in order to confirm that the containment arrangements were intact, and to alert the operators and supervisors to unexpectedly high levels of airborne activity, and to initiate evacuation of the area.

MFUs fitted with HEPA filters were employed to extract air from the D&D work areas. MFUs were fitted with a **local battery backed alarm** to warn of operational failure. Each MFU was fitted with an additional wired alarm repeater module which was detached and located inside the D&D work area.

No modifications were required to existing plant alarm systems, except for those required to the existing fire alarm systems. In buildings being deplanted an **automatic fire alarm system** was/is installed, which according design detects fire ignition at an early stage and trigger fire suppression via the automatic fire fighting system. Obviously, due to the usage of flame cutting during D&D operations (in most cases) automatic operation of the **fire fighting system** was inhibited. It was proposed to retain this current system in its current operational state, but where hot cutting activities were executed the fire detectors in these areas were selectively disabled temporarily by enclosing them in a temporary enclosure for the duration of the hot works. During hot work operations additional arrangements for fire detection were organised involving a nominated person(s) keeping a manual fire watch and raising the alarm if a fire break out. For this reason a fire call points were installed on the emergency exit routes from each floor of the building which represent a break glass devices that allow the operator to inform the control room of the presence of a fire in working area. The automatic fire detection system was/will be maintained until the very final end of decommissioning activities in that building. It was/will be decommissioned and deplanted in tandem with the electrical supplies and services just prior to building demolition.

Other services systems

A number of service pipelines, namely compressed air and water supply/drainage lines, were required to be retained until the end of D&D operations as they were used during preparation activities and during D&D operations. Individual protection measures for particular areas against fire and/or impact damage were incorporated in the relevant working procedures. In general, the main protection provided for these service lines consists of clear identification of the lines to avoid erroneous dismantling.

The **compressed air supply lines** in buildings/areas which undergo deplanting were in need of upgrading in order to provide a single outlet equipped with a pressure gauge and flow meter, and to provide flow of specific parameters to D&D workshop in specific location (for example, strict requirements exist for the operation of Vacuum Blast system and shall be in line with equipment specification).

The **water supply pipelines** were not removed at initial deplanting phases as they were required for D&D activities, instead, they were dismantled in the final phase of deplanting just prior to D&D equipment removal works. During D&D implementation, operations requiring water supply (or producing waste water) were scheduled only for the removal of concrete structures (concrete bursting operations) in order to aid man access or re-open construction openings. No further water demand was used during the D&D activities.

The waste water producing operations (making construction openings, enlarging the doorways, etc.) did not generate significant amounts of waste water, because water required and used for technological cooling quickly evaporated due to the heat generated by the bit during the core drilling operation. In other cases such effluent was of limited volume and was directed to the existing **water drainage system**. Due to the low contamination levels in buildings deplanted, the radiological content of the waste water effluent was considered to be negligible.

Buildings **telecommunications systems** were retained and maintained until the final phase decommissioning activities as it was considered adequate for systematic decommissioning and deplanting. The communications systems were/will be decommissioned and deplanted in tandem with the electrical supplies and services just prior to building(s) demolition. Up to the point of decommissioning the telecommunications system was maintained in a fully operational state.

5.1.1.2 Preparatory activities for Jose Cabrera NPP decommissioning project

See 5.2.2.2

5.2 Preparation of infrastructures and buildings for decommissioning (storages, capabilities for material sorting and treatment...)

The approach to decommissioning in terms of decommissioning strategy and plant/facility end state, strongly hinges on internal and external factors which are key enablers or constraints for a decommissioning project and may include:

- the funding situation (available funding for decommissioning);
- the policy, socioeconomic factors and regulatory framework (national, regional and local regulations);
- the facility design and status;
- waste routes and disposal maturity and availability;
- good practices and experience;
- waste acceptance criteria;
- stakeholder perceptions and influence.

The strategic decisions that derive and that must be taken in relation to these key enablers and constraints include:

- the nuclear material and/or spent fuel management strategy;
- the materials and waste management strategy;
- the remediation/site clean-up strategy;
- the dismantling strategy;
- the facility modification strategy;
- the technological strategy;
- corporate strategies;
- the authorisation process.

Some of these strategic decisions significantly influence the activities related to the preparation of infrastructures and buildings during the last phases of operation, after final shutdown and during the entire decommissioning phase [394].

It is therefore clear that there are currently no standard methodologies and best practices concerning the preparatory activities and the preparation of infrastructure and buildings applicable in all situations and conditions and that an analysis must be made by evaluating case by case taking into account all the boundary conditions listed above.

For all these reasons, only general concepts and common principles are given in this section, providing some examples of international case studies.

[394] "Preparing for Decommissioning During Operation and After Final Shutdown". NEA-OECD. 2018.

The radioactive waste infrastructure required for decommissioning is very different from that required for operations.

Planned and systematic preparation for decommissioning is very important for further effective implementation of dismantling activities and to better face the challenge of managing the materials and waste that will be produced during the decommissioning, possibly in addition to the operational waste already present on the site.

In order to optimise and increase the reliability of the decommissioning strategy and therefore of the planned preliminary activities, great importance should be given to the recording and constant updating of the physical, chemical and radiological inventory of the plant/facility. The objective of an accurate materials inventory is to sort material into categories in order to identify the most suitable routes and the best decommissioning methodologies. Categorising components, systems and structures is also vital to identify opportunities for radioactive waste reduction (such as re-use/recycling or clearance if applicable), to minimise decommissioning costs, and also to meet waste acceptance criteria for disposal [395], [396].

As just mentioned, the inventories may have a large impact on the waste routes.

The selection of the waste routes in a decommissioning project depends on other several factors, in particular:

- The decommissioning strategy: deferred or immediate dismantling (if the strategy is immediate decommissioning in order to delicense and release the site in a short time, the waste should be transported offsite as soon as possible – either for direct disposal, or to an external treatment or interim storage facility; for deferred decommissioning, it can be useful to have an onsite waste treatment facility. This might include intermediate storage to allow for decay and provide flexibility in the rate of waste flow through the facility);
- The total waste management cost including disposal. Indirect costs such as the impact on the decommissioning schedule and investment in infrastructure and organisation should be included (e.g. a low VLLW disposal cost may favour disposal without treatment as it could be significantly cheaper to consider all VLLW and all potentially contaminated waste as VLLW; on the other hand, if the regulatory system prescribes waste minimisation or if the disposal costs are high, this would favour investments in advanced waste treatment centres and/or agreements with external treatment facilities. If there are large differences in cost between VLLW and LLW disposal, then treatment to reclassify LLW to VLLW prior to disposal is favoured);
- The potential to carry out release of material from the practical, regulatory and public perspectives;
- The national programme for management and disposal of radioactive waste, including the availability of final repositories and related waste acceptance criteria (WAC);

[395] "Preparing for Decommissioning During Operation and After Final Shutdown". NEA-OECD. 2018.
[396] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.

- What is needed to create/adapt on-site waste management infrastructure for handling, interim storage and transportation including reviewing the capability of existing on-site treatment facilities and capabilities;
- The availability of dedicated external waste treatment facilities.

In terms of sustainability, the 'waste hierarchy' should be generally applied to routing materials from nuclear facilities. According to the waste hierarchy, the preferred end state is re-use or recycling of the waste as material or, more preferably, the avoidance of waste generation in order to preserve natural resources as well as reduce the environmental impact.

In addition, treatments (such as decontamination, thermal treatment, segmentation, melting, supercompaction, etc.) that can reduce the volumes requiring disposal as radioactive waste should be considered [397], [398].

In order to perform some of these treatments the following types of areas may need to be created in support of selected strategies of dismantling and materials and waste management: areas for system and component cutting, areas for decontamination and characterisation and areas for conditioning/processing and packaging of materials and radioactive waste arising from dismantling as well as storage areas for materials and radioactive waste to facilitate the management of systems and components treated or to be treated.

The decision on the level of on-site waste treatment is closely related to the design and status of the facility to be decommissioned, waste acceptance criteria for storage and disposal, clearance criteria (if applicable), transportation limitations and whether there are service providers in radioactive waste treatment.

Taking into consideration the existing infrastructures at the nuclear facility and logistical considerations for tackling the materials and waste streams and basing on good practice, the strategy may include:

- To use an auxiliary building for accommodating centralised systems and auxiliary facilities and infrastructures: cutting workshops, decontamination workshops, waste conditioning areas, zone for waste storage, etc. An existing building can be adapted to this function.
- To install confined cutting areas into the buildings to be dismantled.
- To distribute storage areas for dismantled materials and waste along the site.

Further considerations in the development of auxiliary installations strategy include:

- Specific requirements for the handling of highly activated wastes;
- Buffer storage areas for dismantled materials and waste with sufficient capacities to accommodate large volumes.

[397] "Preparing for Decommissioning During Operation and After Final Shutdown". NEA-OECD. 2018
[398] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.

Modifications and adaptations in this context should be implemented during the transition as part of the preparatory activities provided, they are allowed under the operating licence.

However, it is noted that facility modifications, in particular in case of replacement by a new system or refurbishment of an existing building, may cause additional amounts of materials and waste when they have to be dismantled, and these amounts need to be included in the materials and waste management strategy[399].

It might be of advantage to share processing and storage infrastructure across a number of nuclear facilities or to use external service providers rather than constructing one or more facilities on each of the sites (for most decommissioning projects, there will normally be a combination of onsite and offsite waste management).

The advantages of a waste management fleet approach are potential cost savings (resulting from economies of scale) and reduced risk (due to repeating the same activities). It is also possible for a number of utilities to cooperate with each other to adopt a fleet-wide approach for their combined plants.

During the decision-making process, existing buildings and facilities should be taken into account. An important consideration is the level of investment in equipment and organisation required for proper and efficient handling and treatment of the material. On the other hand, the necessity to transport the waste material from the decommissioning site to an external site and, if required, returning the conditioned waste package back to the site, can result in significant costs. Long distance shipments of waste for treatment, especially in complicated regulatory environments, drives investment towards local waste treatment solutions [400], [401].

For this evaluation it is important to analyse:

- value of early removal of waste in terms on decommissioning progress;
- overall life cycle costs including investments in buildings, equipment, organisation, etc.;
- decommissioning liabilities for new facilities;
- how an in-house waste treatment or storage facility will affect the decommissioning project;
- transport safety constraints including stakeholders' sensibility on nuclear transport;
- national waste management strategy;
- national regulatory framework.

An investment in a separate waste treatment or storage facility will separate dismantling from the waste management which may be a significant advantage for a decommissioning project [402], [403].

---

399 "Preparing for Decommissioning During Operation and After Final Shutdown". NEA-OECD. 2018.
400 "Preparing for Decommissioning During Operation and After Final Shutdown". NEA-OECD. 2018.
401 "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.
402 "Preparing for Decommissioning During Operation and After Final Shutdown". NEA-OECD. 2018.
403 "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019

5.2.1 Description of the waste route options

Typical waste route options are closely related to the preparation of infrastructures and buildings for decommissioning. Some examples of these routing options are discussed further below.

Disposal without treatment [404]

Disposal of waste after dismantling only requires segmentation to fit into waste containers, and conditioning of the waste packages. In most cases this is one of the simplest ways to manage the waste, as decontamination or treatment for volume reduction are not carried out.

The only objective is to qualify the waste for disposal, i.e. to meet the specific waste acceptance criteria. Such criteria differ significantly from country to country, but also between waste classes and repositories within individual countries.

This waste route option is attractive in countries with low disposal costs and no regulatory requirements on waste minimisation, clearance and recycling.

Local waste treatment centre within the facility [405]

A low investment alternative is to establish a local waste treatment centre inside the facility to be decommissioned (for example in the turbine hall/building of a Boiling Water Reactor (BWR)). The main challenge is to make the waste treatment centre available in accordance with the decommissioning schedule.

The design requirements for this type of waste treatment centre depend on the waste management strategy (including clearance criteria and waste acceptance criteria for storage and disposal), transportation limitations, access to external treatment facilities, available disposal space, the possibility to dispose of large components and associated costs. Typical installations are cold cutting equipment, mechanical and/or chemical decontamination units, equipment for clearance measurements and radiological analyses, as well as arrangements for conditioning of disposal packages.

This waste route option is attractive for organisations aiming to carry out the decommissioning activities themselves.

Local waste treatment centre outside facility but onsite [406]

A fairly costly but attractive alternative is to build a new local waste treatment centre outside the facility being decommissioned but onsite. One important advantage is that it has a low impact on the dismantling process. It is important to remember that such a facility should be licensed, built, commissioned, and upon completion of service, decommissioned.

---

[404] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.
[405] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.
[406] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.

These waste treatment facilities can be permanent or modular structures. The advantage with a modular facility is that it may be possible to be moved to another site for re-use when the project is over.

External treatment and conditioning [407]

In most countries, it is possible to transport radioactive waste to a dedicated external waste treatment facility. Such facilities can either be part of a fleet approach, part of a national programme, or owned by external commercial service providers. By transporting the waste for treatment at a dedicated facility at another location, some of the decommissioning work is transferred away from the site, allowing the onsite staff to focus on the main tasks. In many cases, this option will lead to a significant reduction of volume for disposal. The overall direct cost can be higher than for local treatment as it will include transport costs as well as the fees of the service provider. However, this should be balanced against the reduced risk, removal of the need for local treatment and storage (including training of staff), and reduced waste volume for disposal associated with this this option.

Mobile waste treatment facilities [408]

For certain waste streams, where the tasks will be undertaken in a time-limited manner (e.g. removal of reactor internals), it may be necessary to bring mobile waste treatment facilities to the decommissioning site instead of moving the waste to an external waste treatment facility.

This applies in particular to waste which is especially problematic from a technical or regulatory perspective, or very costly to transport (e.g. ILW resins and waste streams which are large in volume). The focus should be on waste streams for which the required equipment is easy to transport and install – for example, contaminated and potentially contaminated concrete, which has to be crushed and measured for clearance.

Mobile or temporary facilities can be developed and provided by and within a nuclear fleet or national programme. They can also be provided by an external service provider.

Interim Storage facilities[409]

While decommissioning waste is a lesser but still important part of the cost of decommissioning a nuclear plant, the failure to provide waste management routes and facilities aligned to the decommissioning programme could lengthen the schedule, hence increasing the project management costs, the largest cost of decommissioning. Furthermore, for the foreseeable future, the lack of radioactive waste disposal facilities will continue to increase the cost of radioactive waste management due to the need to provide additional interim storage solutions. It should be noted that interim storage often lasts longer than initially expected.

For these reasons the interim storage needs must be carefully assessed and the on-site and/or off-site interim storage facilities must be identified (e.g. on-site existing buildings can also be adapted to this function).

---

[407] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.
[408] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.
[409] "Methodology to Manage Material and Waste from Nuclear Decommissioning". WORLD NUCLEAR ASSOCIATION. 2019.

This approach allows for the separation of the process of generating waste via dismantling, from its transfer to a disposal site, and hence avoids bottlenecking and delays during dismantling operations – due to materials and waste management logistics – and provides more time for establishing waste routes for problematic waste streams.

Interim storage can provide different functions during a decommissioning project, for example:

- Storage of dismantled materials and/or waste prior to further treatment or handling steps.
- Storage of materials and/or waste for declassification or free release ("decay storage").
- Storage of containers with radioactive waste prior to final packaging of the waste form for disposal.
- Storage of the final waste form for disposal until the final disposal site is ready to accept waste.

Table 5.2-1 Summary of the waste route options

<table>
<tr><th>Waste Route Options</th><th>What is working</th><th>What is missing</th><th>Assessment and Possibility for improvement</th></tr>
<tr><td>Disposal without treatment</td><td>• Simple, no waste treatment required.
• Lower need for interim storage in case of disposal availability.</td><td>• Higher volume of waste for disposal.
• Higher volume of waste for interim storage in case of lack of disposal facilities.
• Not always compliant with the regulatory framework and with the main principles of waste management.</td><td rowspan="5">• Availability and international harmonisation of Waste Acceptance Criteria (WAC).
• Harmonisation of the regulatory framework, waste management strategies and clearance criteria.
• Sharing experiences, especially in terms of times, costs and lessons learned in order to allow strategic choices based on solid and reliable data.</td></tr>
<tr><td>Local waste treatment / conditioning centre using existing building within the facility</td><td>• Reduction of waste for disposal.
• Lower consumption of materials and resources and minimisation of potential waste to manage in the future.
• No off-site transport.
• Usually already licensed area.
• No need of outsourcing services.</td><td>• Not operational from the start of the dismantling project and not available to the end of it (greater need of on-site interim storage capacities).
• Variety and efficiency of treatments potentially limited by the available spaces and logistics.
• In some cases, renovation or adaptation of existing structures/buildings can be expensive both in terms of cost and time.</td></tr>
<tr><td>New waste treatment / conditioning centre outside facility but onsite or within the facility</td><td>• Reduction of waste for disposal.
• Construction of the facility has a lower impact on the decommissioning schedule.
• More flexibility and variety of treatments.
• No off-site transport.</td><td>• Investment in new infrastructure.
• Licensing of new facility.
• New systems, structures and components subject to decommissioning in the future (potential additional waste and costs).</td></tr>
<tr><td>External treatment / conditioning facility operated by a service provider</td><td>• Reduction of waste for disposal.
• Some of the decommissioning work is transferred away from the site, allowing the onsite staff to focus on the main tasks.
• Already licensed facility/area.
• Use of existing infrastructure.
• No investment required – service provided by external company and included in treatment fee.
• Possibility of carrying out treatments abroad that cannot be carried out in the country where the waste is produced.</td><td>• External transport of waste required.
• Important to have good coordination between stakeholders (in case of commercial facility, capacity utilisation could be an issue for the service provider).
• Little flexibility with respect to the types of treatments specifically performed by the provider.
• Sometimes outsourced service costs are considerable.</td></tr>
<tr><td>Mobile waste treatment / conditioning facilities</td><td>• Reduction of waste for disposal.
• Shared equipment costs.
• No transport of untreated waste – if facility taken to waste location
• Lower consumption of materials and resources and minimisation of potential waste to manage in the future.</td><td>• Availability (booking) of the facility in line with the decommissioning schedule.
• The potential need for decontamination before transporting from site to site for other treatment campaigns can be an issue.</td></tr>
</table>

5.2.2 Experiences/Case Studies

5.2.2.1 Italian Case Studies [410, 411]

Reuse of Garigliano and Caorso (Nuclear Power Plant) NPPs Turbine Building as Material and Waste Treatment Facilities

In the Turbine building of Garigliano NPP, after the removal of components of the steam cycle, it is foreseen that there will be installation of some treatment stations to be used for the dismantling of components coming from the reactor building. This solution has the advantage of avoiding the realisation of new buildings with a minimum volume of 30.000-40.000 m3. In addition, thanks to the direct connection between the reactor and turbine buildings, this solution will also facilitate component handling operations and will also avoid risks connected to possible spread of contamination.

Another successful example of existing building reuse is the Turbine building of Caorso NPP. In this building, at turbine level, a material management facility ("SGM – Stazione Gestione Materiali") has been installed to perform size reduction and decontamination treatments and processes aimed to material release or waste volume reduction. The levels below the SGM have been completely emptied in order to arrange buffer areas to store radioactive waste. At the lowest level a Waste Treatment Station composed of supercompaction and cementation unit is nearing completion.

Also in this case, this solution has the advantage of avoiding the realisation of new buildings with a minimum volume of 50000-60000 m^3.

For both plants, this strategic choice will lead to a reduction in the radioactive waste production of some 100 cubic meters.

Reuse of Garigliano NPP "Ex-Diesel" Building, and Casaccia FCF "OPEC-2" Building as temporary Storage Facilities

In some of the decommissioning activities carried out by Sogin, the adaptation of existing buildings has resulted in the creation of temporary storage facilities for radioactive waste.

Ex-Diesel building of Garigliano NPP was built in the 70s for the housing of the 2 emergency diesel engines.

A complete structural and plant engineering renovation was carried out to make it suitable to host historical LLW conditioned in a cement matrix (e.g. resins and sludges).

The refurbishment concerned:

- realisation of new reinforcement and covering structures;

[410] "Circular Economy in Nuclear Decommissioning - Sogin experience". Bastianini E. (SOGIN). 2019.
[411] "The circular economy principles in the Italian nuclear programme. Sogin, a case study". SOGIN. 2019.

- construction of new plants;
- Installation of new cranes.

OPEC-2 facility was built in 1971-1976 as an extension of the OPEC 1 hot cell laboratory to house a laboratory for post-irradiation tests on fuel elements in the ENEA Casaccia Research Centre.

A complete structural and plant engineering renovation was carried out to make it suitable to host not unconditioned ILW (α-contaminated waste).

The restructuring concerned:

- seismic and structural adjustment;
- fire system and functional adjustment;
- construction of new plants.

However, it should be noted that it has not always been possible and convenient to reuse existing buildings as material/waste treatment facilities or as storage facilities. In some Italian experiences, it was decided for several logistical, economic, licensing, etc. reasons to demolish existing buildings and reconstruct them with similar volumes (e.g. Trino NPP Test Tank Building or Garigliano NPP former Compaction Building) or to directly build new buildings where the authorization processes allowed it.

Si.Co.Mo.R: a modular facility for radioactive waste conditioning

To optimise the use of resources and minimise the production of radioactive waste a possible solution is to avoid the construction of new waste treatment and conditioning facilities at each site and to opt for the use of dedicated, external and centralised waste treatment and conditioning facilities.

On the other hand, in Italy, the current licensing framework does not allow the treatment of materials coming from a specific facility/plant to take place in another site. For this reason, in Italy this optimisation on regional or national level need a strong collaboration with the nuclear safety authority before its implementation. Another element to be considered is the management of a large number of transports and the need for further temporary storage facilities on the sites identified as treatment/conditioning centres.

Meanwhile a modular plant, called SiCoMoR, “Sistema di Condizionamento Modulare per Rifiuti – Modular Conditioning System for Radioactive Waste” has been developed.

The main properties of the SiCoMoR facility are:

- the possibility to pre-assemble each module in a workshop;
- the flexibility to couple with other modules on the site of installation, by means of the coupling flange, which has a sealing gasket. This characteristic makes the facility completely “transportable”, enabling a new radioactive waste treatment facility to be brought into

service. The modularity concept also makes the facility operable in different configurations and for different production capacities;

- allowing it to be decontaminated, disassembled and transferred to another site for a new conditioning campaign;

In the current configuration SiCoMoR provides a conditioning process for radioactive waste using a homogeneous solidification by cementation in cylindrical containers with lost-paddle.

This solution has been judged simpler than the construction, at each site, of a fixed facility having the same purpose. SiCoMoR has a design life of 25 years and during its design life this facility is expected to be used on four different sites.

This approach allows an optimisation in technological systems, use of raw materials and production of waste resulting from decommissioning.

5.2.2.2 Preparatory activities for Jose Cabrera NPP decommissioning project

The José Cabrera NPP decommissioning project is a dynamic process that consists of a sequence of activities including, among others, disassembly, decontamination, declassification, demolition of buildings and the restoration of the site.

The main activities of the project may be broken down into seven major sequential groups and two of them are required in the overall process (radiological characterisation and radioactive waste management).

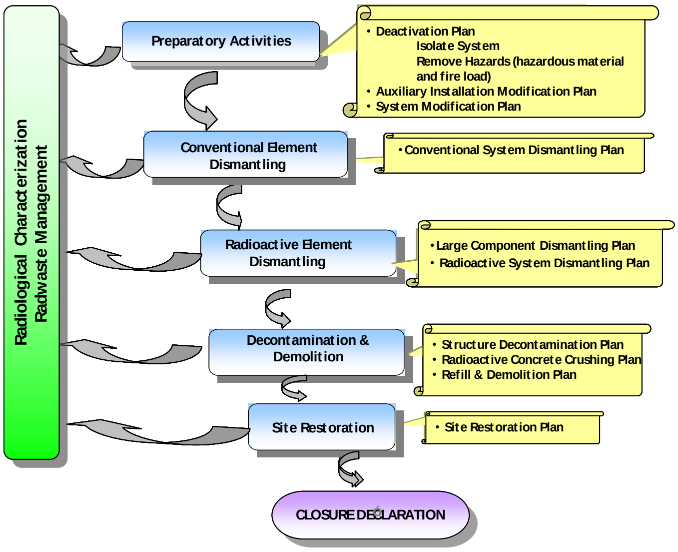


Figure 5.2-1 Project major sequential groups

Preparatory Activities

A number of preparatory activities were performed during the initial period of the project (2010-2012) in order to facilitate subsequent D&D operations:

- Radiological and Physical Inventory of the plant
- Decontamination of systems
- Discharging systems and components
- Draining circuits and systems
- Removal of hazardous materials
- Removal of non-radiological components
- Modification/Construction of new auxiliary systems / facilities

Deactivation Plan

With view to eliminate risks and interferences during work performance, the Deactivation Plan was applied, in order to remove from service those systems that were not required during dismantling. Moreover, inflammable or toxic products that might pose a problem during the works were removed from the plant.

The objective of the definitive deactivation was to assure that the system, or the associate equipment and components were out of service in a safe and stable form

All systems not required were deactivated (drained and/or de-energized) prior to equipment removal and isolated from other systems maintained to support decommissioning activities, according to the Definitive Tag-out Plan and Risk Reduction/Elimination Plans.

Figure 5.2-2 Tag-out of Components

Auxiliary Systems and infrastructures modification

The initiation of dismantling works requires that a series of auxiliary systems and installations are available and adapted to new requirements. Before the dismantling operations start, different systems, equipment and infrastructures need to be modified to resize them, avoid hazards and interferences adapting their functionality to the new activities to be performed on site.

The main auxiliary systems of Jose Cabrera NPP which were adapted during the preparatory activities were:

- Feed water systems (irrigation, potable water/water treatment plant, sanitary water)

- Fire protection systems (feed water, portable fire extinguisher, fixed systems, detection)
- Compressed air
- Ventilation systems
- Demineralised water.
- Auxiliary steam /Auxiliary boiler for steam generation.
- Fuel tank and fuel distribution.
- Waste management systems (resins & liquid effluents, drainage collection & tanks)
- Evaporator (treatment of liquid effluent)
- Electrical System
- Instrumentation, Communications & Control Room

Figure 5.2-3 Improvement of ventilation system

Mention may be made also for the improvement and adaptation of the Turbine building as a dismantling auxiliary installation. The turbine building housed the turbine generator unit and the auxiliary equipment needed to produce electrical power during operation of Jose Cabrera NPP. The turbine and its auxiliary systems were dismantled, and the building was transformed into an Auxiliary Dismantling Building (ADB). The purpose of the ADB was to condition the radwaste arisen from the dismantling of the containment building. The ADB was fitted with a decontamination workshop, facilities for radwaste conditioning and areas to store containers temporarily before they were shipped to the low and medium level disposal centre in El Cabril.

Figure 5.2-4 Initial situation of Turbine building and adaption as ADB

In order to optimise the management of the radioactive waste to be treated during the dismantling operations, existing waste storage areas I, II and III, as well as the clearance area, were refurbished.

Figure 5.2-5 Refurbishment of rad waste store

<u>Conclusions</u>

- The early start of the preparatory activities enables a clear optimisation of the dismantling program, so the detailed definition of its scope should begin as soon as possible.
- For the dismantling it is possible to reuse many of the existing systems and installations in the plant. In almost all cases it is necessary to adapt the elements to the new requirements of the dismantling activities, as well as their evolution throughout the dismantling, to avoid possible interference during work.
- It was very useful to establish a detailed plan of the needs and requirements for all phases of the project and integrate it from the beginning in the license documentation.
- Activities not covered by the license documentation or required in the technical instructions of the regulatory body were managed according to the design modification procedures.

5.3 Systems decontamination (internal)

Remediation of legacy nuclear facilities is a complex logistical challenge with an ever-increasing importance with the cessation of reprocessing operations in the United Kingdom. Whilst the ultimate aim is to perform remediation operations to an agreed end state, a series of discrete steps corresponding with ever increasing invasiveness to plant, is likely occur to in practice to enable hazard reduction.

The first of these steps is performed following plant operations and is usually called Post Operational Clean Out (POCO). Here existing plant facilities and services are used to reduce the hazard (initial decontamination) without significant breaks in containment. (i.e. keeping vessels and pipework largely intact.) Following use of 'native' reagents to 'wash-out' contaminated infrastructure, a step termed 'Enhanced POCO' retains the plant integrity as described in the previous step but enables the use of additional reagents and decontamination techniques with the aim to reduce hazards to the point of performing more invasive decommissioning activities (e.g. hands on decommissioning tasks.). (IAEA, 1989)[412], (Nuclear Decommissioning Authority (NDA), 2020)[413] systems decontamination (internal) therefore considers both POCO and Enhanced POCO decommissioning steps where decontamination of plant is required without significant breaks in containment using a range of techniques. By definition this defaults mainly to in-situ techniques with minimal additional plant intervention. (Note that decontamination in this context is not limited to reagent washing alone, but obviously limited by the scope in this context.) The following sections summarise potential technologies for use in the context of systems decontamination (internal) which are summarised in Table 5.3-2.

5.3.1 Chemical Dissolution/Washout

Whilst POCO considers reagents native to plant operations, the additional reagents potentially added for enhanced decontamination require further assessment, particularly in areas such as compatibility with downstream processes and waste treatment plants and subsequent production of suitable waste products for storage and disposal.

Thought also needs to be given to the most appropriate technique and volume of reagents required to be optimal in terms of decontamination but to minimise additional waste burdens associated with decontamination. For example, should a reagent be used to completely flood a vessel or would a series of targeted partial washes be more effective? When should POCO washes be stopped due to diminishing returns in terms of decontamination factors (DFs)?

Table 5.3-1 considers potential reagents for POCO type operations identified for use as potential wash-out liquors. For each reagent the following items were considered:

- Reasoning behind the selection of reagent

[412] IAEA, "Decontamination and Decommissioning of Nuclear Facilities," IAEA-TECDOC-511, Vienna, 1989
[413] Nuclear Decommissioning Authority (NDA), "Business Plan: April 2020 to March 2023," SG/2020/68, 2020

- Hazards associated with storage, handling and deployment
- Secondary chemical reactions, such as possible reactions that may occur at different pHs with changes in process conditions.
- Compatibility with downstream assets, plants and processes. (E.g. Vitrification or effluent process streams.)

In each case a reagent is classified according to hazard and brief assessment of compatibility of waste treatment plants is considered.

Table 5.3-1 Reagents Identified for Potential Use During POCO

Type	**Typical Reagent**	**Purpose**
Acids	Mineral (e.g. nitric, sulphuric, hydrochloric) Organic (e.g. malonic, oxalic, citric)	Dissolve metal oxide films. Lowers pH to increase metal ion solubility. (Organics tend to be used with other reagents.)
Oxidisers	Permanganate, cerium	Change in oxidation state to make metal ions more soluble.
Bases	Sodium carbonate, ammonium carbamate, sodium hydroxide	Neutralisation and removal of organics/paints/rust from mild steel surfaces.
Reductants	Hydrogen peroxide, hydrazine	Removal of organics/paints/rust from mild steel surfaces.
Chelators and Complexes	EDTA	Stabilise metal ions in solution.

5.3.2 Electrochemical

Electrochemical decontamination has been shown to be highly effective at removing the surface contamination from metals[414]. This process is typically achieved by immersing the metallic surface of interest in an electrolyte and connecting it to a positive terminal of a DC power supply, forming the anode. A counter electrode is also placed in the electrolyte to complete the electrochemical cell. When the electrical current is applied, metal on the surface of the anode is oxidised and dissolves into the electrolyte along with the radioactive species that have been incorporated into the surface. Further

414 C-Tech Innovation, “Electrochemical Nuclear Decontamination,” [Online]. Available: https://www.ctechinnovation.com/technology/elendes-electrochemical-nuclear-decontamination/

treatment of the contaminated electrolyte is then required. Electropolishing is a mature technology that is applied in a range of industrial applications, including metal purification, coatings, post-weld treatments and surface cleaning, to remove the surface of metallic items (Figure 5.3-1). Electrochemical decontamination offers a possible methodology for enhancing the effectiveness of native reagents without the complications of adding new chemical agents. Previous nuclear deployments have mostly utilised ex-situ immersion bath equipment, however various projects are developing electrochemical decontamination devices for the in-situ decontamination of plant components.

Figure 5.3-1 Image of a non-nuclear industrial electropickling cell

5.3.3 Abrasive Methods

In the context of the internal cleaning of vessels the application of abrasive methods is somewhat limited as this section effectively considers the in-situ cleaning of vessels. More 'traditional' surface abrasive methods; grinding, shaving and scabbling, typically are performed using hand-held tools and are therefore considered outside the context of this section. One technique that maybe considered in this context is the application of ultrasound.

Ultrasound requires the surface to be decontaminated to be immersed in a fluid. The application of ultrasound via a transducer generates cavitation within the fluid which dislodges the contamination from the surface.[415]

Traditionally ultrasonic cleaning has been performed by immersing the item to be decontaminated within a treatment bath, although transducers have been adapted to work in the context of in-situ tank decontamination[416] (Lebedev, Krasilnikov, Vasiliev, Dubinin, & Yurmanov, 2012). Other methods for in-situ deployment include external transducers that are capable of being clamped onto the outside

[415] K. Suslick, "Sonochemistry," Science, vol. 247, no. 4949, pp. 1439-1445, 1990
[416] N. Lebedev, D. Krasilnikov, A. Vasiliev, G. Dubinin and V. Yurmanov, "Development and Application of the Ultrasonic Technologies in Nuclear Engineering," in International Conference NPC, Paris, 2012

of pipes to enable in-situ cleaning of pipework to effectively provide in-situ cleaning without breaking into pipe containment.[417]

5.3.4 Wet Surface Cleaning

5.3.4.1 Water jetting

Water jetting (including low, high and ultra-high pressure) involves spraying items to be decontaminated with water to remove contamination from a surface. This can also be used to remove paint and other coatings from surfaces and can be performed in or ex-situ. It is a simple but relatively powerful technique that has had many reported application and future uses for decontamination at Sellafield.[418]

5.3.4.2 Foams

Foams offer a low volume alternative to liquor based decontamination processes for the delivery of decontamination agents. They have the advantage of generating a large reactive surface area for decontamination which, once the foam is collapsed, generates a minimal amount of contaminated liquor for disposal.[419] A foam is essentially an unstable colloidal system based on a liquid and gaseous phase which requires a surfactant to be generated. The chemistry relating to the stability of foams is quite critical to the 'quality' of the foam produced (i.e. lifetime before collapse) and therefore the decontamination performance of the technique. A wide range of reagents can be added to the foam formulation for the purposes of decontamination (e.g. acids or oxidising agents). Care needs to be taken before deployment to consider principle areas such as the potential of the media to re-foam after use and the route for disposal given the use of surfactants in the formulation. Examples of use include the decommissioning of transport flasks and various nuclear facilities. [420]

[417] Brunel University, "Brunel Innovation Centre - Our Projects," [Online]. Available: https://www.brunel.ac.uk/research/Institutes/Institute-of-Materials-and-Manufacturing/Structural-Integrity/Brunel-Innovation-Centre/Our-Projects. [Accessed 1 July 2020]

[418] G. Yates, "Multiple Applications of Water Jetting in the Nuclear Industry," in Fluid Mechanics of Cleaning and Decontamination SIG Summer Conference, Cambridge, 2018

[419] G. Boissonnet, M. Faury and B. Fournel, "Decontamination of Nuclear Components Through the Use of Foams," in Foams and Emulsions. NATO ASI Series (Series E: Applied Sciences), Dordrecht, Springer, 1999

[420] G. Boissonnet, M. Faury and B. Fournel, "Decontamination of Nuclear Components Through the Use of Foams," in Foams and Emulsions. NATO ASI Series (Series E: Applied Sciences), Dordrecht, Springer, 1999

Table 5.3-2 Summary of Technologies Identified for Systems Decontamination

Technology	**Details of Application (Field/Type)**	**Advantages**	**Disadvantages**	**Areas for Development/Gaps**
Chemical Dissolution/Washout	Immersion Recirculation Surface contact Sub-surface application	High DF achievable Well established Suitable for complex geometries. (E.g. Pipes) Suitable for almost all radionuclides	May generate large waste volumes depending on application. Can be expensive and time consuming. Corrosive to base materials	Impact/optimisation on waste streams and compatibility with waste forms. In situ characterisation to optimise washout.
Electrochemical	Immersion In-situ Decontamination (Electro-polishing Electro-etching)	Quick processing time High reliability and efficiency Generates small amount of secondary waste Can remove majority of radionuclides and other metallic contaminants Established method.	Requires application of electrical current. Surface needs to be electrically linked to be decontaminated. E.g. No painted or coated surfaces. Can only be used with electrically conducting materials.	Impact/optimisation on waste streams and compatibility with waste forms
Abrasive Methods	Ultrasound	Improvement in DF with minimal additional reagents.	Much experience using technology ex-situ, but little experience in-situ.	In-situ deployment is lower TRL. Development would enable wide potential applications.

Technology	Details of Application (Field/Type)	Advantages	Disadvantages	Areas for Development/Gaps
Wet Surface Cleaning	Water Jetting In-situ, (ultra-high/high/low pressure)	Simple, well known technique. No additional chemicals required.	Comparatively large amount of liquor generated. Contamination maybe dispersed widely during cleaning.	Optimisation and understanding of secondary contamination due to fog generated during the technique.
	Foams	Low volume of reagent required compared with other wet techniques.	Complex chemistry. I.e. In terms of foam stability	Formulation and optimisation of foam for task. Disposal route compatibility.

5.3.5 Experiences/Case Studies

5.3.5.1 Italy – Trino NPP Chemical Decontamination of the Steam Generators

In the period March / July 2004, at the Trino NPP, the "in line" Chemical Decontamination of the Steam Generators was carried out.

The process (multi-step and multi-cycle) was based on an Acid Permanganic/Chemical Oxidation Reduction Decontamination with the use of UV light to reduce the redox potential of the solution and make it more aggressive for the attack of the metal.

In total, 16 decontamination and 3 clean-up cycles were performed. The Decontamination Factor (DF) was about 100. The total activity removed was about 70% of the initial one, mainly due to Co-60.

The average thickness of oxide removed was 10.3 µm with a total amount of metal removed equal to 298.6 kg.

6. Dismantling

Decommissioning of nuclear facilities involves the tasks of segmentation of metallic components and the cutting and demolition of reinforced concrete structures. Mostly, dismantling and segmentation activities precede the demolition tasks of a building. Dismantling and segmentation refer to cutting activities (piping, pumps, tanks, and reactor internals) and achieve similar ends while demolition refers to completely demolish a building to reduce it to rubble. Various techniques have been used for dismantling the components and structures and new techniques are being developed continually. Dismantling methods are chosen based on radiological criteria, availability of suitable equipment, complete knowledge of the problem, structured timings, and cost-effectiveness of the proposed solutions.

Safety of the workers in terms of minimum dose rate is an important consideration. This can be primarily accomplished by introducing remote operating technologies for the dismantlement. The other factors that highlight the feasibility of the various dismantling techniques include a reduction in the volume of waste, working efficiency, and impacts on the surrounding environment. This chapter presents an overview of different sets of topics that are largely highlighted as the state of the art decontamination, segmentation, and demolition techniques used for the decommissioning of nuclear facilities with the emphasis on upcoming needs in the decommissioning projects and past experiences in using different technologies for particular tasks.

A number of national and international initiatives, programmes and working groups are gathering expertise on tasks and knowledge about dismantling of nuclear facilities as a whole or on specific sub topics. Some of them are listed below.

International initiatives

<u>IAEA Initiatives:</u>

- DAROD project:
  This project aimed at sharing Experiences derived from the challenges associated with Decommissioning and Remediation of Damaged Nuclear Facilities (DAROD) and at disseminating practical information from the various stakeholders (regulators, owners, operators, governments and the public) involved in the post-emergency phase.
  It was based on 8 case studies: Fukushima Daiichi NPS in Japan, Three Mile Island NPP, Unit 2, in USA, Chernobyl NPP, Unit 4, in Ukraine, A1 NPP in Slovakia, First Generation Magnox Storage Pond in Sellafield (UK), Marcoule nuclear site in France, Industrial Uranium Graphite Reactors (IUGR) in Russia, Al Tuwaitha in Iraq.
  In this domain, use of robot technology is essential in areas that are too hazardous for people, or where access is difficult.

Referring to the case studies, products for dealing with high levels of radioactivity and damaged situation do not exist on the shelf for immediate availability and it was noticed that there was a lack of proven technology.

In all cases it was realised that training of the workers in their own facilities was necessary to achieve system performance with higher reliability, higher productivity and improved safety.

These developments also took into account the difficulties of access which are specific to each case.

- IDN WIKI:

  The IAEA International Decommissioning Network (IDN) created a Web based tool to support information sharing among its members. Large part is dealing with remote systems and associated lessons learned.

- IAEA-TECDOC-1817: "Selection of Technical Solutions for the Management of Radioactive Waste" [421]

  This document gives some examples of methodologies for the management of the treatment of radiological embedded elements with an overview of the particular actions to convert the arising waste material into defined waste packages, fulfilling specific waste criteria. For a smooth operation, the procedure steps have to be compatible with each other. Therefore, an overall management system is needed.

  The document describes a linear decision tree approach to evaluate specific technological options, Cost-based approach, Risk assessment and Multi-Attribute Analysis

NEA Initiatives

- Report of the NEA Committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM):

  The NEA Working Party on Decommissioning and Dismantling (WPDD) reviewed the current labour-intensive approach to decommissioning and dismantling and provided a report on the R&D and Innovation Needs for Decommissioning of Nuclear Facilities. This report provided an update on the challenges of current R&D and reported the WPDD consensus concerning priorities for future R&D and opportunities for collaboration among organisations and NEA member countries [422].

  This report consists of 300 pages with more than 700 references and also addressed robotics and remote-controlled tools through the themes "Characterisation" and "Technologies for

[421] IAEA-TECDOC-1817; SELECTION OF TECHNICAL SOLUTIONS FOR THE MANAGEMENT OF RADIOACTIVE WASTE; INTERNATIONAL ATOMIC ENERGY AGENCY; VIENNA, 2017; ISBN 978–92–0–104717–5.

[422] NEA/OECD, Radioactive Waste Management, "R&D and Innovation Needs for Decommissioning of Nuclear Facilities", 2014.

segmentation and dismantling". In a further report [423], 2 challenges related to robotics and remote-controlled tools were finally retained, out of a list of 7 challenges: use of remote sensing and satellite and use of robotics.

- Reports of the NEA Co-operative Programme on Decommissioning (CPD):
  NEA's Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects (CPD) exchanges and shares the information from operational experience in decommissioning installations that is useful for current and future projects.
  One of these reports highlights the generic results obtained by a CPD Task Group on Decontamination and Dismantling of Concrete Structures that undertook a comprehensive review of proven technologies and methods for decontamination, demolishing and disposing of concrete structures [424].
  Another report produced in 2011 - "Remote handling techniques in decommissioning" [425] - describes generic results obtained by a Task Group analysing the needs for remote technologies, existing technologies able to meet these needs, the lessons learned and identified where improvements or further developments should be made in this domain.
  Remote technologies described in the report are normally categorised into several areas, such as detection, segmenting, decontamination, handling and sampling equipment.

- International Workshop on the Use of Robotic Technologies at Nuclear Facilities [426]
  This workshop sponsored by NEA was organised in USA in 2016 by US National Institute of standards and Technology (NIST) and chaired by NRC, DOE and NIST. It was an opportunity to exchange information between government agencies, industry and academia and:
    - better understand the state of robotics and remote systems in challenging non-nuclear environments and their potential applications at nuclear facilities;
    - Get Ideas and strategies for enhancing existing databases or compendiums with quantifiable performance data;
    - Get Consensus on standard test methods to assess performance;
    - Understand End-user strategy for setting thresholds of capabilities necessary for deployment;
    - Understand the Regulatory approach for technical review of strategies for integrating technology, standards, training and regulations to address implementation;

[423] Needs and emergency technologies for decommissioning, Gerard Laurent, In Solutions, Norway, digidecom 2017

[424] NEA/OECD, Radioactive Waste Management (NEA/RWM/R(2011)1), "Decontamination and Dismantling of Radioactive concrete structures", 2011. https://www.oecd-nea.org/upload/docs/application/pdf/2020-01/rwm-r2011-1.pdf

[425] NEA/OECD, Radioactive Waste Management (NEA/RWM/R(2011)2), "Remote Handling Techniques In Decommissioning", 2011. https://www.oecd-nea.org/upload/docs/application/pdf/2020-01/rwm-r2011-2.pdf

[426] Video of the workshop on NIST Website at https://www.nist.gov/news-events/events/2016/02/international-workshop-use-robotic-technologies-nuclear-facilities

- Identify ground-breaking opportunities, and improvement needed: user-friendly, reliability, flexibility and minimum preventive maintenance.

➢ RWMC (Radioactive Waste Management Committee) and CDLM[427] (Committee on Decommissioning and Legacy Management):
An Expert Group on the Application of Robotic and Remote Systems in the Nuclear Back-end (EGRRS) started in 2019 [428] with the objectives:
- to find the different factors influencing the development of RRS (Robotic and Remote Systems) in radioactive waste management, decommissioning and legacy management;
- to connect multiple stakeholders (R&D institutions, implementers, users, regulators, etc.) interested to foster harmonised understanding, terminology and approaches;
- to identify state of the art;
- to organise participation of students (middle school – university);
- to promote potential professionals - nuclear competency building;
- to train professionals;
- to demonstrate to decision-makers the benefits of robotics application in nuclear projects and engage their participation and support.

EGRRS launched a Survey mid-September 2020 on Barriers and Impediments: the goal of this survey is to identify barriers and impediments that hinder the application of Robotic and Remote Systems (RRS) in the nuclear back-end field. Based on the initial challenges identified by the EGRRS Matrix, further challenges and ways to address them will be elaborated.
In addition, a second survey is being developed to identify the "Current status and trends" of robotics and remote systems in the nuclear back-end field. EGRRS members will ensure that both surveys will be compatible and not overlap.

➢ NI 2050 initiative [429]
The NI 2050 initiative described Robotic and Remote Systems (RRS) application as a crosscutting issue that is important in all stages of the nuclear fuel cycle.
A project was selected aiming at embedding an emerging multinational project of nuclear decommissioning demonstrator for graphite nuclear reactors[430].
In the field of remote controlled or robotics operations, this project will address (also through the use of mock-up tests):
- Remote concrete cutting and drilling tools considering concrete characteristics and different configurations;

[427] https://www.oecd-nea.org/rwm/cdlm/
[428] NEA/OECD, "Summary of the NEA Workshop on the Application of Remote and Robotic Systems in Nuclear Back-End Activities–Way Forward in System Implementation", 30-31 January 2019. https://www.oecd-nea.org/rwm/workshops/2019/robsysnba/docs/summary.pdf
[429] http://www.oecd-nea.org/ndd/ni2050/
[430] R&D Cooperative Programme Proposal Nuclear Reactor Dismantling Demonstrator https://www.oecd-nea.org/download/ni2050/documents/04PIERACCININI2050Decommissioning_.pdf

- Remote handling of large concrete blocks;
- Remote cutting and handling of thick metallic structures (vessel, peripherals, core support…).

➢ "ARTERD" project in the NEST Framework [431],[432]:
The NEA launched the Nuclear Education, Skills and Technology (NEST) Framework in partnership with its member countries to help address important gaps in nuclear skills capacity building, knowledge transfer and technical innovation in an international context. The NEST Framework is developed as an NEA joint undertaking gathering private and public organisations from interested countries (not-necessarily NEA member countries).
The NEST ARTERD Project, led by JAEA/CLADS and the University of Tokyo, has been dedicated to advanced remote technology for decommissioning under intense gamma-ray radiation environments (e.g. robotics, virtual reality).
The NEST ARTERD Project also addresses other technologies such as radiation hardness and smartness, radiation imaging, advanced radiation measurement and remote spectroscopic analysis and so on which should have close cooperation with Advanced Remote Technology in order to grasp working environments of decommissioning.

European Commission Initiatives:

The European Commission organised a number of initiatives including:

➢ H2020 LD- SAFE [433]:
The aim of the NFRP-2019-2020-09 call was to capitalise existing technologies for characterisation and risk assessment, dismantling, on-site waste management and environmental remediation in order to gain needed efficiencies in the decommissioning of nuclear power reactors. One very challenging dismantling task in the focus of the industry is the segmentation of the reactor pressure vessel and internals as limitations are known for all conventional cutting techniques currently used including mechanical cutting, plasma arc cutting or abrasive water jet cutting.
In this context, the laser cutting technology for nuclear dismantling, an adaptation from the manufacturing industry developed by over a decade of R&D efforts, is identified in Europe and elsewhere in the world as a promising alternative.
The objective of the LD-SAFE project is to validate the laser cutting technology for the dismantling of the most challenging components of power nuclear reactors in air and underwater.
LD-SAFE project will remove the last barriers to enable the replacement of conventional cutting techniques and prove by 4 specific objectives that the technology is mature (TRL7):

[431] https://www.oecd-nea.org/jcms/pl_21786/nuclear-education-skills-and-technology-nest-framework
[432] https://www.oecd-nea.org/jcms/pl_24328/nest-advanced-remote-technology-and-robotics-for-decommissioning-arterd
[433] http://www.ldsafe.eu/

1) Demonstration of the capabilities of a versatile laser cutting solution to address the key technical challenges in decommissioning of large nuclear facilities
2) Environmental and safety assessment of the implementation of laser cutting for nuclear reactor decommissioning
3) Technical validation of the laser cutting prototype in operational environment (TRL7)
4) Demonstration of the economic advantage of using the laser cutting technology for the forthcoming reactor decommissioning market.

- H2020 INNO4GRAPH (INNOvative tools FOR dismantling of GRAPHite moderated nuclear reactors [434]:

  The decommissioning of closed graphite moderated nuclear reactors worldwide is still in its early stages with most reactors in "safe store" condition. For these reactors, there are still considerable industrial and technical challenges that remain to be tackled even after more than 30 years of operational shutdown of the first unit. Challenges related to the dimension and complexity of the structure as well as costs for decommissioning urge for novel, more performant, safer and cost-efficient solutions for dismantling operations.

  INNO4GRAPH, with the participation of all European graphite reactor operators aims at developing a set of physical and digital tools and methods to be used both upstream of dismantling operations (for material characterisation and decision-making) and during dismantling operations (for handling and cutting). Tools such as 3D modelling of dismantling scenarios, measurement tools for mechanical and physical properties, mock-ups of the graphite stack and laser-cutting during dismantling operations will be developed.

- H2020 RoMaNS:

  RoMaNS (Robotic Manipulation for Nuclear Sort and Segregation) project (2015-2018) has developed advanced robotics technologies needed for handling hazardous nuclear waste. Firstly, the development of a new robot arms and hands that is capable of highly dexterous and sophisticated behaviours, while still being resilient in high radiation fields. Secondly, in this project, "tele-presence" systems which enable human operators to control the remote "slave" robot while feeling the forces experienced by the robot when it contacts objects is introduced. Additionally, "autonomous" robot control methods were also developed. Now the operator can simply click the cursor on an object displayed on the video monitor and the robot's vision system will detect the object and automatically guide the robot to grasp it. Finally, in this project, a variety of novel approaches to "shared control", where both human and AI collaborate to control the robot, with the human controlling part of the motion, while the AI takes care of other aspects was developed [435].

- EUROBOTICS:

[434] https://cordis.europa.eu/project/id/945273
[435] https://cordis.europa.eu/project/id/645582/reporting

It aims to boost European robotics research, development and innovation and to foster a positive perception of robotics. Currently, it engages in the preparation of the envisaged partnership on AI, Data and Robotics in the Horizon Europe programme (2021-2027).

- SPARC:

  The SPARC's roadmap for the robotic development in Europe could be a good reference and example for international initiatives on Robotic and Remote Systems (RRS) in the nuclear back-end.

Other Initiatives

- The ISO developed standards, applicable to robotics as well as:
  - Criteria for design and operation of confinement systems for nuclear worksite and for nuclear installations under decommissioning [436];
  - Robots and robotic devices — Safety requirements for industrial robots — Part 2: Robot systems and integration [437] ;
  - Within Technical Committee (TC) 85/SC 2, there was a Working Group 24 on "Remote handling devices for nuclear applications". But this group was dissolved because of lack of expert attendance.

  The ISO developed a number of standards, applicable to robotics and is currently developing new standards including R*obotic Vocabulary (ISO/CD 8373 Robotics)* which could be taken considered in a future initiative.

- ENRICH – A New Robotics Competition:

  ENRICH is an initiative by The European Robotics group to bring together roboticists, radiological and nuclear professionals, and specialists to give a better insight into possible robotic applications. European Robotics is a non-profit organisation bringing together representatives from the user community, industry and the research community in the field of Hazardous Materials Incident Response Operations[438].

- EPRI Report 3002018418 [439]:

  This discusses the conceptual design of an alternative system and approach for segmentation of reactor vessel internals. Based on a time-and-motion study, the system may reduce segmentation time by 60% for full deployment, and by 20% for partial deployment. The approach also simplifies project planning. The report includes a fully developed conceptual design, including component specifications. Hyperlinks to video files showing system operation are included throughout the report. The hope is that segmentation vendors consider adoption of all or portions of the

---

[436] ISO 16647:2018

[437] ISO 10218-2:2011

[438] https://enrich.european-robotics.eu/

[439] https://www.epri.com/research/products/000000003002018418

conceptual design. EPRI does not intend to continue system development through to deployment, but would be happy to collaborate on such development. Likewise, EPRI does not intend to patent or license the design.

- Over the past years, EPRI has developed and published several lessons learned documents and workshop proceedings related to decommissioning of nuclear power plants.

  These lessons learned documents and workshop proceedings have provided a sound reference base for reactor facilities that will eventually undergo decommissioning. Many of these experience reports and workshops were developed in conjunction with U.S. nuclear plants engaged in different phases of decommissioning.

- EPRI Report 3002018420 [440]:

  It discusses the development and demonstration of an autonomous system for radiological characterisation of large land areas and floors. In short, the project involved combining an existing autonomous robot with an existing radiation detection system. The performance of the system was demonstrated at Kewaunee in 2019 with the support of Dominion.

  The report provides specifications, results and lessons learned, as well as a state-of-the-art review. A video of the Kewaunee demonstration is embedded in the report. The system performed quite well overall. The hope is that these results may be used to support development and deployment of similar systems. EPRI does not intend to license or commercially deploy the system, nor to offer the system for sale or rental. The intent of the project was simply to show the approach was viable. Alternative robots may be used, however if there is interest in using the EPRI robot, EPRI can facilitate arrangements.

- German-Japanese Symposium on Technological and Educational Resources for the Decommissioning of Nuclear Facilities:

  The German-Japanese Symposium on Technological and Educational Resources for the Decommissioning of Nuclear Facilities was held in Tokyo with around 170 participants from industry and science. In overview on the newest technical innovations, developments and trends and presented new technologies and strategies for the decommissioning of nuclear facilities was given and discussed by the participants[441].

  The International Research Institute for Nuclear Decommissioning (IRID) have teamed up with TEPCO's Decontamination and Decommissioning (D&D) Engineering Company which focuses on R&D required for the decommissioning of the Fukushima Daiichi Nuclear Power Station (NPS). The integrated management of R&D being carried out by IRID follows the Mid-and-Long-Term Roadmap for the Decommissioning of TEPCO's Fukushima Daiichi NPS Units 1-4. IRID has developed remotely-controlled shape-deformation type robots that can enter the Primary

---

[440] https://www.epri.com/research/products/000000003002018420

[441] https://www.dwih-tokyo.org/en/activities/event-reports/german-japanese-symposium-on-technological-and-educational-resources-on-the-decommissioning-of-nuclear-facilities/

Containment Vessels via a narrow access ports for the decontamination of the routes to allow access for workers and to detect leakage points. That joint symposium helped to share the knowledge on innovative technologies for decontamination and segmentation of large radioactive components.[442].

- US – DOE-EM (Department of Energy's Office of Environmental Management):
  DOE provided a Robotics Roadmap in 2018 that identified needs for which robotics and remote system may provide benefits. Based on these needs, key technologies were identified and assessed to provide specific recommendations to stakeholders, leadership, sites, and the technology development community [443].

- DARPA:
  The Defense Advanced Research Projects Agency (DARPA) is an advanced-technology branch of the U.S. Department of Defense. The purpose of the agency is to try out new technologies and make them operationally ready, if possible, and to reach beyond current technology to do something new. They have promoted robotic technology by organising DARPA Robotic Challenge (DRC). The primary technical goal of the DRC is to develop human-supervised ground robots capable of executing complex tasks in dangerous, degraded, human-engineered environments[444].

- NIST:
  U.S. National Institute of Standards and Technology (NIST) objective is to develop and deploy measurement science, standards, and test methods that advance manufacturing robotic system performance, collaboration, agility, autonomy, safety, and ease of implementation to enhance U.S. innovation and industrial competitiveness. They have organised an international workshop on the use of robotic technologies at Nuclear facilities with collaboration of NEA/OECD, US.DOE, US.NRC, CNSC, and UK Atomic authority in 2016[445].

- JAEA:
  It established the Collaborative Laboratories for Advanced Decommissioning Science (CLADS) in 2015. It conducts R&D activities related to decommissioning of the Fukushima Daiichi NPP of the TEPCO. Their research is focused on remote technologies for radiation imaging systems and in-core detection using laser and fibre optics. Moreover, they are also working on decommissioning technologies and waste management [446].

---

[442] German Research and Innovation Forum Tokyo, "German-Japanese Symposium on Technological and Educational Resources for the Decommissioning of Nuclear Facilities", Conference preceding, April 21, 2015.
[443] DOE EM "Research and Technology Roadmap, Robotics and Remote systems for Nuclear Clean-up", 2018.
[444] https://www.darpa.mil/program/darpa-robotics-challenge accessed 13.01.2021
[445] https://www.nist.gov/news-events/events/2016/02/international-workshop-use-robotic-technologies-nuclear-facilities accessed 13.01.2021
[446] https://clads.jaea.go.jp/en/rd/

- METI/ NDF:
  measures and projects towards technological development for decommissioning

- NRTDC:
  contribution to decommissioning of Fukushima Daichi 1F building through remote operated technologies.

- IRID and TEPCO's worker's training programs.

6.1 Segmentation of large irradiated metallic components

In activities related to the NPPs or Back End facilities decommissioning, cutting and segmentation of components and structures having different sizes, geometries and materials is required (e.g. stainless-steel, carbon steel, concrete, plastic materials, etc.). Some examples in NPPs are the reactor pressure vessel and its internals, steam generators and heat exchangers, piping and supports, tanks, concrete structures, etc. Some examples in Back end facilities are reprocessing plants dissolvers or fission products evaporators.

Also the environmental conditions in which the cutting tools will operate could be widely different ranging from under water activities (e.g. for reactor pressure vessel internals) to those carried out in the presence of contaminated objects/environments or in the presence of high radiation fields.

The following factors should be considered to select the appropriate cutting technology and the related equipment to be used:

- materials and geometries to be cut;
- maximum cutting thickness;
- environment of use (in air and/or underwater);
- possibility to operate by remote control;
- quantities and types of secondary wastes produced during the operations;
- risks from nuclear and conventional safety point of view (e.g. spread of contamination, doses to the operators, dangers/hazards e.g. explosive gases, high energy, etc.);
- risk of contamination or ease of decontamination of the tools to use;
- timing constraints for the operations;
- costs, including additional components, spare parts and consumables and necessary human resources.

For the above reasons, it is possible to state that there is not a special technology suitable for any situation and that the choice of the appropriate cutting technique must be examined on a case by case basis taking into account, at least, all the parameters mentioned above.

6.1.1 Description of techniques [447]

This section focuses on the most commonly used methods for the segmentation of Reactor Vessel Internals (RVIs), Reactor Pressure Vessels (RPV), other large components and systems in nuclear power plants and facilities during decommissioning activities. The techniques considered are:

- Plasma Arc Cutting (PAC)

[447] OECD Nuclear Energy Agency (NEA) – R&D and Innovation Needs for Decommissioning Nuclear Facilities (2014).

- Abrasive Water Jet Cutting (AWJC)
- Electro-discharge Machining (EDM)
- Metal Disintegration Machining (MDM)
- Arc Saw Cutting (ASC)
- Mechanical Cutting Methods (Saws etc)
- Oxy-fuel Gas Cutting
- Laser Cutting

These methods are the basis for the specialist equipment currently used for segmentation. Other available techniques that have not been adopted in practice or techniques under development are not considered here.

6.1.1.1 Plasma Arc Cutting (PAC)

PAC is classified as a thermal cutting method. There are two basic types of thermal cutters; flame producers and arc producers. The more common are the flame-producing methods where a flame is established by igniting fuel gases. The arc-producing technique establishes electrical arc between the tool and the workpiece. In either method, a section of the workpiece is melted away.

The PAC technique is based on the establishment of a direct current arc between a tungsten electrode and any conducting metal. The arc is established in a gas, or gas mixture that flows through a constricting orifice in the torch nozzle to the workpiece. The constricting effect of the orifice on both the gas and the arc results in very high current densities and high temperatures in the stream.

The stream, or plasma, consists of positively charged ions and free electrons. The plasma is ejected from the torch nozzle at a very high velocity and, in combination with the arc, melts the contacted workpiece metal and blows the molten metal away. A typical cut starts at the metal edge, and a through cut is made in a single pass by moving the torch at a fixed rate of speed in the direction of the cut and at a fixed nozzle spacing relative to the workpiece.

The PAC technique can also be used with a water injection option. This technique directs a radial jet of water that impinges on the plasma stream near the torch nozzle. The effect of the water jet is to further constrict the plasma stream, which results in even higher current densities. The cutting effect is a narrower kerf (the width of the cut), higher quality cut surface, and reduced smoke generation. Most reactor internals segmentation is accomplished underwater, so the water injection method is not used. However, some plasma cutting on internal components has taken place in a dry containment area, and this added feature may be considered. Controlling the water flow and generation of additional wet waste will most likely offset any benefits of using this water injection option.

A typical PAC system requires a direct current power supply of up to 1,000 Amps. Mechanical and/or automated PAC systems and hand-held PAC systems are available. An automatic PAC system would include torch positioning equipment, torch travel system, air, starting gas, and plasma gas supply systems, pilot arc high frequency power supply, PAC power supply and mechanical travel controls.

Off-the-shelf track systems are available for use with small plasma torches. In most cases however, specially fabricated designs are required when segmenting reactor components. The tracks may be fastened to the workpiece by various methods, including strap and magnetic fasteners (for ferrous materials). While portable, hand-held PAC systems are available from various manufacturers as an off-the-shelf item, in most instances, mechanically driven automated PAC systems are used for reactor component segmentation. The torch is advanced and guided on the tracks by a motor that is remotely operated. Remote operations reduce operator exposure.

A typical underwater PAC system would consist of a manipulator and plasma torch that could be operated from a remotely located cutting control console. The console allows programming direction, speeds, speed transitions, gas flows, and start delay times to control the underwater PAC process. The torch manipulator system would include a radial drive system and a torch standoff system to ensure proper standoff is obtained. Torch standoff is one of the most critical parameters in PAC and also one of the most difficult to control, especially when cutting over uneven surfaces or obstructions.

Although a positive mechanical torch touch system can be used to obtained proper standoff, it is very difficult to maintain. Underwater TV camera systems are useful in monitoring and setting the torch standoff as well as for inspecting cuts and placement of handling equipment. Hydraulic drive systems would most likely be the method of choice on the basis of design and fabrication simplicity, operational reliability, maintainability and cost

Hand-held plasma torches can be used in-air to segment some components that have a lower radioactive dose rate as long as strict contamination control can be achieved. Respirators for torch operators are required, as are high-volume ventilation systems that draw the contaminated fumes through high-efficiency particulate air (HEPA) filters. Additional methods of controlling exposure and contamination are the use of torch handle extensions and fabricated, non-flammable contamination control envelopes (containments) similar to a glove box in which the cutting is performed. The design of the envelope must allow fresh air to circulate freely to help filter contaminated fumes. When using this dry cutting approach and filtering the air, it is very important to use spark arrestors at the capture point on the ventilation intake to prevent molten metal or sparks from entering the pre-filter or HEPA filter system.

As a rule, hand-held plasma torches should not be used for materials that are more than 1½-in.(3.8 cm) thick because of their reaction to the gas flow and plasma jet. Portable units are rated by amperage; as the amperage rating increases, so does the cut capacity. For example, a 1 in. (2.5 cm) think stainless steel flat plate can be cut with a 100A unit at approximately 12 in./min. (30 cm/min). Generally for machine operation, a 50A unit can cut material that is approximately 5/8-in.(1.6 cm) thick. This operating efficiency is decreased by 10–15% when hand-held units are used because reaction forces hinder effective control of the torch.

The PAC technique is capable of cutting all metals. As indicated by performance results, thicker sections of material can be cut in air than underwater. In some instances, greater than 8 in. (20.3 cm) of stainless steel were able to be cut in air, while only 5 in. (12.7 cm) thicknesses were able to be cut underwater. Moreover, the design basis for PAC systems indicates that thicker sections of carbon steel than stainless steel can be cut.

PAC systems can also be used to pierce metals (in this practice, the cut does not have to start at the edge of the segment to be cut). Some data shows that metals could be pierced with a nozzle standoff distance of 0.625 in. (1.6 cm). This standoff distance is about twice the distance used during cutting, minimises blowback of the molten metal, and extends the life of the torch head. Thicknesses of materials being pierced will vary depending on the type of metal and whether it is in air or underwater.

PAC systems can provide some of the most rapid cutting rates of all the other segmentation methods. These systems, especially hand-held torches, can be deployed in multiple configurations. In some instances, making a hole in a component, cutting for preparation of the installation of a more controllable segmentation method, or finishing a cut to separate a component are times when the flexibility of a PAC system can prove to be invaluable.

Regardless of the segmentation method chosen as the primary method of cutting, there should always be a portable PAC system with capable and qualified operators available. Even when PAC was not the primary method of cutting, its limited use has proven to be an invaluable addition to a segmentation project.

6.1.1.2 Electro-Discharge Machining (EDM)

EDM is based on the principle of thermomechanical erosion in metals through the accurate control of fine electrical discharges (i.e. sparks). The spark is generated through the gap between two charged electrodes, a cutting tool and a workpiece, both of which are submerged in dielectric fluid. As the tool is energized, a potential difference is established with the workpiece, which is large enough to cause a breakdown of the dielectric fluid. Arcing then occurs across the gap, resulting in localised heating. Small molten particles lift off the surface of the metal as a result of the thermal expansion caused by the localised heating. The dielectric fluid also acts as a cooling medium that resolidifies the particles

The cutting rates are proportional to the amount of energy at the gap between the tool and the workpiece. The gap controls the energy and is therefore critical to the process. The system operator can adjust the gap as a function of voltage across the gap. The frequency of the discharges controls the resulting surface finish. Higher discharge rates produce rougher surfaces, which in decommissioning activities may be of little concern. Removal rates are influenced by the average current in the discharge circuit; they are a function of the electrode characteristics, the electrical parameters, and the nature of the dielectric fluid. In practice, this rate is normally varied by changing the number of discharges per second or the energy per discharge. The tool has great influence on the removal rates. It is usually made of copper-tungsten, graphite, or copper alloys. Tool wear is important to both cost and tolerances. The ratio of tool material removed to workpiece material removed varies with different combinations and should be kept low.

All materials that are sufficiently good conductors of electricity can be cut with this process. By utilising electrodes fabricated in the shape of the desired hole, penetrations in virtually any shape can be made with EDM. An approximation of removal rates for continuous operation is 5 in^3/hr. (82 cm^3/hr). This is relatively slow compared with other methods. This technique has the benefit of not generating any material chips, slag, or other large particles; and it can be performed at low temperatures. Because the tool does not come in contact with the workpiece, relative machining forces are low, a factor that makes this process amenable to remote operations.

One of the key parameters in the control of the process is the distance between the electrodes which ideally should remain constant but varies as the workpiece is machined and the discharge electrode wears. Control systems are used to counter this by maintaining the standoff distance and also by controlling the discharges in such a way that the electrode wear is reduced. To make a specific cut, the EDM tool is guided along the desired path very close to the work; ideally it should not touch the workpiece, although in reality this may happen due to the performance of the specific motion control in use.

Because EDM is performed in a dielectric fluid, it is ideally suited for underwater applications and has been used to perform underwater modifications to reactor internals. In particular it is well suited for hole boring operations or point machining, e.g. the cutting of brackets and bolts that are inaccessible to other tooling. Bolts that cannot be de-torqued can have the heads burned off to allow component disassembly without cutting.

6.1.1.3 Metal Disintegration Machining (MDM)

MDM is similar to the EDM technique discussed above, except that the cutting pulses are generated by vibrating the electrode, rather than current/voltage control. It uses a constant-current power source. A potential difference is established across the gap as the electrode (i.e. tool) is brought close to the workpiece. This causes a very high-energy pulse to be generated just as the tool makes physical contact (unlike EDM) with the workpiece. The principal differences between MDM and EDM are as follows:

- MDM has simpler electronics because of the constant current power supply
- MDM has a lesser degree of control of cut rate and surface finish
- MDM has less wear on tool

- MDM has higher reactionary machining forces because of contact with workpiece
- Overall MDM is faster but less precise than EDM.

The applications of this process are essentially the same as the EDM process e.g. hole boring operations.

6.1.1.4 Arc Saw Cutting (ASC)

The arc saw is a circular, toothless saw blade that cuts any conducting metal without physical contact with the workpiece, eliminating any reaction forces between the two. This means there are no reaction forces between the blade and the workpiece. The cutting action is obtained by maintaining a high current electric arc between the blade and the material being cut while the water (pool or spray) cools the blade and washes out the swarf. The blade, made of any electrical conducting metal such as tool steel, mild steel or copper, rotates at 300-1800 rpm, causing removal of the molten metal created by the arc in the kerf of the cut. The molten metal then condenses in the form of highly oxidised pellets as it is expelled from the kerf. The depth of the cut, up to 0.9 metres, is determined by the blade diameter and the motor drive head diameter.

The arc saw is capable of cutting any electrical conducting material. High conductivity materials (e.g. stainless steel, high alloy steels, aluminium, copper and Inconel) produce the best results. Although carbon steel cuts produce slag build-up in the kerf, which impedes the cutting rate of speed, most materials are cut rapidly and cleanly. Other materials, such as magnesium, titanium and zirconium, will produce hydrogen gas when cut, resulting in the possibility of small, localised ignitions.

The arc saw can be operated under water, or in air with water spray. However, under water is the preferred medium since in-air cutting produces significant amounts of smoke, greater noise and a rougher cut. Cutting in air requires adequate ventilation controls to filter the resultant particulates. Underwater cutting produces a small quantity of steam bubbles, which quickly condense as they rise within the pool.

Cutting speed: Retech, Inc., the original designer and supplier of the arc saw (its patent has run out), claimed its maximum cutting speed on stainless steel under water is 1290 cm2 of cut surface per minute for its large diameter saw. This was accomplished with a 0.9 metre diameter blade, using a 480-volt AC, three-phase, 750 kVA input power supply, and cutting at up to 40 000 amps and 25 volts DC.

For the large-diameter (0.9 metre diameter blade) high-power saw, the cutting speed under water is approximately ten times faster than plasma arc torches rated for the same service, and 100 times faster than known mechanical cutters.

Cutting thick cross-section materials: The arc saw cutting capabilities are limited only by the diameters of the blade and the drive head. With a drive head of approximately 20 cm diameter, and a 0.9 metre blade diameter, the maximum thickness of cut is 35.6 cm. The arc saw is especially suited for cutting stainless steels because they are non-magnetic. For under water cutting, no other tool can perform this depth of cut.

Cutting tool reaction forces: Since the arc saw never touches the workpiece, there are no reaction forces between the two. This means the support system and end-effector (manipulator positioning device) does not have to be built to resist high forces typical of mechanical cutting methods.

Cutting through multiple thicknesses: One of the distinguishing features of the arc saw is its ability to cut through multiple thicknesses of steel in a single pass. As the blade encounters a new workpiece surface, an arc is automatically struck and melting begins. When it passes through the surface, the arc is extinguished at that location but continues at the original location until the cut is completed. The arc current is used as feedback to automatically control the rate of advance into the workpiece.

Swarf diameter: The arc saw was used extensively at the Japanese Atomic Energy Research Institute (JAERI) in segmenting the reactor vessel. The swarf particle size and distribution were recorded as 98% of the total, being greater than 37 micrometres. Most of the swarf was 100 µm in diameter and readily removed from the water by gravity and later vacuuming, while the rest of the fines were collected on a 0.9 µm filter and removed.

This is significantly larger than the swarf from the plasma arc torch, which will make swarf removal from the pool much easier.

Gas generation: The arc saw does not use gases for cutting, and therefore should not generate a rising gaseous plume to carry radioactive particulate to the pool surface. Any steam produced by the saw heat should be rapidly condensed in the cutting region. Hydrogen generation, by disassociation of water, recombines under water. Only when cutting in air is there some minor hydrogen generation, which quickly recombines in air with a crackling sound.

Any steam produced can be captured and vacuumed away by an underwater vacuum system provided to collect particulate generation in the vicinity of the saw blade. In addition, to facilitate viewing through the water surface, an acrylic plastic (Plexiglas) or polycarbonate (Lexan) viewing window can be floated over the arc saw and vacuum suction maintained at the plastic-water interface to ensure particulate is continually removed.

Pool heating: The arc saw cutting power requirements of approximately 20 000 amps at 25 volts DC would likely generate some pool heating. Virtually all of this power goes directly into the arc heating the metal in the kerf. Obviously, some of it will also contribute to pool water heating, just as will occur from the hot chips from mechanical cutting. This heat generation should be far less than the 20 000°C flame of the plasma arc torch.

Blade life: Retech, Inc., has done testing on arc saws and determined that the blade consumption for carbon steel blades on stainless steel workpieces is approximately 6.5 cm2 of blade loss per 71.0 cm2 of work lost. This relatively low rate of blade consumption will permit cutting virtually all day without a blade change. Blade change-out can be accomplished remotely under water to minimise downtime and can be accomplished in less than 30 minutes.

6.1.1.5 Abrasive Water Jet Cutting (AWJC)

A water jet cutter is a tool capable of slicing into metal or other materials using a pure jet of water at high velocity and pressure, or a mixture of water and an abrasive substance. The process is essentially the same as water erosion found in nature but greatly accelerated and concentrated. It is often used during fabrication or manufacture of parts for machinery and other devices. It is the preferred method when the materials being cut are sensitive to the high temperatures generated by other methods. It has found applications in a diverse number of industries from mining to aerospace where it is used for operations such as cutting, shaping, carving, and reaming.

The AWJC technique commonly used in the segmentation of reactor components utilises a ultra high pressure stream of water containing an abrasive media, typically garnet. In this process, a small diameter, high-velocity water jet, and a stream of solid abrasives (garnet) are introduced into a specially shaped abrasive jet nozzle from separate feed ports. A part of the water jet's momentum is transferred to the abrasives, whose velocity rapidly increases. The AWJC system consists of four basic components.

- An ultra high-pressure positive displacement pump referred to as an intensifier pump. This pump pressurizes the water in the range of $2.7x10^3$ – $4.1x10^3$ bars.
- An attenuator located downstream of the intensifier pump, which smoothes the pump induced pressure fluctuations.
- The Cutting Head, where the high-pressure water flow in the range of 3.8–11.4 lpm is forced through a small orifice of approximately 0.05–0.165 cm in diameter emerging as a high velocity jet and exits through a nozzle. Nozzle sizes from 0.508 to 1.65mm internal diameter (ID). The cutting head also includes a mixing chamber where the water mixes with abrasive material from the abrasive delivery system.
- An Abrasive delivery System - the abrasive delivery system in conventional AWJC cutting utilises compressed air to provide an even distribution of the media to the cutting head. The resultant supersonic slurry exits the nozzle at a small standoff distance for the intended cutting surface. More modern systems premix the abrasive with water and are known as Abrasive Water Suspension Cutters.

The kerf can be changed by changing parts in the nozzle, as well as the type and size of abrasive. Typical abrasive cuts are made with a kerf in the range of 1.016–1.27 mm, but can be as narrow as 0.508 mm. Non-abrasive cuts are normally 0.178–0.33 mm, but can be as small as 0.076 mm, which is approximately the width of a human hair. These small jets can make very small detail possible in a wide range of industrial applications.

Commercial waterjet cutting systems are available from manufacturers all over the world, in a range of sizes, and with water pumps capable of a range of pressures. Typical water jet cutting machines have a working envelope as small as 0,1 m^2, or up to ten of m^2. Ultra-high pressure water pumps are available from as low as 276 MPa up to 689 MPa. However, when using water jet cutting methods in a nuclear environment, standard off-the-shelf equipment must be carefully evaluated and modification may be required.

When being considered for the segmentation of reactor components in an underwater environment, several considerations apply:

- When the process is applied to irradiated or contaminated surfaces the resulting slurry, consisting of cut metal particles and abrasives, requires collection and treatment as radioactive waste.
- The abrasive (garnet) is not re-usable.
- Blow-through (the high-pressure water stream that penetrates the material being cut) must be controlled, as to not penetrate critical barriers in the water environment or damage the adjacent equipment.

This technique is adaptable to underwater cutting, although this results in an approximate 30%–40% reduction in the maximum depth of cut that could be achieved in air. This operating efficiency can be improved by an air mantle nozzle which surrounds the cutting water jet with an air stream reducing friction between the cutting jet and surrounding water. As with other material penetration cutting equipment, the cutting head positioning and standoff gap is very important.

6.1.1.6 Mechanical

Mechanical segmentation tooling and processes have been developed by applying the basic techniques used in the conventional machining industry and modifying these techniques such that they are supportive of remote operation and maintenance.

Primarily these techniques are an assortment of material working processes in which power-driven machine tools, such as saws, lathes, milling machines, shears and drill presses, are used to achieve a desired geometry. Machining is part of the manufacture of almost all metal products and substantial experience and lessons learned are available in the non-nuclear practice of metal segmentation and machining as it has been evolving for the past 150 years.

The most traditional mechanical machining processes that have been modified to develop some of the successful segmentation equipment include sawing, milling, and shearing.

Although the terms machining and segmentation without qualification usually implies conventional techniques, there is nothing conventional about the equipment used in the segmentation of reactor components in a remote and most likely underwater environment. Examples of some of the processes used to date in the development of this equipment include:

- Sawing
- Milling
- Shearing

6.1.1.6.1 Sawing

A saw is a tool that uses a hard blade or wire with an abrasive edge to cut through softer materials. The cutting edge of a saw is either a serrated blade or an abrasive. A saw may be worked by hand, or powered by steam, hydraulics, electricity or other power.

In a modern serrated saw, each tooth on a saw blade is bent to a precise angle called its set. The set of teeth is determined by the kind of cut the saw is intended to make. For example, a rip saw has a tooth set that is similar to the angle used on a chisel. The idea is to have the teeth rip or tear the metal apart. Some teeth are usually splayed slightly to each side of the blade so that the kerf is wider than the blade itself and the blade does not bind in the cut.

There are several types of materials used in the blades for cutting reactor components:

- Steel – Used in almost every existing kind of saw blade because it is the lowest cost however the traditional steel blade must be conditioned or coated to provide additional strength and cutting capability.
- Diamond – Fixed into the saw blade, wire or wire bead to form a diamond blade. A diamond is a super hard material and can be used to cut hard and brittle material. They are not especially effective on metal. This type of blade falls into the abrasive saw blade category.
- High Speed Steel (HSS) – The whole saw blade is made of HSS. HSS saw blades are mainly used to cut steel and other types of metals. If high strength steels (stainless steel) are to be cut, the blades must be made of cobalt HSS.
- Tungsten Carbide – Normally there are two ways to use tungsten carbide to make saw blades:
    a) Carbide-tipped saw blades: The saw blades teeth are tipped (via welding) with small pieces of sharp tungsten carbide block. The type of blade is also called tungsten carbide tipped (TCT) saw blade.

b) Solid carbide saw blades: The whole blade is made of tungsten carbide. Comparing with HSS saw blades, solid carbide saw blades have a higher hardness under high temperatures and are more durable.

- An abrasive saw uses an abrasive material embedded into the blade or cutter rather than teeth as would be the case with the serrated saw.

Several saw types that have been used in the segmentation of reactor components include circular saws, bandsaws, reciprocating saws and wire saws.

<u>Circular Saw</u>

The circular saw is a metal disc or blade with saw teeth on the edge as well as the machine that caused the disk to spin. It is a tool for cutting that is either hand-held or fixed to a ridged frame or base. While traditional circular saws are almost exclusively driven by electricity, those used in the segmentation of reactor components are typically driven by hydraulics. As the circular blades strikes the metal, it makes a chip. The teeth guide the chip out of the workpiece, preventing it from binding. As the blade is advanced the cut continues until the desired depth of cut is obtained or until the cut depth reaches the diameter of the hub to which the blade is attached.

Characteristics:

- Cutting is by teeth on the edge of a thin blade
- The cut has a narrow kerf and good finish
- Cuts are typically straight and relatively accurate
- The saw usually leaves burrs on the cut edge

Cold saw blades are circular metal cutting saw blades categorised into two types: solid HSS or TCT. Both types of blades are resharpenable and may be used many times before being discarded. Cold saw blades are used to cut metal using a relatively slow rotational speed, usually less than 5000 SFM (Surface Feet/Minute) (25m/s), and a high chip load per tooth, usually between .001–.003 in. (0.025–0.08mm) per tooth. These blades are driven by a high power motor and high-torque gear reduction unit or an AC vector drive. During the cutting process, the metal is released in a shearing action by the teeth as the blade turns and the feed mechanism moves the blade forward. They are called “cold saw blades” because they transfer all the energy and heat created during the cutting process to the chip. This enables the blade and the work material to remain cold.

The first type of cold saw blade, solid HSS, may be made from either M2 or M35 grade tool steel, alloyed with additional cobalt. Solid HSS saw blades are heat treated and hardened appropriately for ferrous and non-ferrous cutting applications. This high hardness gives the cutting edges of the teeth a high resistance to heat and wear. However, this increased hardness also makes the blades brittle and not very resistant to shock. In order to produce a high quality HSS cold saw blade, you must start with very flat and properly tensioned raw material. The blades must be press quenched after hardening to prevent them from being warped. The term HSS does not necessarily mean what it implies. These blades are usually never run at surface speeds higher than 350 SFM. Solid HSS cold saw blades may be used for cutting many different shapes and types of metal including: tubes, extrusions, structural sections, billets, bars, ingots, castings, forgings etc. These blades may also be coated with special wear resistant coatings such as titanium nitride (TiN) or titanium aluminium nitride (TiAlN).

The second type of cold saw blade, TCT, is made with an alloy steel body and tungsten carbide inserts brazed to the tips of the teeth. These tips are ground on all surfaces to create tangential and radial clearance and provide the proper cutting and clearance angles on the teeth. The alloy body is generally made from a wear resistant material such as a chrome vanadium steel, and heat treated. The TCT blades are capable of operating at much higher temperatures than solid HSS, therefore, TCT saw blades are usually run at much higher surface speeds. This allows carbide-tipped blades to cut at faster rates and still maintain an acceptable chip load per tooth. These blades are commonly used for cutting non-ferrous alloys, but have gained significant popularity for ferrous metal cutting applications in the last 10 years. The tungsten carbide inserts are extremely hard and capable of very long wear life. However, they are less resistant to shock than solid HSS cold saw blades. Any vibration during the cutting process may severely damage the teeth. These cold saw blades need to be driven by a backlash free gear box and a constant feed mechanism like a ball-screw feed.

Band Saw

A bandsaw is a power tool which uses a blade consisting of a continuous band of metal with teeth along one edge to cut various workpieces. The band usually rides on two wheels rotating in the same place, although some small bandsaws have three wheels. The saw motor may by powered by various methods however most band saws used in the nuclear application are powered by electricity or hydraulics.

Bandsawing produces uniform cutting action as a result of evenly distributed tooth load. Bandsaws are particularly useful for cutting irregular or curved shapes, but can also be used to produce straight cuts. The radius of a curve that can be cut on a particular saw is determined by the width of the band and its lateral flexibility. Brushes or brushwheels are sometimes used to remove chips from the blade as it exits the material.

Bandsaws are available in vertical and horizontal designs. Common band speeds range from 40–5,000 ft/min (12–1,500 m/min). Horizontal bandsaws may employ a gravity-fed, hydraulic fed or screw fed blade. Those saws controlled by hydraulics feature a cylinder bleeding through an adjustable valve. When the saw is set up for a cut, the operator raises the saw and the material is clamped in place and/or the saw is attached to the piece to be cut. The blade slowly descends into the material, cutting it as the blade moves.

Advances have been made in bandsaw blades used to cut metals. The development of new tooth geometries and tooth pitches has produced increased production rates and greater blade life.

Reciprocating Saw

A reciprocating saw is a type of saw in which the cutting action is achieved through a push and pull reciprocating motion of the blade.

The term reciprocating saw is commonly assigned to a type of saw used in construction and demolition work. This type of saw has a large blade and handle that can be orientated to allow the saw to be used on vertical surfaces.

The reciprocating sawing method has been used on a large scale in the segmentation of reactor components and other non-reactor components in the decommissioning field for a number of years. The saws were hydraulically driven and attached to the component requiring segmentation and used in both underwater and dry environment.

Wire Saw

A wire saw is a machine using a metal wire or cable for cutting. There are two types of wire saw machines; continuous (or endless or loop) and oscillating (or reciprocating). Sometimes the wire itself is referred to as a “blade”.

The wire can have one strand or many strands braided together. The wire saw uses abrasion to cut. Depending on the application, diamond material may or a mixture of metal powders may be used as an abrasive. A single strand saw can be roughened to be abrasive, abrasive compounds can be bonded to the cable, or diamond impregnated beads (and spacers) can be threaded on the cable. Wire saws are often cooled and lubricated by water or oil.

Diamond wire saws impregnated with diamond dust of various sizes can cut through almost any material that is softer than the diamond abrasive. The technique has been demonstrated to be very effective on concrete; however, it has also been used in some metal cutting applications. Unlike cutting concrete, where the concrete fines clean the diamonds, the performance of diamond wire saws is greatly reduced when cutting metals because the diamond cutting surface tends to fill with the fine metallic material (glazing) and this substantially reduces the abrasiveness of the cutting surfaces. In some instances, drawing the wire through a piece (section) of concrete during the metal cutting operation can increase the life of the cutting wire by providing some cleaning of the metallic material from the cutting surface.

One advantage of wire cutting is the ability to thread the wire into locations that other cutting systems cannot reach and enable greater thicknesses to cut through. Diamond wire cutting is versatile and has been used for cutting openings in containments and biological shields at operating and decommissioning facilities. In principle, the thickness is limited by the fact that the wire needs to be passed around the piece being cut. The drive motor needs to be powerful enough to overcome the resulting friction, which is proportional to the length of the kerf. The loop is made up of lengths of wire assembled for particular operations. The lengths must be about equally worn otherwise the least worn lengths will do all the work and will have a shorter life. The coring unit is mounted with the diamond wire saw to drill so that wire can pass through the holes. This sawing unit moves along the rail laid along the circumference of the cavity by remote control.

Current uses require substantial set-up time that may include coring holes through which diamond wire is threaded. This can limit its use in high radiation or hazardous environment situations. Controls are required for highly contaminated items to reduce the possible spread of contamination due to swarf, which can be carried from the cutting area by the wire. This can result in significant secondary waste when water cooling of the wire is used. To protect the machine from the water some shields are attached to the machine. In order to reduce the contamination by radioactive concrete particles, rotating brushes are installed near the rod[448]. It is also possible to cut in dry conditions when the cutting wire is cooled by local injection of cold compressed air (-10 to -15°C). Dust emissions can be reduced using a sealed collection system located at the outlet of the wire.

Using diamond wire for cutting does have the problem of being less robust (wire snapping when fatigued, bent, jammed or tangling) than solid cutting blades and possibly more dangerous because when the wire brakes it can whip.

Dry cutting of reinforced concrete has been successfully demonstrated and applied at BR3 (Baryte concrete), Rheinsberg, KNK, the CIEMAT PIMIC project and WAK (OECD/NEA, 2011). Dry diamond wire cutting was also performed in the United States to sever hot leg and cold leg nozzles close to the reactor vessel at Connecticut Yankee. HEPA ventilation on the enclosure at the outlet of the wire proved effective in controlling airborne radioactivity even though hot spots were present on the nozzles due to the thermal sleeves. Development of diamond wire end-effectors used for off-shore underwater cutting may have applications in decommissioning that warrant further R&D.

Diamond wire sawing was used at the Rancho Seco NPP in California to cut the steam generators in half, facilitating removal and transport to the disposal site. Accordingly, the level of difficulty and challenge is small compared to reactor vessel and internals segmentation.

When compared to the other nuclear sawing applications, wire saws typically prove to be less expensive but much less effective for cutting reactor components but can be useful in limited application to support an overall reactor component and civil structure segmentation project.

[448] Handbook on decommissioning of nuclear installations. European commission, Luxembourg 1995

6.1.1.6.2 Milling

Milling is a process that uses a machine tool to remove material from solid items. Milling machines are often classified in two basic forms, horizontal and vertical, which refers to the orientation of the main spindle. Both types range in size from small, bench mounted devices to room sized machines. Unlike a drill which penetrates the material vertically, milling can cut by moving the cutter radially as well as specially designed cutters that can cut in both vertically and radially. Cutter movements can be precisely controlled by means of precision ground slides and leadscrews. Milling may be manually operated, mechanically automated, or digitally automated via computer numerical control (CNC).

Milling machines can perform a vast number of operations, from simple (e.g. slot and keyway cutting, planing, drilling) to complex (e.g. contouring, die sinking). Cutting fluid is often pumped to the cutting site to cool and lubricate the cut and to wash away the resulting swarf, however, when using this technology underwater, cooling is unnecessary.

In the vertical mill the spindle axis is vertically orientated. Milling cutters are held in the spindle and rotate on its axis. The spindle can generally be extended allowing plunge cuts and drilling. A horizontal mill has the same sort configuration except the cutters are mounted on a horizontal arbour.

There are many types of cutters to choose from when using milling as a means of segmentation. They come in a variety of shapes and many sizes. There is also a choice of coatings, as well as rake angle and number of cutting surfaces.

Shape: Several standard shapes of milling cutter are used in industry today.

Flutes / teeth: The flutes of the milling bit are the deep helical grooves running up the cutter, while the sharp blade along the edge of the flute is known as the tooth. The tooth cuts the material, and chips of this material are pulled up the flute by the rotation of the cutter. There is almost always one tooth per flute, but some cutters have two teeth per flute. Often, the words flute and tooth are used interchangeably. Milling cutters may have from one to many teeth, with 2, 3 and 4 being most common. Typically, the more teeth a cutter has, the more rapidly it can remove material. So, a 4-tooth cutter can remove material at twice the rate of a 2-tooth cutter.

Helix angle: The flutes of a milling cutter are almost always helical. If the flutes were straight, the whole tooth would impact the material at once, causing vibration and reducing accuracy and surface quality. Setting the flutes at an angle allows the tooth to enter the material gradually, reducing vibration. Typically, finishing cutters have a higher rake angle (tighter helix) to give a better finish.

Centre cutting: Some milling cutters can drill straight down (plunge) through the material, while others cannot. This is because the teeth of some cutters do not go all the way to the centre of the end face. However, these cutters can cut downwards at an angle of 45 degrees or so.

Roughing or Finishing: Different types of cutter are available for cutting away large amounts of material, leaving a poor surface finish (roughing), or removing a smaller amount of material, but leaving a good surface finish (finishing). A roughing cutter may have serrated teeth for breaking the chips of material into smaller pieces. These teeth leave a rough surface behind. A finishing cutter may have a large number (4 or more) teeth for removing material carefully. However, the large number of flutes leaves little room for efficient swarf removal, so they are less appropriate for removing large amounts of material. Roughing type cutters are of greater use during decommissioning segmentation tasks.

Coatings: The right tool coatings can have a great influence on the cutting process by increasing cutting speed and tool life, and improving the surface finish. Polycrystalline Diamond (PCD) is an exceptionally hard coating used on cutters which must withstand high abrasive wear. A PCD coated tool may last up to 100 times longer than an uncoated tool. However the coating cannot be used at temperatures above 600 degrees C, or on ferrous metals. Tools for machining aluminium are sometimes given a coating of titanium aluminium nitride (TiAlN). Aluminium is a relatively sticky metal, and can weld itself to the teeth of tools, causing them to appear blunt. However it tends not to stick to TiAlN, allowing the tool to be used for much longer in aluminium.

Shank: The shank is the cylindrical (non-fluted) part of the tool which is used to hold and locate it in the tool holder. A shank may be perfectly round, and held by friction, or it may have a Weldon Flat, where a grub screw makes contact for increased torque without the tool slipping. The diameter may be different from the diameter of the cutting part of the tool, so that it can be held by a standard tool holder.

Slot Drill

Slot drills are centre-cutting endmills, generally two-(sometimes three- or four-) fluted cutters that are capable of drilling (plunge-cutting) straight down into the material and then moving laterally to cut a slot. The plunge-cutting action is possible because at least one (diametrically opposite) pair of teeth extend all the way to the centre of the end face. Such a feature of endmills is called “centre-cutting”. Slot drills are so named for their use in cutting keyway slots. The term slot drill is usually assumed to mean a two-fluted, flat-bottomed endmill if no other information is given.

End Mills

End mills are those tools which have cutting teeth at one end, as well as on the sides. The words end mill are generally used to refer to flat bottomed cutters, but also include rounded cutters (referred to as ball nosed) and radiused cutters (referred to as bull nose, or torus). They are usually made from HSS or carbide and have one or more flutes. They are the most common tool used in a vertical mill.

Roughing End Mill

Roughing end mills quickly remove large amounts of material. This kind of end mill utilises a wavy tooth form cut on the periphery. These wavy teeth form many successive cutting edges producing many small chips, resulting in a relatively rough surface finish. During cutting, multiple teeth are in contact with the workpiece reducing chatter and vibration. Rapid stock removal with heavy milling cuts is sometimes called hogging. Roughing end mills are also sometimes known as ripping cutters.

Ball Nose Cutter

Ball nose cutters are similar to slot drills, but the end of the cutter is hemispherical. They are ideal for machining 3-dimensional contoured shapes in machining centres, for example in moulds and dies. They are sometimes called ball mills in shop-floor slang, despite the fact that that term also has another meaning. They are also used to add a radius between perpendicular faces to reduce stress concentrations. There is also a term bull nose cutter, which refers more to a cutter having a corner radius less than half the cutter diameter.

Slab mill

Slab mills are used for machining large broad surfaces quickly. In many industrial applications they have been superseded by the use of face mills which cut on the end rather than side of the cutter, but they can still be useful in reactor segmentation tasks.

6.1.1.6.3 Shearing

Shearing, also known as die cutting, is a process which cuts stock without the formation of chips or the use of burning or melting. The most commonly sheared materials are in the form of sheet metal or plates, however rods can also be sheared. Shearing-type operations include: blanking, piercing, roll slitting, and trimming. Various types of portable shears have been used in reactor internals segmentation projects to remove/segment small diameter instrumentation and components. They are hydraulically operated and usually suspended by means of a cable or pulley into the water cavity that houses the component(s).

A punch (or moving blade) is used to push the workpiece against the die (or fixed blade), which is fixed. Usually the clearance between the two is 5–10% of the thickness of the material, but dependent on the material. Clearance is defined as the separation between the blades, measured at the point where the cutting action takes place and perpendicular to the direction of blade movement. It affects the finish of the cut (burr) and the machine's power consumption. This causes the material to experience highly localised shear stresses between the punch and die. The material will then fail when the punch has moved 15–60% the thickness of the material, because the shear stresses are greater than the shear strength of the material and the remainder of the material is torn. Two distinct sections can be seen on a sheared workpiece, the first part being plastic deformation and the second being fractured. Because of normal heterogeneities in materials and inconsistencies in clearance between the punch and die, the shearing action does not occur in a uniform manner. The fracture will begin at the weakest point and progress to the next weakest point until the entire workpiece has been sheared; this is what causes the rough edge. The rough edge can be reduced if the workpiece is clamped from the top with a die cushion. Above a certain pressure the fracture zone can be completely eliminated. However, the sheared edge of the workpiece will usually experience work-hardening and cracking. If the workpiece has too much clearance, then it may experience roll-over or heavy burring.

The process of straight shearing is done on sheet metal, coils, and plates. It uses a guillotine shear.

Various types of metals are used in shear blades:

- Low alloy steel is used for materials that range up to 1/4 in. thick
- High-carbon, high chromium steel is used for materials that also range up to 1/4 in. in thickness

Shock-resistant steel is used for materials that are equal to 1/4 in. thick or more.

6.1.1.7 Oxy-fuel gas cutting

Oxy-fuel gas cutting has been proposed for a number of potential projects in which it was planned to segment the thick steel RPV.

The process has been called various other names, such as, burning or flame cutting. The oxygen-fuel gas flame is the mechanism used to raise the metal workpiece to a suitable temperature for cutting to be performed. The actual cutting operation is performed by a separate oxygen stream.

None of the common fuel gases, such as propane or acetylene, will burn hot enough to heat steels to a sufficient temperature for cutting if they are burned in air. The gases must first be mixed with oxygen and then ignited to provide a sufficiently hot flame. The oxygen which is used to burn the fuel gas is called preheat oxygen, as distinguished from the cutting oxygen, which takes a separate path through the cutting torch and exits through a central hole in the cutting tip.

The fuel gas and two sources of oxygen must therefore be provided through separate ports in the cutting torch to allow independent pressure regulation of each of the gases. The preheat oxygen and fuel gas are normally controlled by adjustable valves and the cutting oxygen is controlled by a lever operated poppet valve.

The operator opens the two preheat valves to allow the preheat oxygen and fuel gas to the cutting tip where they are mixed to form a highly combustible mixture. The gas exits the tip through a ring of fine holes at the front of the tip, where they are ignited by the user. Flame characteristics are adjusted by adjusting the valves. To perform cutting the poppet valve is opened to allow the cutting oxygen to exit the central hole of the cutting tip to produce an extremely hot cutting flame.

Oxy-fuel cutting torches can be used either as hand-held or machine mounted tools though remote deployment would be necessary for the RPV segmentation project.

Oxy-fuel cutters are relatively inexpensive to buy and maintain and can cut some very thick metal sections (over 50cm).

6.1.1.8 Laser cutting

Thanks to the emergence of new high-power industrial sources in the early 2000s, laser technology is now suitable in Decommissioning. A minimum power of 1 kW is required per cm thickness of steel to be cut [449]. With these new multi-kilowatts sources it is possible to cut steel more than 20 centimetres thick.

Their wavelengths, which are close to 1 µm, enable the power of the beam to be transmitted via optical fibre over several dozen or even several hundred metres, from the generator to the 'laser head' placed at the end of a suitable remotely-controlled carrier. The source is thus isolated from the restrictions of the worksites. It is not only reliable, but also provides good energy efficiency, and thousands are produced every year. The ratio between the optical power of the beam and the electric power consumption, which was previously around 1% for the first Nd-Yag sources, is now 30%. Given the optical power of this class 4 equipment, its use is only possible in a closed location with no operator present, thus only in automatic, robotic or remote-controlled mode.

Using laser cutting for dismantling differs from the conventional use of this process for the flow or machining of parts for which high-quality cutting surfaces is required: precision, surface finish and small clearances. The surface finish does not matter for dismantling applications. What is more important is that the dross remains attached to the cut pieces to limit its dispersion and make it easier to remove. However, a high cutting capacity is required and above all a high tolerance with regard to positioning of the head in relation to the area being cut. This simplifies the programming of the trajectories of the carrier or the robot.

The characterisation of the process concerns, first of all, the maximum cutting thickness depending on the material, then the productivity or forward speed which depends on the thickness being cut. Charts have been drawn up, for example for 316L austenitic stainless steel, etc. The cutting capacity for 316L steel is roughly 1 cm/kW for a reduced speed of around 2 cm/min.

Two particularly interesting characteristics of the laser cutting process in air, as used on remote-controlled projects, are the positioning tolerance for the tools and the ability to cut several thicknesses

[449] C. CHAGNOT *et al.*, "Cutting performances with new industrial continuous wave ND:YAG high power lasers. For dismantling of former nuclear workshops, the performances of recently introduced high power continuous wave ND:YAG lasers are assessed." *Nucl. Eng. Design*, (2010), doi:10.1016/j.nucengdes.2010.06.041

simultaneously. It is the propagation of energy in the form of light which provides this ability to cut several thicknesses. The effect of this propagation must be taken into account and controlled by the choice of the power and speeds used. Digital simulation tools and charts can be used to predict the temperature rise of the structures in the background and assess the potential impact of the laser beam on them.

Laser cutting of metal structures in air, in the context of remote-controlled dismantling projects, has reached a satisfactory initial stage of maturity. Specific applications for which the cut cannot pass right through the material, because the parts are too thick or because the material is on a support, are also being studied, such as the case of the fuel debris at Fukushima, deposited on a concrete basemat following the accident. As water is used as the radiological barrier on some dismantling sites, the use of laser cutting in industrial conditions under water is also an important new area for development.

Laser cutting of metal structures in air, in the context of remote-controlled dismantling projects, has reached a satisfactory initial stage of maturity.

Specific applications for which the cut cannot pass right through the material, because the parts are too thick or because the material is on a support, are also being studied, such as the case of the fuel debris at Fukushima, deposited on a concrete basemat following the accident. As water is used as the radiological barrier on some dismantling sites, the use of laser cutting in industrial conditions under water is also an important new area for development.

6.1.1.9 Summary

These cutting processes differ from one another in terms of their performance levels: compatibility with the material, maximum thickness cut, maximum forward speed permitted for a given thickness, and also environmental factors, i.e. the waste they produce such as gaseous effluents, liquid effluents, solid particles, dross and projections. They also differ in their ease or restriction of use.

Mechanical processes transmit contact forces to the equipment which have to be taken up. Tools which penetrate into the cut must be kept in the working axis or working plane and their progress must not be hindered by narrowing of the cutting flanks. Weight, size and accessibility restrictions, such as access to the rear surface, must also be taken into consideration. While considerable thicknesses can be cut by successive passes, the productivity of processes involving the removal of shavings is low. Mechanical processes are characterised by low production of gaseous effluents, aerosols and liquid effluents, including solid particles (except for pressure jet cutting). In contrast, non-contact thermal processes do not require any take-up of forces and the trajectory restrictions are greatly reduced. However, there are often more gaseous effluents and aerosols than with mechanical processes. There can also be a substantial amount of dross and projections. A classification of processes according to quantity of aerosol produced and loss of mass is given the following graphs[450]:

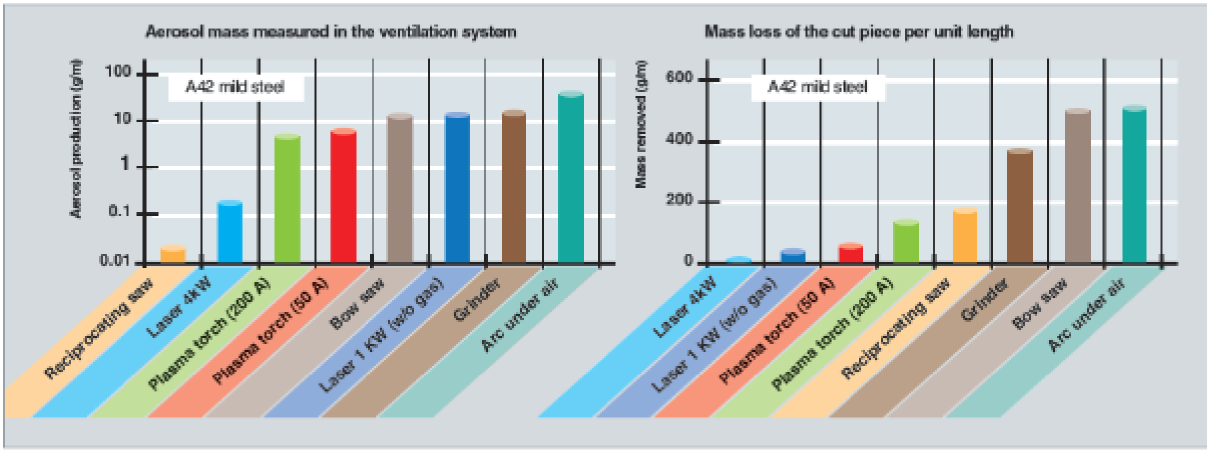


Figure 6.1-1 Aerosol Mass and Mass Loss: comparison between different cutting techniques

[450] G. PILOT "Synthesis of results obtained with laser cutting, a promising Dismantling tool", *Proceedings of ICONE 18*, 2010, Xi'an, China.

Table 6.1-1 Summary of Technologies Identified for segmentation of large irradiated metallic components

Segmentation Technique	What is working	What is missing	Assessment and Possibility for improvement
Plasma Arc Cutting (PAC)	• Fast cutting (e.g. 10mm thick stainless steel at~500+ mm/min) • Adaptable to computer controlled machining • Some flexibility in deployment when using hand-held equipment • Small reaction forces • Cuts most metals up to ~120mm in water, greater thickness in air	• Requires conductive material • Requires large amounts of energy • Creates fine debris that can cause contamination and cleanup problems • High energy increases soluble radioactivity • Requires high degree of filtration • Difficult to maintain torch head standoff • Increased water treatment equipment disposal volume • Off-gases can cause radioactive material to be driven to the surface of the pool	Technological
Electric Discharge Machining (EDM)	Capable of remote handling (under greater water depths)	• Slower cutting (liner cut speed ~1mm/min) • Cutting material is extremely fine which causes contamination and cleanup problems • Increased water treatment equipment disposal volume	Technological
Metal Disintegration Machining (MDM)	Capable of remote handling (under greater water depths)	• Slower cutting (liner cut speed ~1mm/min) • Cutting material is extremely fine which causes contamination and cleanup problems • Increased water treatment equipment disposal volume	Technological
Arc Saw Cutting (ASC)	• Most materials are cut rapidly and cleanly • Can cut thick sections (up to 90cm) • No reaction forces between blade and metal • ability to cut through multiple thicknesses of steel in a single pass	• High energy input • Requires conductive material • Carbon steel cuts produce slag build-up in the kerf	Technological
Abrasive Water Jet	• Medium speed cutting (typical~50mm/min in water) • Suitable for all materials	• High amount of secondary waste and debris due to the abrasive material (~1kg/min operation)	Technological

Segmentation Technique	What is working	What is missing	Assessment and Possibility for improvement
Cutting (AWJC)	• Can cut relatively thick sections – up to 500mm in air, ~300mm in water	• Creates fine debris which can cause cleanup problems • Requires high degree of filtration • Cutting head standoff critical • Increased water treatment equipment disposal volume	
Mechanical Cutting	• Cutting fines are larger and easier to filter • Limited water treatment / filtration required • Suitable for a wide range of material thicknesses	• Slower cutting (typical~10mm/min) • High equipment maintenance • Component shape impacts deployment • Increased segmentation equipment disposal volume	Technological
Oxy-Fuel (also with Powder injection)	• Fast cutting (typical ~100 mm/min at RPV thicknesses) • Can cut thick sections (50cm or greater)	• High energy input • Use of explosive flammable gases • Vaporisation of work piece • Not appropriate for RVIs due to the aggressive nature of the process, the cutting debris it creates and the off gases	Technological
Laser cutting	• Robustness and reliability • Fast cutting • Can cut thick sections (200 cm or greater) • Effortless cutting Easily compatible for remote control with minimised remote carrier • Limited maintenance with no wear parts • Reduced size, mass and umbilical • Limited aerosols generation	• Industrialisation for specific applications for which the cut cannot pass right through the material, because the parts are too thick or because the material is on a support, • use of laser cutting in industrial conditions under water is also an important new area for development.	Find a way to give confidence to operators and regulators

6.1.2 Experiences/Case studies

6.1.2.1 Reactor Vessel Internals Segmentation

In the case of a situation where the Reactor Vessel Internals (RVIs) must be segmented for the disposal there is not one definitive answer for which is the best method of cutting.

Several cutting techniques have been used previously with varying degrees of success. Plasma Arc Cutting (PAC), Electro Discharge Machining (EDM), Metal Disintegration Machining (MDM), Abrasive Water Jet Cutting (AWJC) and mechanical segmentation equipment have all been used to segment RVIs. However, the trend in the last 15 years for RVI segmentation projects has been to adopt mechanical segmentation methods. Furthermore, in the last years, also contractors which previously used or proposed thermal/AWJC techniques are now using mechanical methods (see Table 6.1-2). So, it is reasonable to assume that this reflects the experience and lessons learned from previous projects.

Table 6.1-2 Cutting technologies used in different projects and for different components

Component	Cutting Technique	Project	Year
Steam Dryer (SD)	Plasma Arc (in particular for segmentation of the outer shell and support rings)	Gundremmingen (Germany) [451]	1994-1997
	Reciprocating (Hack) Saw		
	Wire Saw (after concrete filling)	Wuergassen (Germany) [452]	2004
	Disc Saw	Olkiluoto 1 & 2 (Finland)	2008-2011
Steam Separator (SS)	CAMC (to cut of small part at cyclones and defined holes in SS components for safe fastening and handling)	Gundremmingen (Germany) [453]	1994-1997
	Reciprocating (Hack) Saw (for cutting-off the separator cyclone)		
	Plasma Arc (for opening the connecting bars and segmenting stand pipes)		

[451] Alba H.H., Eickelpasch N., Schmidt D., Steiner H., Innovative Underwater Cutting Procedures for the Dismantling of Two German Nuclear Power Plants. WM'99 Conference. 1999.

[452] Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.

[453] Alba H.H., Eickelpasch N., Schmidt D., Steiner H., Innovative Underwater Cutting Procedures for the Dismantling of Two German Nuclear Power Plants. WM'99 Conference. 1999.

Component	Cutting Technique	Project	Year
	Band Saw	Olkiluoto 1 & 2 (Finland)	2004-2006
	Compass saw / Band Saw	Wuergassen (Germany) [454]	2005-2006
Feedwater Sparger (FWS)	Band Saw (segmenting after separation from vessel)	Gundremmingen (Germany)	1994-1997
	Reciprocating (Hack) Saw (to separate FWS from the vessel)		
Core Spray System (CSS)	Different Shearing Tools	Forsmark 1, 2 and 3 (Sweden) Oskarshamn 1 & 2 (Sweden)	2003-2004
	EDM (to separate CSS from the CSC)		
	Band Saw		
Core Shroud Cover (CSC)	Band Saw	Oskarshamn 1 & 2 (Sweden)	2003-2004
	Tube Cutting Tool (to cut stand pipes on the CSC)		
	Disc Saw (to cut CSC support beams)	Olkiluoto 1 & 2 (Finland)	2004-2006
	Abrasive Water Suspension Jet Cutting (AWSJC)	Wuergassen (Germany) [455]	2007
Core Shroud (CS)	Band Saw	Forsmark 1 & 2 (Sweden) [456]	2000-2001
	MDM tool (turning holes for the band saw)		
	Band Saw	Wuergassen (Germany) [457]	2007
Upper Core Grid (UCG)	Abrasive Water Suspension Jet Cutting (AWSJC)	Wuergassen (Germany) [458, 459]	2007

[454] Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.
[455] Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.
[456] Larsson H., Anunti A., Edelborg M., Decommissioning Study of Oskarshamn NPP. 2013
[457] Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.
[458] Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.
[459] Kleimann J., Water Abrasive Suspension (WAS) Cutting Under Water in Decommissioning Nuclear Power Plants. 2009 American WJTA Conference and Expo. Houston, Texas. 2009.

Component	Cutting Technique	Project	Year
Lower Core Grid (LCG) (Core Support Grid)	Remotely Controlled Hydraulic Shearing Tool	Forsmark 1 & 2 (Sweden)	2000-2001
	Band Saw (outer ring)		
	Plate Shearing Tool	Oskarshamn 1 & 2 (Sweden)	2003-2004
	Band Saw		
	Abrasive Water Suspension Jet Cutting (AWSJC)	Wuergassen (Germany) [460, 461]	2007
Control Rods Guide Tubes (CRGT)	Band Saw and Nibbler	Wuergassen (Germany) [462]	2004
Drive Shaft Shroud Tubes, Upper Support Plate, Deep Beam Sections, Upper Support Columns, Plenum Cylinder, Guide Tubes Assembly, Upper Core Plate, etc…	Diamond Wire Saw	Rancho Seco (California) [463]	2004-2005
	Reciprocating Saw for Upper core plate		
	Pole Saw to remove Guide Tubes		
	PAC to remove shell sections of the Plenum Assembly		
	Disc saw		
	Different Disc Cutting Tools and Shearing Tools	Josè Cabrera (Spain) [464]	2012-2013
Upper Core Barrel (UCB)	Band Saw	Josè Cabrera (Spain) [465]	2012-2013
Baffle Plates, Former Plates	Band Saw	Josè Cabrera (Spain) [466]	2013

[460] Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.
[461] Kleimann J., Water Abrasive Suspension (WAS) Cutting Under Water in Decommissioning Nuclear Power Plants. 2009 American WJTA Conference and Expo. Houston, Texas. 2009.
[462] Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.
[463] Anderson M.G., Fennema J.A., Mechanical Cutting of Irradiated Reactor Internal Components. WM'07 Conference. Tucson, Arizona. 2007.
[464] Segerud P., Westinghouse segmentation – José Cabrera RVI Segmentation, January 21, 2014, Stockholm, Sweden.
[465] Segerud P., Westinghouse segmentation – José Cabrera RVI Segmentation, January 21, 2014, Stockholm, Sweden.
[466] Segerud P., Westinghouse segmentation – José Cabrera RVI Segmentation, January 21, 2014, Stockholm, Sweden.

Component	Cutting Technique	Project	Year
Lower Core Barrel (or Core Barrel)	Disc Saw (vertical cuts)	Rancho Seco (California)[467]	2004-2005
	Milling Machine (horizontal cuts)		2004-2005
	Band Saw	Josè Cabrera (Spain) [468]	2013
Thermal Shield (TS)	Disc Saw (vertical cuts)	Rancho Seco (California)[469]	2004-2005
	Milling Machine (horizontal cuts)		2004-2005
	PAC (for the Upper Cylinder Wall, 70mm thick)	Karlsruhe (Germany) [470] Research Reactor	2005
	CAMC (for the Lower part, up to 130mm thick)		
	Disc Cutting Tool	Josè Cabrera (Spain) [471]	2013
Lower Core Plate, Core Support Columns, Intermediate Diffuser Plate, Lower Support Plate, etc…	Reciprocating Saw and Disc Saw	Rancho Seco (California) [472]	2004-2005
	Band Saw and Disc Saw	Josè Cabrera (Spain) [473]	2013

The major advantage of mechanical segmentation is that the form and consistency of the secondary wastes generated make it very easy to collect and manage them in an underwater environment and they do not usually cause water clarity issues. Typically, the material generated in the segmentation process is metal chips that can be mechanically separated from the water without sophisticated water treatment systems.

Disadvantages of mechanical segmentation include the fact that the cutting speed is generally slower than other techniques. As most of the equipment is specially designed for a particular project or component,

467 Anderson M.G., Fennema J.A., Mechanical Cutting of Irradiated Reactor Internal Components. WM'07 Conference. Tucson, Arizona. 2007.
468 Segerud P., Westinghouse segmentation – José Cabrera RVI Segmentation, January 21, 2014, Stockholm, Sweden.
469 Anderson M.G., Fennema J.A., Mechanical Cutting of Irradiated Reactor Internal Components. WM'07 Conference. Tucson, Arizona. 2007.
470 Loeb A., Eisenmann B., Prechtl E., Research Reactor MZFR, Karlsruhe, Germany - Under Water Thermal Cutting of the Moderator Vessel and of the Thermal Shield.
471 Segerud P., Westinghouse segmentation – José Cabrera RVI Segmentation, January 21, 2014, Stockholm, Sweden.
472 Anderson M.G., Fennema J.A., Mechanical Cutting of Irradiated Reactor Internal Components. WM'07 Conference. Tucson, Arizona. 2007.
473 Segerud P., Westinghouse segmentation – José Cabrera RVI Segmentation, January 21, 2014, Stockholm, Sweden.

equipment breakdowns can cause significant delays. Routine maintenance must be incorporated into the segmentation schedules to accommodate frequent replacements of the cutting blades. Finally, mechanical cutting methods generally require more time to test and deploy.

In the case of Reactor Vessel segmentation, the main techniques used in the last years are AWJC, mechanical techniques and oxy-fuel cutting.

With respect to the AWJC it could be stated that it is not as fast as Plasma Arc Cutting method but requires less energy and it presents lower dose levels for workers. However, the main disadvantage of AWJC is the secondary waste created by the addition of the abrasive grit material needed in the water stream to cut the metal.

With respect to oxy-fuel cutters it could be stated that they are cheap to buy and maintain and can cut some very thick metal sections (over 50 cm). They are also able to cope with significant variations in both the thickness of the material to be cut and the distance between the work-piece and the tool tip.

However, oxy-fuel cutters give off a large amount of heat. This can cause heat stress to the operator and workers in the area if they are operated for a prolonged period. The high heat input to the work piece can also cause problems. The width of the cut made by an oxy-fuel cutter can be around 2-3 cm wide. The material removed by a cut of this width is either vaporised or melted. The vaporised material causes an airborne hazard (possibly radiological depending on the cut item) and can cause ventilation system filters to block very quickly, resulting in additional secondary waste. The melted material re-solidifies on surfaces behind or below the cut where it can cause problems for subsequent cutting and removal operations.

So, it is possible to state that for RPV segmentation the situation is less clear compared to that of the internals: the number of completed projects is small and all three main feasible techniques have all been used successfully. The selection of any particular technique used for segmentation is likely to be based on a combination of what can be justified as suitable with regulatory authorities and contractor experience of the particular technique.

6.1.2.2 Experiences in segmentation: JOSÈ CABRERA NPP (SPAIN) [474], [475], [476]

- Reactor type: 1-Loop PWR Westinghouse
- Electric power: 160 MW e
- Operations: 1969-2006
- Operator: Union Fenosa SA
- Decommissioning: 2010 - 2016

[474] Josept Boucau, Gonzalo Medinilla, Westinghouse Experience in Reactor Vessel Dismantling Projects, October 18, 2018.

[475] Josept Boucau, et.al., Best Practices for Preparing Vessel Internals Segmentation Projects, PREDEC 2016, February 16-18, Lyon, France.

[476] Segerud P., Westinghouse segmentation – José Cabrera RVI Segmentation, January 21, 2014, Stockholm, Sweden.

The dismantling activities of the internals were carried out by Westinghouse, as main contractor from September 2010 to November 2013.

The internals were segmented underwater in the fuel pool, inside the reactor building, through the use of mechanical cutting systems whose reliability had already been tested by Westinghouse in previous similar experiences.

In addition to the verification and adaptation of existing plant systems, the implementation of the activities required the design and implementation of some preliminary activities necessary to allow the cutting of the components inside the fuel pool.

These preparatory activities have been very expensive, and in general have taken longer than expected.

With the reactor cavity and the fuel pool flooded, first the upper internals and then the lower internals were removed from the vessel, transported to the fuel pool through the enlarged passage channel and then segmented.

The disk was used for cutting the plates (top support plate, upper and lower core plate) and for the control rods tubes and the thermal shield.

A band saw was used to cut the upper barrel and the core barrel, on a vertical central pillar positioned inside the lower internals.

Cutting and packaging of the internals lasted just over a year, about 2600 hours of total work. Considering about 60 t, the cutting operations led to a total of 432 spools for a total of 418 m of linear cut.

The use of mechanical cutting systems, mainly disc and band saw, under the water head has proven, as anticipated, to be very reliable and has avoided aerosol and visibility problems characteristic of thermal cutting systems. However, the production of the cutting residues was greater than expected and the cleaning of the fuel pool required the use of additional filtering systems to remove the residues.

Also the dismantling of the vessel was made by Westinghouse as main contractor, from June 2013 to April 2015. The vessel decommissioning has required the following:

- Dry cutting of the nozzles, through the use of a diamond wire saw
- Elevation of the vessel by means of a suitably built hydraulic system, and dry handling of the vessel through the fuel passage channel
- Positioning of the vessel on a suitable support stand in the fuel pool
- Segmentation of the vessel, under the water head, in the fuel pool through a band saw, with a central pillar, positioned inside the vessel.

The segmentation of the vessel, approximately 114t, involved the execution of 240 m of cutting and the production of 140 spools.

Overall, the cutting and packaging operations began in June 2014 and were completed in 10 months.

6.1.2.3 Experiences in segmentation: STADE NPP (GERMANY) [477, 478, 479, 480]

- Reactor type: 4-Loop PWR
- Electric power: 630 MWe
- Operations: 1972-2003
- RVI and RPV decommissioning: 2007 - 2010

The plant is owned by E.On Kernkraft GmbH (two thirds) and Vattenfall Europe AG (the remaining third). Also in this case, the dismantling activities of the vessel and its internals have been divided into two separate contracts, starting from 2007.

In January 2007 AREVA was awarded the contract for the dismantling of internals. The activities were completed in August 2009.

The cutting strategy involved segmentation operations under the water head, using both the reactor cavity and the fuel pool. With this in mind, it was necessary to carry out a series of preliminary activities, preparatory to the subsequent cutting and packaging phases.

The segmentation of the internals was carried out under water head, flooding both the reactor cavity and the fuel pool. In particular, once removed from the vessel, the upper internals and lower internals were transferred to special supports, previously positioned respectively in the fuel pool (inside the Reactor Building) and in the reactor cavity.

The cutting systems mainly used are mechanical ones and in particular the chop saw (for control rods guide tubes, control rods drive mechanism, support columns and baffle plates), the band saw (for upper barrel and core barrel) and cutters of different types (both disc and pointed) to remove the connections between some components of the lower and upper internals.

EDM (Electro Discharge Machining) technology was used for specific applications, in particular for the removal of the circumferential screws present outside the Core Barrel, which were intended to keep the Former Plates anchored inside the Core Barrel.

The other technology used during the segmentation of the internals was abrasive water suspension jet cutting (AWSJC), performed inside a confined area ("water tank") in the fuel pool. This technology was used for cutting the plates and / or grids of both the upper and lower internals (top support plate, upper grid, upper and lower core grid).

---

477 Andreas Loeb, Dieter Stanke, Decommissioning of the reactor pressure vessel and its peripheral facilities of the Nuclear Power Plant in Stade, Germany, WM2011 Conference, February 27 – March 3, 2011, Phoenix, Arizona, USA.

478 Bruhn, J.H., AREVA: Experience in dismantling and packing of pressure vessel and core internals, BULATOM International Nuclear Forum on Nuclear Energy – challenges and prospects – June 9-11, 2010, Varna, Bulgaria.

479 Hans-Otto Rohwer, AREVA Experience in Dismantling of the Primary Circuit, ENKO 2014, April 23–24, 2014, Bratislava, Slovak Republic.

480 Annette Bender, Nee Schmitz, Dismantling of RPV Internals - Experience within AREVA NP, 2010

At the end of the operations on the internals, the cavity was drained and all the operations necessary to free, clean and decontaminate the reactor cavity and the fuel pool were carried out.

In June 2008 a consortium formed by Siempelkamp NIS Ingenieurgesellschaft GmbH (as a leading company) and E.ON Anlagenservice GmbH (EAS) was awarded the contract for the dismantling of Vessel and its peripheral systems. The activities were completed in November 2010.

The dismantling of the RPV and its peripheral systems was carried out completely dry (with the reactor cavity and the pool drained) and can be divided into the following main phases:

- Removal and packaging of the "shield compartments" surrounding the upper part (flange) of the vessel;
- Removal of the 8 pipes of the primary cooling circuit and their packaging in containers for final disposal;
- Removal and packaging of the thermal insulation of the RPV consisting of a steel sheet containing aluminium sheets;
- Removal and packaging of the "flood container" (Neutron Shield Tank) between RPV and biological shield;
- Segmentation and packaging of the RPV in containers suitable for final disposal (according to Konrad's waste acceptance criteria).

For the phases relating to the removal of the peripheral systems and surrounding the RPV, mechanical (e.g. pipe disconnector and cutters) and thermal (oxyfuel) cutting technologies were used.

Concerning the segmentation of both the flange and the cylindrical part of the RPV, oxy-fuel technology (thermal break) was used, through the use of an oxy-propane torch.

The Vessel of the Stade NPP was made of steel with high mechanical resistance, internally coated with a layer of chromium-nickel stainless steel (called "cladding" or "plating").

The main characteristics of the Vessel are shown below (the Vessel head is excluded from the following data) :

- External diameter = 4.700 mm;
- Overall Height = 7.942 mm;
- Maximum Thickness (Flange) = 478 mm;
- Wall thickness (cylindrical part) = 199 mm;
- Cladding thickness = 7 mm;
- Total mass = 209 t

After separating the vessel from the primary circuit pipes and after removing the systems surrounding the upper part of the vessel, the flange was separated from the cylindrical part. This separation, performed by means of an oxy-propane torch, allowed the polar crane to lift the RPV cylindrical part (the assembly consisting of flange and cylindrical part would have weighed more than the capacity of the polar crane) and to position it on the turntable, inside the fuel pool. As mentioned above, these operations were performed in dry environment.

Once the Vessel was positioned on the turntable, the thermal insulation around the vessel was removed, cut and packaged, using both the main and auxiliary 6-axis manipulator.

Then, the segmentation of the cylindrical part of the RPV began with the oxy-propane torch, proceeding from the top down. The segments obtained by segmentation were packaged, during the operations, in shielded containers (prismatic in the case of LLW or cylindrical in the case of waste classified as ILW and deriving from the core region) in communication with the work environment through a system of shielding hatches obtained in an area straddling the reactor cavity and the fuel pool and indicated with the name of "packaging station".

The segmentation of the RPV ended with the cutting of the flange which had been positioned in a temporary storage location. For this purpose, a high quality cutter was used to create V-notches in the internal part of stainless steel (cladding) before proceeding with the thermal cut by means of an oxy-propane torch.

6.1.2.4 Experiences in segmentation: CHOOZ A NPP (FRANCE) [481]

- Reactor type: 4-loop PWR built in a cave inside a hill
- Electric power: 305 MWe
- Operations: 1967-1991
- Decommissioning: in progress

In 2010 EDF commissioned Westinghouse in consortium with Nuvia France to dismantle the reactor.

For the internals, the project foresees their segmentation under water in the reactor cavity using mechanical cutting systems, remotely controlled, as done in the previous Jose Cabrera experience.

Initially the upper internals will be segmented. The upper internals comprise different plates (instrumentation plate, guide tube support casting and the upper core plate), 52 guide tubes and 4 support columns that connect the two plates. Control rod guide tube extensions and CRDMs are above the guides tubes. Once the upper package is removed from the vessel and positioned in the reactor cavity, the following actions will be carried out:

[481] Boucau J., Mirabella C., Nilsson L., Kreitman P.J., Obert E., Chooz A, First Pressurized Water Reactor to be Dismantled in France. WM2013 Conference. Phoenix, Arizona, USA. 2013.

- The CRDM and some components connected to the instrumentation plate will be cut by means of hydraulic shears.
- The extension tubes, the support columns and the guide tubes will be cut with the disc saw.
- the upper core plate, the guide tube support casting and the instrumentation plate will be cut with the band saw.

Then, the lower internals will be removed and segmented. In particular, their segmentation will be performed:

- by means of a band saw attached to a column positioned in the middle of the lower internals, upper barrel, core barrel, baffle plates and former plates will be segmented. The cutting strategy foresees making a series of vertical cuts, then rotating the blade 90 degrees and making horizontal cuts to remove the single spools;
- by means of a disc saw the "shroud tubes" will be segmented;
- by means of a band saw the lower support casting, the core radial supports, the lower core plate and the last remaining components will be segmented.

The Chooz A vessel weighs about 177 tonnes (head excluded) and is made of carbon steel with an inner stainless-steel cladding. The RPV is externally covered with thermal insulation made of rock wool.

The first step will consist in cutting the 8 nozzles using an orbital cutting tool inserted inside them.

Then the vessel will be extracted from the reactor pit through the polar crane and positioned on a support above the protective liner of the reactor cavity.

The thermal insulation, probably vitrified because of irradiation, will be removed by cutting with a disc saw.

At the end the segmentation of the vessel will be carried out, underwater, from top down with a band saw attached to a column positioned inside the vessel.

6.1.2.5 CEA UP1 Reprocessing plant Marcoule (France)

UP1 dissolvers are very thick and hard equipment ("Uranus" stainless steel), with a 'blind' cell subject to levels of irradiation up to 1 Gray/h, thanks to previous decontamination, and components with complex shapes (tanks, tube circuits, neutron meters, etc.).

CEA choose a laser process in its scenario because it produces lower aerosol emissions than most of the other thermal cutting processes, limits the production of dross (mass loss in cut parts) and, enables a high tolerance of tool positioning and allows simultaneously cutting several thicknesses, which is the case with UP1 dissolvers.

ONET Technologies® was selected by CEA to design, build, and operate all of the equipment specific to dismantling and to remove/condition the resulting waste. This project took place over a period of seven years (2011-2018) and represents 15,000 hours of engineering work and produced 11 tons of primary waste and 60 tons of secondary waste.

The remote-controlled system includes a telescopic mast seven meters range, with a MAESTRO manipulator arm interfaced with specific tools: laser torch developed by CEA, disc cutter, nuclear measurements, gripper, and a projection and suction. The laser torch did its works and the disc cutter, initially planned as back-up solution, was not used.

A TruDisk 6 kW laser source was used. The source, its cooling unit and the station for controlling the source and the remote-controlled functions are all incorporated in a transportable shelter. The laser beam is transported from the source to the entrance of the worksite cell via an initial 100 m long, 400 μm core diameter optical fibre. This fibre feeds a coupler which reinjects the beam into a second 30 m long, 600 μm core diameter fibre. The second fibre enters the cell and ends at the laser head.

The laser device also has a 5-bar dry, dust-free compressed air supply and two cooling units: one for the source and the other for the optical fibre coupler. The laser head and its fibre connector are supplied by the compressed air source. This air is used as an assisting gas to expel the molten material in the kerf and as a cooling gas to offset the temperature rise of the optical components caused by the intense flux passing through them. The optical power losses in the fibres, the coupler and the laser head are estimated to be a few percent of the power transmitted, i.e. a few Watts.

The head developed by the CEA for the UP1 project and used by the industrial operator is a robust, air-cooled head for operation up to 14 kW. As it has no retention area, it is also easy to decontaminate. It is characterised by an F1 focal length of 85 mm and an F2 focal length of 255 mm. The waist diameter is 3 times the diameter of the fibre, i.e. 1.8 mm for a 600 Hm fibre.

The F2 focal length of 255 mm is a compromise between the size constraints, the dispersion of the energy beyond the impact point and a long Rayleigh length which gives the process a good positioning tolerance of around 50 mm for cutting in air. The Rayleigh length characterises the distance at which the area of the beam is doubled in relation to the minimum area obtained at the waist.

6.1.2.6 BELGIAN REACTOR N.3 (BR3) [482, 483]

- Reactor type: 2-loop PWR
- Electric power: 10,5 MWe
- Operations: 1962-1987
- RVIs and RPV Decommissioning: 1989 – 2000

The BR3 reactor was the first pressurized water reactor (PWR) installed and operated in Europe. While its rated power level was low (40 MWth, 10.5 MWe net), it contained all the features of commercial PWR

[482] J. Dadoumont, V. Massaut, M. Klein, Y. Demeulemeester. "Decommissioning of a small reactor (BR3 reactor, Belgium)"
[483] https://science.sckcen.be/en/Facilities/BR3

power plants. The reactor was used at the beginning of its lifetime as a training facility for future NPP operators.

The reactor was shut down in 1987 after 25 years of operation.

The shutdown of BR3 and subsequent dismantling, for which various strategies were feasible, was an opportunity to conduct a demonstration experiment for the "European nuclear community".

In 1989, BR3 was selected by the European Union as one of four pilot dismantling projects, included in the third EU five-year research programme on decommissioning of nuclear installations. The project started in 1989. The first part of the pilot project (1989-1994) involved the decontamination of the primary loop and the dismantling of all the highly radioactive reactor internals. In 1994, an extension of the contract was signed with the European Union, covering the dismantling of the first set of reactor internal components, which were removed from the reactor many years before. The main goal of this contract was to allow the comparison of an immediate dismantling operation with a deferred operation after a 30-year cooling period.

After gaining a great deal of experience in remote dismantling of highly radioactive components during the dismantling of the two sets of internals (for this purpose, SCK•CEN tested a number of cutting techniques such as band saw, circular cutter, spark erosion and plasma torch), in 1996, it was decided to carry on with the dismantling of the BR3 reactor pressure vessel (RPV).

The dismantling of the RPV was also part of a European contract.

The first technical acts of this important project were executed at the end of 1997. During the feasibility phase of the RPV dismantling, a decision was made to cut it under water in the refuelling pool of the plant, after having removed it from its cavity. In 1999, the 29 tons heavy reactor vessel was pulled out of the BR3 in order to be decommissioned. The RPV was cut into segments using a milling cutter and a bandsaw machine. These mechanical techniques have shown their ability for this kind of operations. In Summer 2000, the last cut on the RPV was carried out and the cut pieces were transferred to the storage facility.

Prior to the segmentation, the thermal insulation situated around the RPV was remotely removed and disposed of.

The BR3 dismantling project carried out direct comparison tests on the available cutting processes, sometime using different techniques on identical work pieces. For example, the reactor internals had been replaced during the operational life of the plant. This allowed for each set to be segmented in a different manner. Comparison testing was also carried out on the Reactor Vessel segmentation, with a mix of band and circular saws being used to determine a preference.

The use of mechanical cutting techniques emerged as the preferred methodology.

6.2 Handling, segregation and loading of segmented elements and secondary waste

In the dismantling process of a nuclear facility, the produced waste has to be conditioned and treated in order to be able to reuse or recycle as much of it as possible. In between the waste processing steps, the material has to be transported to the various treatment locations. All waste that is kept in final storage has to be packed in to waste packages, that fulfil the acceptance criteria of the final disposal site.

The processing of the waste is not a simple task due to the great variety and huge mass of materials. For example, the mass of waste produced by the dismantling of two German nuclear power plants Würgassen (BWR) and Stade (PWR) is given in Table 6.2-1.

Table 6.2-1 Material from controlled areas for Würgassen and Stade

	Released (t)	**Controlled Recycling (t)**	**Radioactive Waste (t)**
Würgassen (BWR)	255,000	3000	4600
Stade (PWR)	124,000	500	3000

Table 6.2-1 shows that there is a significant mass of waste that can be released or recycled, there for segmentation and conditioning are key aspects of the waste treatment process. For the processing of radioactive waste to packages proper conditioning, qualification and safe storage are key points, since the reconditioning is very costly [484].

6.2.1 Handling / Transport

There is perception that the transportation of waste is a problematic task. In some cases, this is true, for example: spent nuclear fuel or high activity liquid waste but in general not for solid waste.

The transportation distance of the radioactive waste differs a lot for the different waste route options (see Section 5.2.1).

The majority of waste that cannot be released is classified as VLLW and LLW. This waste forms can be transported in conventional containers and vehicles. A standard IP2 sea container meets the requirements of transport regulations. In the case of intermediate-level waste, a special container or a shielding inside the container may be required to not exceed the maximum dose rate allowance. In case of components that are too big to fit an IP2 container, segmentation prior to transportation has to take place or special transport concepts have to be applied. Components without surface contamination can often be shipped

[484] Methodology to Manage Material and Waste from Nuclear Decommissioning; Waste Management & Decommissioning Working Group, World Nuclear Association, Produced by: World Nuclear Association, Published: February 2019, Report No. 2019/001

without outer packaging, whereas components with outer face contamination, for example, Turbine rotors need some over packing [485].

In the case of onsite transportation and handling of the lower waste categories, standard (transparent) plastic bags and cardboard boxes have been proven a good choice. Sufficient labelling and declaration has to be kept in mind [486].

6.2.2 Segregation and conditioning

Segregation and conditioning are the two fundamental points in predisposal waste processing since these processes mainly reduce the waste that is sent to the final storage facilities.

6.2.2.1 Segregation

After the production of the waste, it is collected and the first segregation takes place. The primary goal is to separate the waste into active and non-active waste streams to minimise the waste that has to be further treated or disposed of. In turn, the active waste stream has to be separated according to:

- Long and short-lived
- Contaminated or activated
- Possible reuse after treatment/decontamination
- According to the subsequent treatment

Thereafter, the materials that have limited potential for reuse after treatment are segregated into fractions to reduce the volume. It should be noted, that there are waste categories for materials that are not processable for any financial, technical, safety or legislative reasons [487], [488], [489]. In the case of liquid waste streams, the mixing should be limited to those streams that are radiologically and chemically compatible.

The technical solutions for the sorting, ranging from manual sorting boxes to complex sorting lines uses remote-controlled tools. Air classifiers are used to sort out plastics and shredders, etc.

6.2.2.2 Experiences/Case studies

To illustrate the segregation processes during the dismantling process, here are some example of how the segregation of waste is done in different facilities:

---

485 Methodology to Manage Material and Waste from Nuclear Decommissioning; Waste Management & Decommissioning Working Group, World Nuclear Association, Produced by: World Nuclear Association, Published: February 2019, Report No. 2019/001

486 TECHNICAL REPORTS SERIES No. 402; HANDLING AND PROCESSINGOF RADIOACTIVE WASTE FROMNUCLEAR APPLICATIONS; INTERNATIONAL ATOMIC ENERGY AGENCYVIENNA, 2001: ISBN 92–0–100801–5

487 Methodology to Manage Material and Waste from Nuclear Decommissioning; Waste Management & Decommissioning Working Group, World Nuclear Association, Produced by: World Nuclear Association, Published: February 2019, Report No. 2019/001

488 IAEA-TECDOC-1817, SELECTION OF TECHNICAL SOLUTIONS, FOR THE MANAGEMENT OF RADIOACTIVE WASTE, INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 2017; ISBN 978–92–0–104717–5

489 IAEA-TECDOC-1815; USE OF THE BENCHMARKING SYSTEM; FOR OPERATIONAL WASTE; FROM WWER REACTORS; INTERNATIONAL ATOMIC ENERGY AGENCY; VIENNA, 2017; ISBN 978–92–0–104617–8

Manual Sorting Box, Shredder and Decontamination Tank with Ultrasonic Agitation for Horia Hulubei National Institute for R&D in Physics and Nuclear Engineering in Romania

The segregations unit at Institute of Physics and Nuclear Engineering in Magurele/ Bucharest includes a shredder to reduce the size of slightly contaminated plastic waste, a decontamination tank with ultrasonic agitation and a sorting box, to sort low-level waste, equipped with a tilting device for 100 l drums and a docking station for 200 l drums. Figure 6.2-1 shows the sorting box [490].

Figure 6.2-1 Sorting box of Horia Hulubei National Institute for R&D in Physics and Nuclear Engineering in Romania.

Sorting and Fragmentation Facility for Waste Treatment Centre at Khmelnitsky NPP in Ukraine

Nukem Technologies is assigned to build a sorting and fragmentation center at the Khmelnitsky NPP. The treatment center is designed to process all types of low and intermediate level solid waste generated at Khmelnitsky NPP. Further conditioning includes thermal and mechanical treatment to decrease the waste volume for final disposal.

The unit holds a container delivery division, the pre-sorting, fragmentation, measurement and the sorting box (see Figure 6.2-2). The waste is sorted according to its further conditioning („metal waste", „combustible/non-combustible waste", „compactable/non-compactable waste", „free released waste") [491].

---

[490] https://www.nukemtechnologies.de/en/projects/ro/manual-sorting-box-shredder-and-decontamination-tank-with-ultrasonic-agitation visited 13.07.2020

[491] https://www.nukemtechnologies.de/en/projects/ua/Sorting%20and%20Fragmentation%20Facility%20for%20Waste%20Treatment%20Centre%20at%20Khmelnitsky%20NPP%20in%20Ukraine visited 13.07.2020

Figure 6.2-2 Part of the Sorting and Fragmentation Facility for Waste Treatment Centre at Khmelnitsky NPP in Ukraine.

6.2.2.3 Conditioning

After the segregation of the waste into different categories, one or more conditioning steps may follow to reduce the waste volume or use fragmentation techniques to fit the material into waste package containers or prepare it for further treatment.

Possible conditioning options to reduce the volume are:

- Thermally treatable
  - E.g. for organic material
- Melting of metals to form ingots
  - Better measurement and predictable long term behaviour
- Compaction
  - For Materials as insulation, inorganics, metals or organics (if thermal treatment or melting is not possible). The Energy consumption for compaction has to be taken into the mind.

Options for fragmentation include:

- Including shredding, granulation or grinding
- Cutters with high-temperature flames
- Various sawing methods
- Hydraulic Shearing
- Abrasive cutting
- Plasma arc cutting

The possibility of spreading contamination has to be kept in mind when using a fragmentation technique [492], [493].

6.2.2.4 Experiences/case studies

There are many different compaction methods available for volume reduction on the conventional and specially designed methods for radioactive waste. On this point, particularly for nuclear applications developed vacuum compaction and In-drum compaction is introduced [494].

In-drum compaction

A standard application for In-drum compaction is volume reduction for waste that is shipped to interim storage or treatment facilities. This machinery is commonly found at waste generating sites, to reduce the material volume inside steel drums, which are used for waste storage and transport. The applied pressure rises up to 100 Tons and therefore is a low-pressure technique. A standard unit is shown in Figure 6.2-3.

Figure 6.2-3 In-drum compactor

The volume reduction factor is generally between three and five. The waste is mostly fed into the compactor by hand. Therefore, this machinery is limited to lower dose rates. The size of compaction material is limited to material fitting in the drum [495].

Vacuum compaction

---

[492] Methodology to Manage Material and Waste from Nuclear Decommissioning; Waste Management & Decommissioning Working Group, World Nuclear Association, Produced by: World Nuclear Association, Published: February 2019, Report No. 2019/001

[493] IAEA-TECDOC-1817, SELECTION OF TECHNICAL SOLUTIONS, FOR THE MANAGEMENT OF RADIOACTIVE WASTE, INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 2017; ISBN 978–92–0–104717–5

[494] TECHNICAL REPORTS SERIES No. 402; HANDLING AND PROCESSINGOF RADIOACTIVE WASTE FROMNUCLEAR APPLICATIONS; INTERNATIONAL ATOMIC ENERGY AGENCYVIENNA, 2001: ISBN 92–0–100801–5

[495] TECHNICAL REPORTS SERIES NO. 402; HANDLING AND PROCESSINGOF RADIOACTIVE WASTE FROMNUCLEAR APPLICATIONS; INTERNATIONAL ATOMIC ENERGY AGENCYVIENNA, 2001: ISBN 92–0–100801–5

Vacuum Compaction is designed to compact, pack, and seal toxic low-level waste at its point of origin. The whole operation is performed under vacuum and the material is packed into highly chemical resistant multilayer plastic bags. The volume reduction factor is generally at least two [496].

Reference: Thermal treatment for radioactive waste minimisation published in EPJ Nuclear Sci. Technol. 6, 25 (2020), gives an overview of the current research on thermal treatment [497].

6.2.3 Loading

The containers for packaging radioactive waste are normally carbon steel drums of at least 200 L. They can be used for immobilising liquid and wet or dry solid wastes.

For the loading of the waste into containers for final or interims storage, a couple of requirements have to be met. The waste package has to meet the acceptance criteria of the disposal site. In general qualitative acceptance criteria for waste packages consigned to a waste repository should cover [498] :"

- The radionuclide inventory
- Radiation levels (dose rate)
- Mechanical properties
- Chemical durability
- Gas generation
- Combustibility and thermal resistance
- Limiting or avoiding free liquids, explosive and pyrophoric materials, compressed gases, toxic and corrosive materials
- Physical dimensions and weights
- Unique identifications
- Responsibilities and organisations
- Compliance with codes, standards and national regulations

In order to be sure that packages are still in physically good shape and safe handling is ensured in the next processing steps, special acceptance criteria for interims storage have to be met. The acceptance criteria for waste packages in storage include [499]:

- The maximum allowable weight per package
- The mechanical resistance of the packages to be stacked,
- The satisfactory corrosion resistance of the package metal

---

[496] TECHNICAL REPORTS SERIES NO. 402; HANDLING AND PROCESSING OF RADIOACTIVE WASTE FROM NUCLEAR APPLICATIONS; INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA, 2001: ISBN 92–0–100801–5

[497] THERMAL TREATMENT FOR RADIOACTIVE WASTE MINIMISATION; MATTI NIEMINEN, MARKUS OLIN, JAANA LAATIKAINEN-LUNTAMA ET AL.; EPJ NUCLEAR SCI. TECHNOL. 6, 25 (2020); HTTPS://DOI.ORG/10.1051/EPJN/2019040

[498] TECHNICAL REPORTS SERIES NO. 402; HANDLING AND PROCESSING OF RADIOACTIVE WASTE FROM NUCLEAR APPLICATIONS; INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA, 2001: ISBN 92–0–100801–5

[499] TECHNICAL REPORTS SERIES NO. 402; HANDLING AND PROCESSING OF RADIOACTIVE WASTE FROM NUCLEAR APPLICATIONS; INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA, 2001: ISBN 92–0–100801–5

- No loss of integrity after a test drop from a height equivalent to that found during transportation
- Sufficient resistance to a standard fire test

Experiences show that proper segregation of the waste arising from the start and appropriate storing into containers, usually in plastic bags or 200 L drums bearing adequate labelling and color-coding, simplifies the waste treatment process enormously.

For safe handling and storage of packed drums or containers, the loaded waste inside the container has to be immobile. To ensure this, the voids are filled by a flow of fluid cement slurry or other grouting material. For complete encapsulation, the vibration of the drum/container is applied when the slurry is purred into the container. The use of cement is common to do to its.

- Good flow characteristic
- Relatively free from water
- The relative simplicity of handling
- The availability of the raw material
- The low cost

The caution has to be exercised, for the immobilisation of non-ferrous metals and alloys. These materials are attacked by the alkaline solution of the wet cement grout. For example, ferrous bearing metals form a protective oxide film over themselves, causing expansion. This may result in cracking and deterioration of the concrete and causing the container to break. A common practice, in this case is the coating with bituminous or polymer material prior to introducing the cement grout or using a polymer-based grout, such as molten polyethylene, sulphur cement or bitumen [500].

---

[500] TECHNICAL REPORTS SERIES No. 402; HANDLING AND PROCESSING OF RADIOACTIVE WASTE FROM NUCLEAR APPLICATIONS; INTERNATIONAL ATOMIC ENERGY AGENCY VIENNA, 2001: ISBN 92–0–100801–5

6.3 In situ Radioactive Waste characterisation and segregation

Developments in insitu radioactive waste characterisation have advanced rapidly in recent years and have enabled waste segregation and rerouting to become best practice in terms of radioactive waste management and disposal. In situ characterisation of higher activity wastes tends to be more straight forward as there is a strong radioactive signature to detect, particularly when looking at higher activity gamma bearing wastes but when considering alpha and beta wastes, particularly as the levels of total activity in the waste decreases, the challenges become greater. Insitu characterisation of wastes is also preferred in many cases over the more traditional sampling and laboratory analysis approach as insitu characterisation tends to provide data in real time, at reduced costs and minimises dose to operatives undertaking the characterisation work.

With all characterisation, it is necessary to identify the characterisation requirements, and this is the same for insitu radioactive waste characterisation as for any other characterisation driver. Establishing initially the Data Quality Objectives[501] and defining the characterisation principles; '*what are the characterisation questions we are trying to answer?*' are equally important. Characterisation and waste sorting and segregation can result in significant cost and schedule savings with the disposal costs falling by orders of magnitude as the waste categories decrease, therefore the cost savings and the preservation of valuable capacity in disposal facilities drive the characterisation of waste and the separation or sorting of lower, from higher activity wastes.

A range of in situ characterisation techniques are currently available and used across the radioactive waste industry (see also Section 4.8).

- Basic characterisation techniques include simple dose measurement readings from which, using waste fingerprints and calculation, a radionuclide inventory in the waste can be developed but these are not without challenge. A commonly used dose based technique is referred to as the 'Dawson technique' where a range of dose readings are taken from around a drum of waste and the total activity calculated from the dose and the agreed fingerprint. This technique, whilst still used has a significant degree of uncertainty associated with the results and has become less favoured currently.
- High/low resolution gamma systems are commonly used, and the software associated with these systems has advanced significantly in recent years. The principle advantage of this approach compared to simple dose readings is the gamma systems can identify and measure a wide range of individual gamma emitting radionuclides directly rather than by interpretation. The data also enables detailed modelling of the distribution of radioactivity within wastes and so introduces the possibility of the wastes being more readily sorted and segregated into different waste routes.
- Neutron Interrogation, for example Passive Neutron Coincidence Counters (PNCC) are less commonly deployed as they are more difficult to deploy but characterise waste (particularly plutonium bearing wastes) by bombarding the waste with high energy neutrons and interrogating the interactions between these neutrons and radio isotopes within the waste.

---

[501] EPA, "Guidance on Systematic Planning Using the Data Quality Objectives Process," EPA QA/G-4, 2006.

- There are additional standoff techniques used to greater or lesser extents to characterise radioactive wastes, some of which are in relatively early stages of development and these include:
    - Alpha Camera
    - Alpha Spectroscopy
    - Laser Induced Breakdown Spectroscopy (LIBS)
    - Raman Spectroscopy
    - Gamma mapping
    - X-Ray Fluorescence

The main challenge associated with many of in situ characterisation techniques is they require a detailed fingerprint of the waste to be established if the material is to be properly characterised. Many of the techniques rely on the development of a fingerprint for the waste which needs to be developed by traditional analytical techniques and without this, can only provide limited characterisation of the wastes.

Characterisation of waste underpins the sorting and segregation of the wastes which ultimately delivers the cost and repository capacity savings required. As with in situ characterisation, a range of approaches to the sorting and segregation of wastes has been developed including;

- Workface sorting: The manual sorting of items such as PPE, works generated wastes etc., undertaken at, or close to the point of work and when the waste is generated. This can be as simple as having a number of different 'waste bins' for certain material types. Workface sorting can also in some circumstances, include basic segregation via the use of handheld frisk probes etc. Working conditions can restrict the amount of sorting and segregation that is achievable. Also, human error can become a factor e.g. placing items in wrong bin.
- Manual sorting (table): Mixed LLW waste is sorted manually usually on a sorting table. Bags / drums are opened, monitored and the contents spread out for sorting in terms of both material types and basic activity check.
- Sorting Line (manual): Step by step sorting / segregation process conducted within a fume hood / glovebox type arrangement. The sorting itself is still manual via gloveports.
- Sorting line (remote): Again this is generally conducted within a glovebox type set up but can also be used for higher activity wastes within shielded cells/caves. The sorting / segregation is via manipulators / robotic arms and CCTV (still operator controlled)
- Bulk scanning: This is widely used where large volumes of radiologically contaminated waste needs to be assessed in terms of its activity. This can vary from small bags of wastes to drums / large bulk bags. Various scanner types / sizes are available commercially.
- Conveyor systems: Convey / gate systems have become more prominent for the sorting and segregation of wastes. Originally developed for decommissioning type works where bulk soil / building demolition materials needed to be sorted on an ongoing basis.
  Such systems can include simple material type sorting e.g. use of Air Classifiers to remove plastic and magnets to remove metal. Radioactivity can also be measured from basic Alpha/Beta to LRGS and X-ray to segregate different activities of waste from within a batch. Some more advanced conveyor system also utilises LIBS to identify and segregate wastes.

6.4 Segmentation of large surface-contaminated components

Segmentation or disassembly and other size reduction techniques may be applied before the conditioning of waste that is bulky or oversized in relation to the intended processing (e.g. worn out components or structures). Processes for segmentation or disassembly typically use cutters with high temperature flames, various sawing methods, hydraulic shearing, abrasive cutting and plasma arc cutting. The need for means of preventing the spread of particulate contamination and for fire protection in case of pyrophoric waste should be considered in the selection of the method and in the operation of the equipment.

A large component can be considered as any part of a nuclear facility that may be removed without being cut, that is conditioned in a non-standard package for disposal or storage and that requires specific consideration by local regulatory body due to its weight, its volume or the extent of its radiological contamination. It means that the related standard process involved by the operator to manage transport, storage or disposal cannot be applied without any modification. Generally, it addresses items, such as steam generators, pressurisers, reactor pressure vessels and heads, or transport casks. The way to manage large disused components is of wide interest, especially among operators of nuclear facilities, decommissioning organisations, waste transporters and waste-management agencies, as well as safety regulatory body and other national authorities. Managing and disposing of large components have been successfully carried out in various countries and the experience gained from it provides the baseline for optimising those decisions.

Depending on the selected strategy, the option for the management of large components will be influenced by the availability of waste storage and treatment facilities. For instance, the recourse to radioactive decay may avoid the use of expensive remote-controlled segmentation techniques or cutting may possibly be carried out manually with less radiation-protection requirements. Through radioactive decay, clearance may be granted and result in either free release or de-categorisation of the waste, thus decreasing the amount of radioactive waste in a particular category. It is crucial to know before making a decision, whether an interim storage facility is already available or if it needs to be constructed specifically. On the other hand, the necessity for interim-storage capacity is a disadvantage due to its large investment costs and to the long-term operation of such facility. Furthermore, the treatment capacity for large components in a dedicated external facility may be very useful in order to optimise decommissioning works.

The final form of radioactive waste will be specified by another key driver, which is the capacity of the disposal facility to accommodate large components. If such a capacity is available there will be no need for total segmentation in order to standardise conditioning. That may avoid costs and doses to workers. Nevertheless, disposal facilities for radioactive waste represent a scarce and valuable resource in all countries, thus always giving waste reduction and categorisation a high priority.

However, the use of an external storage or treatment facility associated with a disposal facility will not be considered if an additional, but major, key driver, that is the feasibility of transportation, prevents those options. The transportation of large components may require suitable rail, road, sea or river transport systems and the development of suitable transport containers. Where there is a need to develop the elements of waste management or transportation systems, that may cause significant delays to decommissioning projects, which, in turn, may have a significant impact on cost. The situation, however, may be very different depending on the national framework.

Economic issues are also key drivers, since they need to reflect the technical difficulties of all phases throughout waste management, including both safety and radiation protection issues. The economic optimisation of decommissioning and decommissioning waste management should also reflect the overall technical optimisation of the selected decommissioning process. The minimisation of the timeframe for the implementation of dismantling has to be considered and may be a key driver for the selection of a management option for large components. Usually, the dismantling of large components is on the critical path of a decommissioning project and there is considerable interest in finding an alternate solution to the standard option, which is full segmentation and conditioning in standard packages. Furthermore, because cutting large components is a complex process, many technical and safety hazards may be avoided when segmentation activities are limited or even better excluded. Finally, the early removal of large components would also improve in-plant logistics for future decommissioning activities. On the other hand, preparatory works may also be required, especially if the installation was not originally designed for the handling and removal of large components, and the necessary timeframe for those activities should be considered. Normally any schedule reduction would likely result in a limitation of occupational doses as well.

One-piece removal of large components has been performed at a number of nuclear facilities as a means of simplifying the dismantling or waste disposal processes. The benefits of this approach are reduced project costs, reduced time-scales, lower operator dose uptake and increased operator safety. This technique is especially attractive when there is close/ready access to either water or rail transportation facilities. Removal of a large component to an adjacent facility, e.g. waste processing facility or special purpose containment, can be used in order to reduce operator dose uptake and/or improve access in order to simplify subsequent segmentation processes. One-piece removal could prove, in some cases, to be more cost effective and result in less radiation exposure than if the components were segmented in situ.

6.4.1 Description of techniques

Segmentation techniques are already discussed in subtopic 6.1.

6.4.2 Experiences/Case studies

Lots of large component segregation case studies can be found on WM symposia website.[502]

---

[502] https://www.wmsym.org

6.4.2.1 Lithuanian experience adopted for Ignalina NPP decommissioning[503], [504], [505], [506]

One of the first D&D project at Ignalina NPP was dismantling and decontamination of the emergency core cooling system (ECCS) equipment located in the Building 117/1. The main systems in Building 117/ were the following:

- Helium Facility equipment (small bore gas system associated pipe work in an annex to Building 117/1);
- ECCS pressure vessels (16 carbon steel water vessels at around 14m high x 1.5m diameter, 80mm thick and 47650 kg mass each with internal contamination) (see Figure 6.4-1);
- Large diameter carbon steel pipe work (up to 400mm diameter);
- Small diameter carbon steel pipe work;
- Large fast acting valves on the main ECCS pipelines to Unit 1;
- Various carbon steel fabrications including floor structures with deck plates, steel platforms and staircases;
- Miscellaneous plant items, such as electric motors, control panels, gauges, etc.

Main part of the equipment which was directed to the dismantling belongs to the ECCS vessels. The contamination of the equipment directed to the dismantling was from the free release level up to low level waste. D&D of Building 117/1 involves different types of activities, tools, equipment and systems.

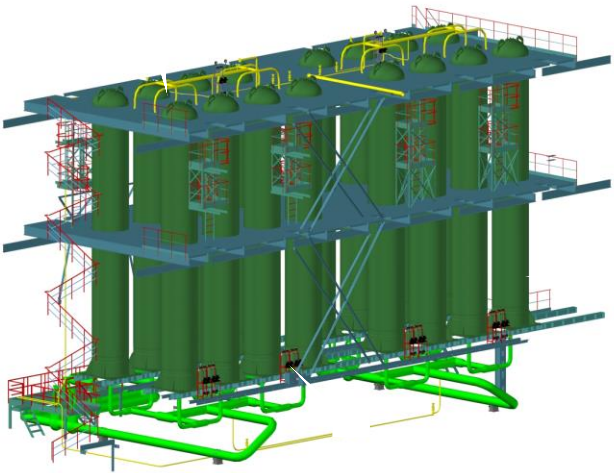
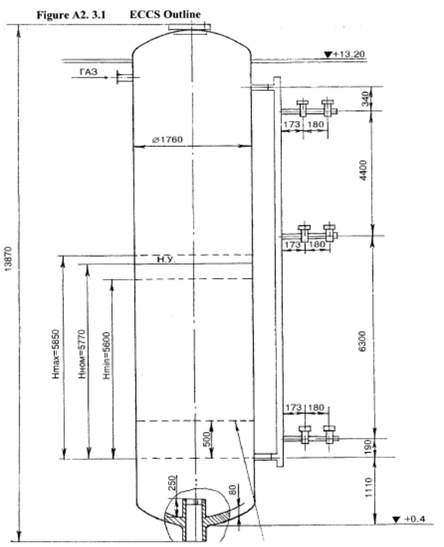


Figure 6.4-1 Ignalina NPP Emergency Core Cooling System pressure vessels[507], [508]

[503] https://www.hindawi.com/journals/stni/2015/650810/
[504] https://ec.europa.eu/jrc/sites/jrcsh/files/jrc_20120911_decommissioning_e_uspuras.pdf
[505] https://www.youtube.com/watch?v=lbboyVmyhzM
[506] https://alara.ee/wp-content/uploads/2018/08/teh-seminar-1/Ignalinadekomisjoneerimineprojektid1.pdf
[507] https://www.hindawi.com/journals/stni/2015/650810/
[508] https://ec.europa.eu/jrc/sites/jrcsh/files/jrc_20120911_decommissioning_e_uspuras.pdf

All 16 ECCS pressure tanks were of the same design. The tanks were welded carbon steel structures with dished ends at the top and at the bottom and a straight tubular section in between. The bottom of the tank was equipped with a supporting collar for the mounting of the tank to its supporting beams. The interior of the tanks was accessible via manholes arranged in the upper dished end and steps arranged in the tubular section. The large diameter discharge pipes of the tanks were located at the centres of the lower dished ends. Small diameter pipe sockets were laterally arranged on the tubular tank section for the connection of the pressurisation system and the filling level indicators.



Main Technical Data ECCS PT:

Height:	13.370 mm
Outer Diameter:	1.760 mm
Wall Thickness:	80 mm
Material:	Carbon Steel 16GS-6
Weight:	47,7 t
Number of Tanks:	16

Figure 6.4-2 - Ignalina NPP Emergency Core Cooling System pressure vessels – Main Technical Data[509], [510]

Prior to the start of ECCS Pressure Tanks (PT) segmentation, several preparatory works were carried out:

- Installation of a 3.2 t Electric Portal Crane above level 13.200 to allow the pick-up and transportation segmented parts down to the intermediate storage area;
- Disconnection from ECCS PT's from all connected systems;
- Disconnection from ECCS PT's and removal of large diameter ECCS pipelines;
- Cut out of floor plate sections and small supporting beams around ECCS PT's to allow installation of scaffolding;
- Scaffolding around ECCS PT to be segmented;
- Installation of mobile ventilation/filtering system and connection of aspiration ducts to ECCS PT discharge socket.

The ECCS PT were cut by application of the autogenous flame cutting method using acetylene and oxygen. The method was chosen due to its proven efficiency in various nuclear decommissioning projects when cutting carbon steel with a large wall thickness. Nominal cutting speeds of approx. 250 – 300 mm/min

[509] https://www.hindawi.com/journals/stni/2015/650810/
[510] https://alara.ee/wp-content/uploads/2018/08/teh-seminar-1/Ignalinadekomisjoneerimineprojektid1.pdf

have been proven when cutting carbon steel plates with 80 mm wall thickness and can, thus, be assumed as achievable when cutting the ECCS pressure tanks. The autogenous flame cutter was installed on an electrically driven tractor system which runs on a guiding track that was mounted to the outer surface of the ECCS PT.

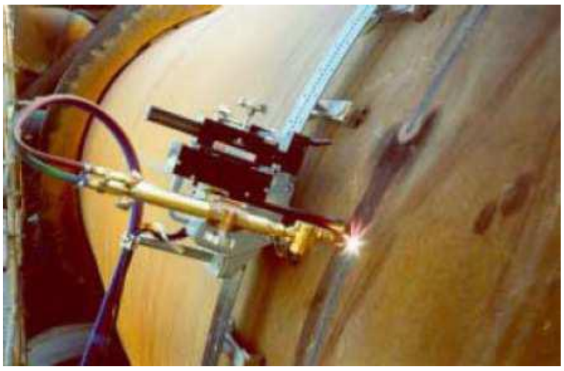


Figure 6.4-3 - Flame cutter on the outer surface of the ECCS PT[511]

The flame cutter and tractor system were controlled from a distance. This reduces personnel dose uptake as it limits the duration of personnel close to the tank and also minimises risk with regards to hot-work in a confined space. In addition, the guiding system provides higher accuracy of the cut which will later facilitate the segmentation of the ECCS parts.

Due to the vectored air flow generated by the ventilation system, cutting fumes and aerosols were directed towards the inner tank and then on to the filter banks. After full penetration of the cutting flame through the wall of the tank, the slag created was washed out into the tank and drop to the tank bottom from where it was removed in the final stage of the PT segmentation process.

Whereas upper and lower dished end sections of the ECCS pressure tanks were segmented manually and removed piece by piece, the segmentation of the cylindrical tank section were carried out using the remotely controlled torch + tractor system.

ECCS pressure vessels rings sections were segmented in the following order:

1. 4 vertical cuts were performed (90° offset, each);
2. Insertion of a supporting ring. The supporting ring were stabilised the segments in the later lifting operation;
3. Attachment of the lifting gear to the 4 segments;
4. 1 radial cut were performed all around the pressure tank (clearing cut of the 4 ring segments);
5. After cut-off and cooling down the ring segment were wrapped in vinyl to prevent spreading of contamination during transportation by the EPC crane.

---

[511] https://www.hindawi.com/journals/stni/2015/650810/

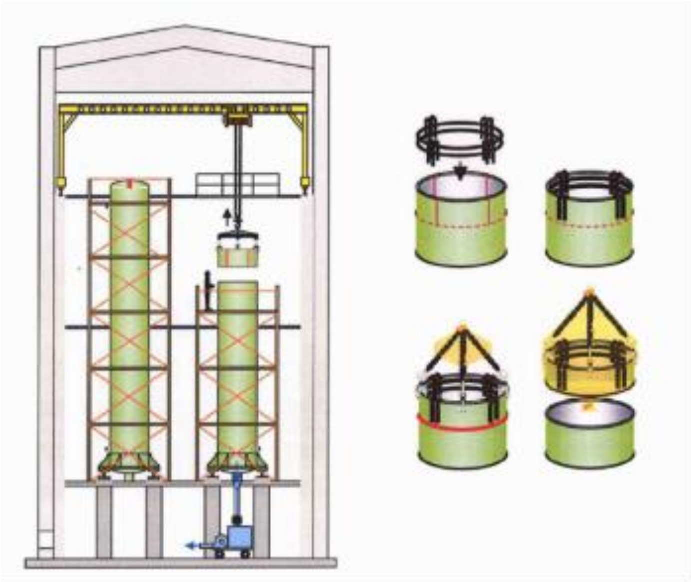


Figure 6.4-4 - ECCS pressure tanks segmentation[512], [513], [514]

The Mobile Filtration Unit (MFU) connected to the base of ECCS vessel operated during segmentation process in order to provide extraction ventilation flow down through the ECCS vessel.

The removal of ECCS vessel segments will continue in this manner until each vessel has been fully removed to its bottom dome and mounting flange. The bottom dome and mounting flange will be best cut up manually into convenient pieces.

This process will be repeated for each ECCS vessel until all 16 have been removed. It is envisaged that two ECCS vessels will be undergoing size reduction at any time in order to provide some degree of parallel work, in order to facilitate this but retain only one crane one ECCS vessel will be undergoing set-up works whilst the other is being cut (the cutting of the other tank then takes place whilst the cutting gear is being repositioned on the first tank).

Note: The cut pieces of ECCS vessel will need to be hoisted up to the top level before the crane can traverse across to the well area to -3.60m level due to the presence of the intermediate floors.

[512] https://ec.europa.eu/jrc/sites/jrcsh/files/jrc_20120911_decommissioning_e_uspuras.pdf
[513] https://www.youtube.com/watch?v=lbboyVmyhzM
[514] https://alara.ee/wp-content/uploads/2018/08/teh-seminar-1/Ignalinadekomisjoneerimineprojektid1.pdf

6.5 Dismantling of surface-contaminated piping and small components

Dismantling is understood as the cutting or disassembly of a component in order to remove it from its original place. This is to be differentiated from segmentation that has the purpose of size reduction or segregation.

In the context of decommissioning of nuclear power plants, the dismantling techniques perform an important task. In order to meet requirements for dismantling, different cutting and separation techniques can be used. For the selection of a dismantling technique suitable for respective applications, however, many boundary conditions are to be considered, because an ideal universally applicable cutting tool may not be available. It may be expedient to use a dismantling technique which is not very suitable in order to save additional high investment and operating costs.

6.5.1 Description of techniques

Different dismantling techniques are presented in Section 6.1 above.

Due to the number of pipe sizes and their locations within the NPPs, a number of dismantling methods and techniques could be used as most appropriate to the particular situation. The dismantling techniques differ with respect to the need for track-mounted technologies for thick-walled large pipework. For the small pipes dismantling the using a 'toolbox' of standard techniques could be most appropriate, because it is not always possible to accurately identify the best system for each cut. Such a 'tool box' would typically contain the following techniques:

- Hydraulic shears;
- Reciprocating saws;
- Electric nibblers;
- Angle grinders;
- Bolt croppers;
- Hacksaws;
- Hydraulic spreaders;
- Hydraulic nut splitters.

For larger diameter pipelines and filters other techniques could be considered such as:

- Band Saw;
- Plasma cutters;
- Oxy-acetylene torch;
- Diamond wire saw.

A 'tool box' approach allows flexibility during implementation. Over-reliance on a single favored technology may restrict the efficiency and effectiveness of the dismantling operations, while the potential to utilise a common process or set of tooling across a number of areas may bring significant benefits.

More importantly for dismantling process is the requirement for manual cutting techniques that minimise secondary waste, and, by application of the Strategic Guiding Principles, optimise costs and resources by simplifying methodology where possible. In that respect, cold cutting techniques are favoured for in-situ dismantling work of contaminated pipelines, tanks and heat exchangers (not including vessels) as the need for Mobile Filtration Unit (MFU) and ventilation modifications is minimised.

Table 6.5-1 Qualitative comparison of hot cutting techniques

Criteria	Plasma cutter	Oxy-acetylene cutter	Notes
Versatility (type of material to be dismantled)	• Carbon steel • Stainless steel	Carbon steel	Oxy-acetylene is totally unable to cut 36% of the initial material
Secondary waste generation	• Swarf • 4 mm typical kerf width	• Slag • 6 mm typical kerf width	For carbon steel, oxy acetylene is likely to produce up to 50% more secondary waste due to kerf width
Set up time	Minimal	Minimal	For handheld torches the set up time is approximately the same
Conventional safety aspects	• Arc flash • Hot work (although heating of Material is minimised) • Metallic fume	• Flash • Hot work • Carbon monoxide and other emissions • Explosion hazard • Increased fire hazard (oxygen) • Manual handling of gas cylinders	Very large air volumes required to dilute CO emissions from oxy-acetylene cutting
Radiological safety aspects	• Airborne contamination	• Airborne contamination	
Cutting speed	Up to 2.5m/min	Up to 0.5m/min	Based on 10mm carbon steel
Type of material to be subject of hot cutting	1) Piping of D>300mm stainless and carbon steel; D>200 thickness more than 10mm 2) Large items vessels and tanks (stainless	Only carbon steel material, the same items as for plasma cutter.	

Criteria	Plasma cutter	Oxy-acetylene cutter	Notes
	steel and carbon steel) 3) Heat exchangers (carbon and stainless) 4) Filter vessels (carbon and stainless)		
Cutting time ((a)1124m in stainless steel up to 12mm thick, (b) 1555m in carbon steel up to 12mm thick, (c) 100m in carbon steel at 16mm thick)	a) 450 minutes b) 622 minutes c) 67 minutes **Total: 1,139 minutes**	a) 11240 minutes (recip. saw) b) 3110 minutes c) 200 minutes **Total: 14,550 minutes**	Assumed cutting speeds at up to 12mm thickness; Plasma 2.5m/min, oxy-acetylene 0.5m/min, reciprocating saw 0.1m/min. At 16mm thick, plasma 1.5m/min, oxy-acetylene 0.5m/min

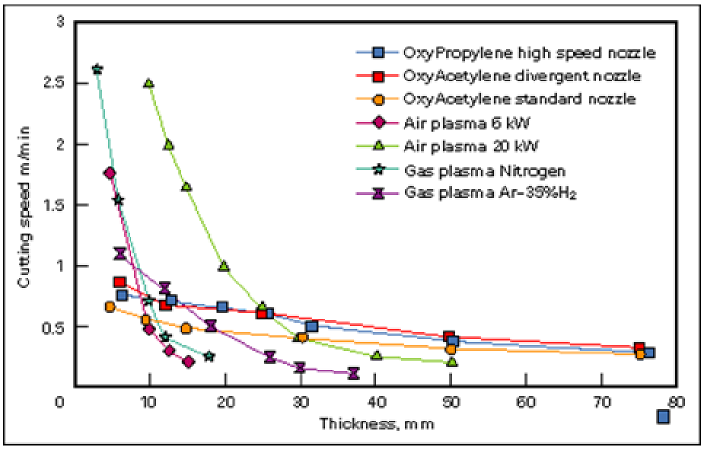


Figure 6.5-1 Comparison of cutting speeds (data produced by The Welding Institute, UK)

6.5.2 Experiences/Case studies

6.5.2.1 <u>Lithuanian experience adopted for Ignalina NPP decommissioning</u>

Dismantling of pipelines

The following sections describe the likely methodology for the dismantling of a typical variety of pipe sections.

*Carbon steel pipe; diameter - 426mm, wall thickness - 24mm, **freely accessible***

For the dismantling of such diameter freely accessible carbon steel pipes the oxy-acetylene track cutter was selected. Following steps were applied for dismantling:

1. Identification of correct section of pipeline to be dismantled;
2. Mark cutting point (approximately 1m lengths required). The layer of paint may be removed from the surface at the cutting locations using a wire brush although this is not strictly required by the design;
3. Conduct radiological check of dismantling area;
4. Erect containment tent;
5. Connect containment ventilation system to pipe to be dismantled, ensure air flow is extracted through the pipe to be cut;
6. Support pipe sections;
7. Attach track cutter guide chain to pipe at cutting location;
8. Attach tractor unit and oxy-acetylene cutter to guide chain;
9. Conduct circumferential cut using oxy-acetylene track cutter;
10. Remove severed pipe section onto pallet on floor;
11. Seal pipe ends using polythene sheet and adhesive tape;
12. Collect slag and fine debris using HEPA filtered vacuum cleaner;
13. Move pallet & pipe section to collection point where steel post pallet is located;
14. Load pipe section into steel post pallet and apply identification tag;
15. Conduct initial radiological measurements.

*Carbon steel pipe; diameter - 426mm, wall thickness - 24mm, **tightly mounted to wall***

For the dismantling of such diameter tightly mounted to wall carbon steel pipes use of the oxy-acetylene track cutter were not possible due to lack of space around of pipeline. Due to that diamond wire saw was selected. Sequence of the dismantling removed some steps described above and consist of the following:

1. Identify correct section of pipeline to be dismantled;
2. Mark cutting point (approximately 1m lengths required);
3. Conduct radiological check of cutting area;
4. Erect containment tent;
5. Connect containment ventilation system to pipe to be dismantled, ensure air flow is extracted through the pipe to be cut;
6. Support pipe sections;
7. Attach diamond wire saw pulleys to wall and floor at cutting location;
8. Locate diamond wire on pulleys;
9. Conduct cut through pipe section using diamond wire saw;
10. Remove severed pipe section onto pallet on floor;
11. Seal pipe ends using polythene sheet and adhesive tape;
12. Collect swarf and fine debris using HEPA filtered vacuum cleaner;
13. Move pallet & pipe section to collection point where steel post pallet is located;

14. Load pipe section into steel post pallet and apply identification tag;
15. Conduct initial radiological measurements.

Using above presented approach different type of pipelines and other small equipment were successfully dismantled at Ignalina NPP. Only the step related to the cutting operation was different as different tool, depending on pipe material and diameter, was selected.

Table 6.5-2 Examples of the used cutting techniques for different pipes and other small equipment

Type of material	Dismantling tool
Carbon steel pipe; 220mm OD, 13mm wall thickness, **freely accessible**	Manual oxy-acetylene cutter
Carbon steel pipe; 220mm OD, 13mm wall thickness, **tightly mounted to wall**	Portable band saw
Carbon steel pipe; 89mm OD, 6mm wall thickness, **freely accessible**	Reciprocating saw (angle grinder to cut pipes of this size instead of saw could be used)
Carbon steel pipe; 89mm OD, 6mm wall thickness, **tightly mounted to wall**	Manual oxy-acetylene cutter (angle grinder to cut pipes of this size instead of saw could be used)
Stainless steel pipe; 32mm OD, 3mm wall thickness	Reciprocating saw, angle grinder or twin disc cutter
Stainless steel tube; 14mm OD, 2mm wall thickness, **freely accessible**	Reciprocating saw, angle grinder or twin disc cutter
Stainless steel tube; 14mm OD, 2mm wall thickness, **tightly mounted to wall**	Twin disc cutter or angle grinder

Usually there are a variety of minor carbon steel fabrications used to support pipelines, controls, pressure gauges and manifolds or provide man access to raised items. Ignalina NPP experience shows that these fabrications must all be dismantled in order to allow their processing for Free Release or Landfill disposal. For this different type of supporting equipment following dismantling techniques was applied.

Table 6.5-3 Examples of the used cutting techniques for different supporting equipment

<table>
<tr><th>Type of equipment/material</th><th>Dismantling tool</th></tr>
<tr><td>Different type of bolted structures</td><td>If bolted structures are encountered the dismantling can be carried out in the following ways;
Use standard hand tools to unbolt the connections;
Use hydraulic nut splitters, oxy-acetylene torch or angle grinders to cut off the fasteners if seized;
Use oxy-acetylene torch, reciprocating saws, angle grinders, twin disc cutter or hydraulic shears (as most appropriate to the particular item) to cut the steel sections beyond the bolted connection point.
The final selection of the most suitable method will need to be taken by the first-hand inspection of the item to be dismantled.</td></tr>
<tr><td>Welded construction</td><td>Fabricated structures of this general section size can be relatively easily cut using a number of the smaller ‘tool box’ techniques, namely;
Oxy-acetylene torch;
Reciprocating saws;
Twin disc cutter;
Angle grinders;
Hydraulic shears;
Electric nibblers;
Portable band saw.</td></tr>
</table>

Where possible, it should be taken that the steel fabrications should be kept in straight pieces to aid decontamination and packing. Also, in some cases, reuse of steel sections on site may be desirable and retention of long pieces of steel section (above the assumed 1m lengths for handling as palletised items) may be envisaged.

6.6 Segmentation of interior concrete structures (e.g., bioshield)

Concrete waste makes up the major portion of decommissioning waste accounting for more than 50%. A significant amount of research has been performed on the volume reduction and recycling of waste involving concrete. However, the dismantling and disposal processes for the radioactive concrete structures need to be established. Radioactive concrete waste is mainly divided into contaminated concrete and activated concrete. Contaminated concrete volumes can be reduced by applying proper decontamination technologies. However, deeply activated concrete is difficult to volume reduce by decontamination. Therefore, an efficient treatment strategy for activated concrete is essential for the minimisation of radioactive waste disposal.

The most significant activated concrete structure in a nuclear power plant is the biological shield around the reactor vessel. The activated biological shield due to thermal and resonance neutron flux during the operational period needs proper dismantling and disposal strategies that enable significant radioactive waste minimisation and disposal cost savings [515]. The evaluation of results from the waste generated from the biological shield lies in the very low-level waste (VLLW) and low-level waste (LLW) [516]. The extent of activation of biological shield concrete depends significantly on the impurities contained in the concrete and these impurities vary greatly depending on different factors such as the manufacturing process and added aggregates. The radiological risk should be minimised in the decommissioning preparation phase through radiation protection management.

In dismantling a biological shield, decontamination must be performed to minimise unnecessary contamination and radioactive waste generation. Surface contamination needs to be accurately identified through surface radiation measurements and removed by applying proper decontamination technologies. The dismantling plan needs to be set up that starts with the detailed structural investigation, estimation of radiation inventories, investigation of existing dismantling methods, and dismantling sequence planning. The segmentation of concrete associated with a biological shield can be divided into the method of cutting from the non-activated area and the method of cutting from the activated area. A segmentation strategy for interior concrete structures should be selected from various ways considered to be applied according to the following requirements;

- Use of remote-control tools
- Dismantle the activated portion without damaging the remaining structure
- Reduction in secondary waste.

The dismantling method described here use cutting because conventional methods such as hydraulic breakers, controlled explosives, expansive grout, and thermic lance are not suitable and are difficult to apply to heavy reinforced and activated concrete [517].

---

[515] Cheol-Seung Cheon and Chang-Lak Kim, "The Dismantling and Disposal Strategy of a Biological Shield for Minimisation of Radioactive Concrete Waste During Decommissioning of a Nuclear Power Plant" September 18, 2017.

[516] G.Y. Cha, S.Y. Kim, J.M. Lee, and Y.S. Kim, "The Effects of Impurity Composition and Concentration in Reactor Structure Material on Neutron Activation Inventory in Pressurized Water Reactor", J. Nucl. Fuel Cycle Waste Technol., 14(2), 91-100 (2016).

[517] Tetsuo HASEGAWA, Makoto ICHIKAWA, Seishi SUZUKI, "Design of machine for dismantling biological shield concrete"

6.6.1 Description of techniques

6.6.1.1 Diamond wire sawing machine

To cut the inner surface of the biological shield, various cutting techniques are available one possibility is diamond wire sawing. Details on this technique are given in Section 6.1.1.6.1.

6.6.1.2 Flame cutting

This technique is used to cut concrete when vibration to the surrounding area is intolerable or when the thickness of the concrete to be segmented (biological shield) exceeds the capabilities of mechanical cutters (not including diamond wire cutters). It involves a thermite reaction in which a powdered mixture of iron and aluminium oxidises in a pure oxygen jet. The high temperature of the jet (8000 °C) causes the concrete to decompose while the mass flow rate through the flame cutting nozzle acts to clear the debris from the material area. The rate of cutting depends on the depth of the concrete being cut. Any kind of rods in the concrete adds iron to the reaction, thus sustaining the flame and assisting the reaction. The heat and smoke that results from the process can be removed with 5-7 hp squirrel cage blower that is directed through a flexible duct that sprays water to hold down smoke particulates. Flame cutters are capable of cutting through a maximum depth of 60in with or without reinforcing rod [518].

6.6.1.3 Thermite reaction lance

The thermite reaction lance consists of a combination of steel, aluminium, and magnesium wires packed inside an iron pipe through which a flow of oxygen gas is maintained. The lance is ignited in the air by a high-temperature source such as an electric arc or an oxygen burning torch. During operation, the thermite reaction at the tip completely consumes the constituents of the lance and causes the temperature to reach 2200-5000°C depending on the environment. Because the lance itself consumed as the cut is achieved, so the replacement of the lance may become necessary. In that case, oxygen is cut off and new lance is then coupled to the holder. The thermite reaction lance can be used in air or underwater. This technique has been used to successfully cut the top of the biological shield that contains approximately equal quantities of concrete and steel. The process has high cutting rates but improvements to the ventilation system are required to deal with the fumes effectively [519].

6.6.1.4 Controlled Blasting

Controlled blasting is normally recommended to remove large segments or heavily reinforced concrete sections. The process includes drilling holes in the concrete, loading them the explosives and detonating them using 1-3 ms delayed firing technique. The delayed firing give rise to fragmentation and controls the direction of material movement. A delay period of approximately one ms/ft of burden provides sufficient time for free face movement. Delayed firing also reduces the vibration impact on adjacent structures. During detonation, each borehole fractures radially and the radial fractures in adjacent boreholes form a

---

[518] Decommissioning handbook by U.S Department of energy Office of Environmental Restoration, March 1994

[519] Decontamination and Demolition of Concrete and Metal Structures During the Decommissioning i of Nuclear Facilities, IAEA TECHNICAL REPORTS SERIES No. 286, Vienna 1988

fracture plane. The wave produced during detonation separates the fractured surfaces and moves the material towards the structure's free face [520].

The methods that are used to drill for blast hole preparation include percussion air-operated drills, electric, pneumatic or diesel driven rotary drills or diamond-core abrasive drills. Percussion drills are the most versatile and can economically drill $1\frac{1}{4}$ in. to 2 in. The selection of the best type of explosive requires an evaluation of the properties of the explosive and concrete themselves. A blasting expert should be employed to select the best explosive for the purpose. Typical types of explosives used for concrete removal include; PETN (pentaerythritol tetranitrate), 85% high-velocity gelatine dynamite, cast TNT (high detonation pressure primers) and water gel explosives [521].

A blasting mat (normally constructed from automobile tyres, tied together into large pieces, rubber mats for smaller debris and filter mats to retain fine dust) can be placed over the blast area. In order to suppress the dust, continuous fog sprays of water should be used before, during and after the blast. The exposed reinforcing bar can be cut using an oxyacetylene torch or bolt cutters. This technique is employed to demolish steel-reinforced, radioactive biological shields [522].

6.6.1.5 Abrasive water-jet cutting

Detailed information on the technique itself are given in section 6.1.1.5

The abrasive water jet can be used to cut reinforced concrete and metal structures. However, the application of this technique for dismantling large reinforced concrete structures in the interior of reactors will be limited since the process results in large volumes of contaminated water and is relatively slow technique because multiple passes may be required as the operator is unable to view the cutting field and rebar may not be cut in the first pass. The cost of the filtration system adds to the high cost of the intensifier, making the overall process fairly expensive [523].

---

[520] LaGuardia, T.S. (1980). Concrete decontamination and demolition methods (PNL-SA--8855). United States.

[521] Decontamination and Demolition of Concrete and Metal Structures During the Decommissioning i of Nuclear Facilities, IAEA TECHNICAL REPORTS SERIES No. 286, Vienna 1988

[522] Decommissioning handbook by U.S Department of energy Office of Environmental Restoration, March 1994

[523] Fr.-W. BACH, R. VERSEMANN, P.WILK, "STATUS AND DEVELOPMENT OF DECONTAMINATION AND DISMANTLING TECHNIQUES FOR DECOMMISSIONING OF NUCLEAR INSTALLATIONS", Proceedings of an International Conference Berlin, 14–18 October 2002

Table 6.6-1 Summary of cutting techniques

Cutting technique	**What is working**	**What is missing**	**Assessment and Possibility for improvement**
Diamond wire saw	- It has been successfully tested for thick concrete and reinforced concrete structures (biological shielding) up to 1000 mm which makes it highly versatile. - It provides flexibility in the operations due to its pulley system, which allows to cut unusual configurations. -Remote cutting can be applied in hazardous, radioactive and underwater environments. - It is able to cut precisely and cleanly with minimal effects on the surroundings. - Production of secondary waste is also low. - It does not produce airborne contamination, thus there is no need for additional purification system such as HEPA ventilation.	-The cutting kerf width results dispersion of contamination. - Cooling water used on the tools can spread contamination. -Additional coring tool is required to bore holes to pass the wire through the loop in order to cut the concrete. - The associated concern of physical hazard caused by mechanical failure. -The transport of contaminant from wire saw, areas along the path of the wire and where drive unit is located.	The thickness is virtually unlimited. However, access to both sides of the concrete wall is required. There is potential for spread of contamination.

Cutting technique	What is working	What is missing	Assessment and Possibility for improvement
Flame and thermic lance cutting	-It is promoted when vibrations to the surrounding area is intolerable. -Slits, holes, or openings can be cut in a wide variety of materials with thermic lance process. - These thermal processes are well suited for cutting irregular surfaces with minimum access.	-High operating temperatures the use of HEPA filters for contamination control, that makes it unsuitable for use in contaminated environment. -The generation of large quantities of heat, smoke and toxic gas reduces the visibility, thus making it not recommended for remote operations.	It is recommended to use only if adequate ventilation is available. Otherwise, it produces large quantities of toxic smoke and gas. Moreover, for underwater operations, large amount of bubbles may impair operator's visibility.
Controlled blasting	-Controlled blasting combined with classic hydraulic hammer is favourable to dismantle inner activated part of the biological shield -This method is well adapted to the cutting of large thicknesses of concrete, even when it is heavily reinforced.	-Reinforcing bars must be cut after fracture. Metal aggregate in heavy concrete slows drilling speed. -The application of blasting method in controlled areas raise a number of safety considerations and regulatory aspects.	Shock and noise levels can be moderated with controls. Contamination can be controlled with blasting mats and fog spray.

Cutting technique	What is working	What is missing	Assessment and Possibility for improvement
Abrasive water jet cutting	-This technique is nonthermal, so it does not create fire hazard. -It can be employed underwater as well, although with a substantial (30-40%) decrease in the cut depth. -It is applicable in the atmosphere as well as under water with a small amount of aerosols produced. -It presents multifunctional uses including for kerfing and delamination tasks. -Remote handling is easy and produces low reaction forces.	-There is large amount of contaminated water during cutting operation. -The amount of manpower required is quite high due to maintenance of the cutting system. -The secondary waste consists of water, abrasives and removed particles of the concrete which needs a well-maintained separation system. -Aerosols emissions can be a concern if proper ventilation system is not present.	Abrasive water jet cutting system can be integrated for the cutting of concrete, embedded reinforcing bars, and lead tubes present in the structure of the biological shield. This results in a higher cutting efficiency in total.

6.6.2 Experiences/Case studies

Table 6.6-2 Experiences related to the biological shield segmentation

Cutting technique	Project	Description	Year
Diamond wire saw	La Crosse (USA)	Cutting was proceeded from the top down and biological shield wall was segmented into 20 ton sections [524].	1987
	JPDR (Japan)	To cut the concrete and reinforcing bars of biological shield, diamond wire saw and coring system was used remotely [525].	1990
	Big Rock Point (USA)	To access the activated concrete, the biological shield concrete was removed. This concrete was cut in 50 pieces, each weighing approximately 40,000 lbs from top to bottom [526].	1997
	KRR-2 (South Korea)	The technologies of a core boring, diamond wire sawing and hydraulic crushing were applied. In total, 1913 tons of concrete was dismantled. Among the dismantled waste, 13.2% of the concrete waste was classified as radioactive waste [527], [528].	1997
	ASTRA (Austria)	The biological shield was divided into blocks of between 7 and 9 tons using Diamond wire cutting. Among 1580 tons of removed concrete waste, only 26.5 ton (1.7%) was classified as radioactive waste [529].	2005

[524] Cheol-Seung Cheon and Chang-Lak Kim, "The Dismantling and Disposal Strategy of a Biological Shield for Minimisation of Radioactive Concrete Waste During Decommissioning of a Nuclear Power Plant" September 18, 2017.
[525] Tetsuo HASEGAWA, Makoto ICHIKAWA, Seishi SUZUKI, "Design of machine for dismantling biological shield concrete"
[526] Cheol-Seung Cheon and Chang-Lak Kim, "The Dismantling and Disposal Strategy of a Biological Shield for Minimisation of Radioactive Concrete Waste During Decommissioning of a Nuclear Power Plant" September 18, 2017.
[527] Cheol-Seung Cheon and Chang-Lak Kim, "The Dismantling and Disposal Strategy of a Biological Shield for Minimisation of Radioactive Concrete Waste During Decommissioning of a Nuclear Power Plant" September 18, 2017.
[528] Seungkook Park, S.B Hong, K.W Lee, U.S Chung, J.H Park, K.W Cho, "Dismantling the bio-shielding concrete of KKR-2", 2006.
[529] Franz MEYER and Ferdinand STEGER, "DECOMMISSIONING OF THE ASTRA RESEARCH REACTOR DISMANTLING OF THE BIOLOGICAL SHIELD", September 22, 2006

Cutting technique	Project	Description	Year
	Jose Cabrera NPP (SPAIN)	The segmentation plan considered the cutting activities with drilling and diamond wire techniques and the extraction of eight pieces (7,2 m * 2,3 m * 1,35 m) from the primary shielding of Jose Cabrera NPP. The cutting sequence included horizontal and vertical cuts (radial and perimetral) to release each piece. Anchorage plates were placed in each piece before initiating extraction operations using the original crane of the reactor building. The eight octagonal extracted pieces (30 tonnes each) were placed in a cutting table located in the reactor cavity in a horizontal position for additional cuts in order to segregate ILLW and VLLW for optimising radioactive waste management.	2016
Abrasive water-jet	JPDR (Japan)	The dismantling was done from bottom to top of the lower protrusion of the biological shield concrete. It was controlled remotely [530].	1990
	VAK Kahl (Germany)	The first application of an abrasive water suspension jet was at NPP VAK in Kahl, Germany. The maximum water pressure of 200 MPa was applied to cut the plate thicknesses up to 132 mm [531].	2006
Controlled blasting	Elk River Reactor (USA)	To demolish the 8 foot thick steel-reinforced radioactive biological shield [532].	1974

[530] Tetsuo HASEGAWA, Makoto ICHIKAWA, Seishi SUZUKI, “Design of machine for dismantling biological shield concrete
[531] Bernd Truetsch, “With Siempelkamp back to a greenfield (VAK) Kahl”
[532] Handbook on decommissioning of nuclear installations. European commission, Luxembourg 1995.

Cutting technique	Project	Description	Year
	JPDR (Japan)	Reduced dose rate (0.03 mSv/h) allowed the use of controlled blasting to dismantle the remaining activated concrete [533].	1990
	KKN Niederaichbach (Germany)	The soft explosion technique in combination with the electrically operated hydraulic excavator with a rock chisel, was employed for the removal of the biological shield of KKN [534].	1995

[533] M.Yokota, Y. Seiki and H. Ishikawa (1990); "Experience gained in dismantling of the Japan demonstration reactor (JPDR) "

[534] L. Valencia, E. Prechtl, "Back to the 'green field': the experience and the results gained from the decommissioning of the Niederaichbach nuclear power plant (KKN) ", Nuclear Engineering and Design 170 (1997) 125-132.

6.7 In situ decontamination of building surface (concrete)

In the process of D&D, continuous efforts are being made to minimise the quantity of radioactive waste being generated. Some urgency exists to develop more efficient and effective decontamination and remediation methods to minimise the production of wastes and to optimise recycling and reuse of materials. The in situ surface clean-up process is very demanding in terms of management, manpower, technical equipment and the specifics of the material or the quantities of materials to be decontaminated. Although non-mechanical techniques are available for decontamination of building structures made of concrete, mainly mechanical decontamination methodologies will be considered in this section. Mechanical or physical decontamination techniques can be divided into surface cleaning techniques and surface removal techniques. Surface cleaning techniques include brushing, wiping, flushing, vacuuming, and use of strippable coatings, where on the one hand the surface remains intact but contamination on the surface (including within paint and coatings) is extricated, on the other hand a coating can have a chemical decontamination aspect, to also remove the surface Alternative removal techniques include grinding, blasting, scabbling, shaving, spalling, hammering, and scaling, have an alternative decontamination mechanism where the contamination is detached by virtue of the removal of a layer of the targeted substrates surface. No matter which decontamination technique is used, any waste produced has to have a previously defined disposal route. Also that it should be ALARP to perform the decontamination and not generate more waste in the process. Moreover, the decision on the decontamination of building structures depends on cost-benefit analysis considering the potential concerns of packaging, shipping, and burial costs when using the surface removal techniques.

6.7.1 Description of techniques

6.7.1.1 Surface cleaning techniques

These physical decontamination techniques remove loose dust and particulate from surfaces. Dry vacuuming is performed using a high-efficiency HEPA filter as a pre-treatment for removing large quantities of loose contaminants. The HEPA filters trap dust and debris to protect against airborne contamination and to prevent recontamination of the air and surfaces just vacuumed. Depending on the nature of the contamination, the dry vacuuming process often occurs in a containment structure. The floors, walls, and ceiling are often one piece or are sealed to prevent the escape of contaminants. Tents are employed having zippered doors. The primary waste stream usually consists of a dusty mixture of concrete and other components, the vacuum packing system is supported by waste drums. When the drums are full, they are sealed and immediately ready for safe disposal. The particles removed must be disposed of in a landfill appropriate for the specific characteristics of the contaminants[535].

6.7.1.1.1 Dusting /brushing/vacuuming/ wiping/ scrubbing

These physical decontamination techniques remove loose dust and particulate from surfaces. Dry vacuuming is performed using a high-efficiency HEPA filter as a pre-treatment for removing large quantities of loose contaminants. The HEPA filters trap dust and debris to protect against airborne

---

[535] IAEA, Technical report series no. 401, "METHODS FOR THE MINIMISATION OF RADIOACTIVE WASTE FROM DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES", Vienna, 2001.

contamination and to prevent recontamination of the air and surfaces just vacuumed. Depending on the nature of the contamination, the dry vacuuming process often occurs in a containment structure. The floors, walls, and ceiling are often one piece or are sealed to prevent the escape of contaminants. Tents are employed having zippered doors. The primary waste stream usually consists of a dusty mixture of concrete and other components, the vacuum packing system is supported by waste drums. When the drums are full, they are sealed and immediately ready for safe disposal. The particles removed must be disposed of in a landfill appropriate for the specific characteristics of the contaminants[536].

6.7.1.1.2 Strippable coating

These coatings are typically polymer-based, coating materials that can be applied to a surface covered with contaminated debris, dust and removable particulate. These coatings are used both as protective coverings to prevent contamination of the permanent surface and as a means of removing contamination[537]. The coatings are best applied by brush, and two or more coats are usually necessary to insure that the material has sufficient tensile strength to be removed from the surface. Curing occurs within 24 hrs[538]. Water-based coatings are available that can reduce the risk of organic vapour release. They help to reach on fairly complex geometries[539], [540].

6.7.1.1.3 Ultra high pressure water jets

The technique is known by a variety of names depending on the pressure range being used. Common terms include water flushing (low pressures), hydroblasting, hydraulic blasting, hydrolasing (up to about 15,000 psi), high-pressure water jetting, ultra- high-pressure water jetting, and water jet cutting (up to about 50,000 psi). The increase in pressures and flow-rates enhance the mechanical effects of the water stream to remove the strongly bonded particulates or trapped in surface blockings and also allow other surface material such as paint layers, coatings, galvanized layers from sheet steel and tenacious deposits and other debris to strip off. Recirculation and treatment systems can also be used to minimise secondary waste production[541].

6.7.1.1.4 Abrasive cleaning

The technique is known by a variety of names depending on the pressure range being used. Common terms include water flushing (low pressures), hydroblasting, hydraulic blasting, hydrolasing (up to about 15,000 psi), high-pressure water jetting, ultra- high-pressure water jetting, and water jet cutting (up to about 50,000 psi). The increase in pressures and flow-rates enhance the mechanical effects of the water stream to remove the strongly bonded particulates or trapped in surface blockings and also allow other

---

536 US EPA, “Technology Reference Guide for Radiologically Contaminated Surfaces”, April 2006.
537 BERNADA, O.A., FILEVICH, A., “Fast drying Strippable Protective Cover for Radioactive Decontamination”, Health Phys. 19 (1970)
538 US NRC, « Decontamination processes for Restorative operations and as a precursor to decommissioning », J. L. Nelson, J. R. Divine, May 1981.
539 IAEA, Technical report series no. 395, “STATE OF THE ART TECHNOLOGY FOR DECONTAMINATION AND DISMANTLING OF NUCLEAR FACILITIES”, Vienna, 1999.
540 US EPA, “Technology Reference Guide for Radiologically Contaminated Surfaces”, April 2006.
541 IAEA, Technical report series no. 395, “STATE OF THE ART TECHNOLOGY FOR DECONTAMINATION AND DISMANTLING OF NUCLEAR FACILITIES”, Vienna, 1999.

surface material such as paint layers, coatings, galvanized layers from sheet steel and tenacious deposits and other debris to strip off. Recirculation and treatment systems can also be used to minimise secondary waste production[542].

Similarly, sponges are also used in which abrasive particles are embedded and comes with different grades of abrasiveness. When impacting on a surface, the sponge gives scrubbing effect by contracting and expanding and decontaminates paint, dirt, and oil from the surface.

6.7.1.1.5 Summary

Table 6.7-1 Summary of techniques identified for in-situ decontamination of building surface (concrete)-Surface cleaning techniques

Technique	**What is working**	**What is missing**	**Assessment and Possibility for improvement**
Dusting /vacuuming/ wiping/ scrubbing	-Applicable to various contaminants like lead based paint chips, PCB's, and asbestos. -Remote operation for these activities is also possible.	-Scrubbing should be avoided on porous or absorbent materials because loosely deposited materials may be pushed deeper into the surface -Difficulties to remove contamination from cervices	These techniques are best suited for smooth surfaces.
Strippable coatings	-They produce a single solid waste and are used where airborne contamination has to be avoided. -They can be used to mitigate other hazardous wastes including PCBs, asbestos.	-Application and removal times are relatively long in some instances. -Cost of the material is high and number of coats are required for better results.	Coatings left in place for extended periods are often difficult to remove. Normally, a single layer of cheese cloth set underneath the coating alleviates this problem

[542]IAEA, Technical report series no. 395, "STATE OF THE ART TECHNOLOGY FOR DECONTAMINATION AND DISMANTLING OF NUCLEAR FACILITIES", Vienna, 1999.

Technique	What is working	What is missing	Assessment and Possibility for improvement
Ultra high pressure water jets	-It is non-specific with regard to contaminant removal that makes it versatile. -Frequently used on difficult-to-access surfaces through use of lances or extensions to the nozzles or through integration into robotic and remotely operated systems.	-Not recommended for fixed and non soluble contamination present in cracks. (previous bad experiences e.g. at AT1 reprocessing plant in La Hague, France where water produced migration of contamination deeper in the concrete) -Large volumes of waste water produced unless recycling possible.	This technique is able to completely remove contamination by removing the layers of base material in which it is contained.
Abrasive cleaning	-This process is most effective on flat surfaces and can also be used for ‘hard to reach’ areas such as ceilings or behind equipment. -HEPA filters can be used that reduces the danger of air borne contamination.	-Labor costs are high as it is relatively slow and labor intensive technique. -There is a potential of cross contamination in this technique.	The system uses no soluble or hazardous chemicals which increases its usability. But needs to consider dust generation and secondary waste in terms of wet techniques.

6.7.1.2 Surface removing techniques

In this technique, the contamination is removed by the removal of an entire layer of the surface between 1-3 mm[543]. Surface removal techniques are used for future land-reuse scenarios and when it is impractical to demolish the building or because of waste minimisation. Aggressive techniques including grinding, spalling and drilling, scarifying techniques, shaving/milling, hammering, scabbling, high-frequency microwaves, laser and induction heating. The use of most of these techniques is limited to specific applications in specific cases. Some of them have disadvantages such as spreading of contamination or produce a lot of undesirable secondary waste. These techniques are discussed briefly [544].

6.7.1.2.1 Concrete grinding/shaving

This technique uses concrete grinder containing a diamond grinding wheel to decontaminate and strip concrete surfaces. The light-weight and hand-held device creates a smooth surface when applied to flat or slightly curved concrete surfaces. It produces little vibration. To make the grinding process effective, a vacuum attachment is introduced that removes dust created by the grinding process. The attachments for dust collection shroud can be designed in a way to attach the vacuum hose of on-site HEPA filtration system. Decontamination rates from 4 to 6 m^3/h machine working time are obtained from this device [545].

6.7.1.2.2 Concrete scarifying

This technique works by abrading coated or uncoated concrete surfaces. It can remove multiple layers of contaminated surfaces until the required depth is achieved with no surface contamination. State of the art technologies are introduced for scarifying techniques that provide the desired profile for new coating systems if the facility should be released for unconditional use [546]. This process is used with the help of following devices;

- **Needle scaling**

This is a scarifying technique that will chip off contamination from a surface. Needle scalers are pneumatically driven and use uniform set of 2, 3, or 4 mm needles to obtain a desired profile and performance. The reciprocating action from the sets of needles helps to decontaminate the concrete surface. Dust and debris removed during the process is collected by specialised shroud and vacuum attachments. They can remove surface contamination from concrete up to 0.5 in thickness [547], [548].

---

[543] NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

[544] IAEA, Technical report series no. 230, "DECOMMISSIONING OF NUCLEAR FACILITIES: DECONTAMINATION, DISASSEMBLY AND WASTE MANAGEMENT", Vienna, 1983.

[545] US EPA, "Technology Reference Guide for Radiologically Contaminated Surfaces", April 2006.

[546] IAEA, Technical report series no. 395, "STATE OF THE ART TECHNOLOGY FOR DECONTAMINATION AND DISMANTLING OF NUCLEAR FACILITIES", Vienna, 1999.

[547] Handbook on decommissioning of nuclear installations. European commission, Luxembourg 1995.

[548] Åke Anunti, Helena Larsson, Mathias Edelborg Westinghouse Electric Sweden AB, "Decommissioning study of Forsmark NPP", June 2013.

- **Scabbling**

This method is also used to remove concrete surfaces. It incorporates several pneumatically operated piston heads which strike and chip a concrete surface. Mostly, the tips of striking heads are made of tungsten carbide. They are both pneumatically and electrically driven machines. Currently, scabblers range from one to three headed hand-held scabblers to remotely-operated scabblers. The most common versions have three to seven scabbling pistons mounted on a wheeled chassis that is manually pushed across a surface. Due to cross-contamination hazard, vacuum attachments and shrouding configurations have been incorporated, which also facilitates decrease in airborne exposures. In principle, floor scabblers may only be moved within some 5 cm of a wall. Other hand-held scabbling tools are therefore needed to remove the last 5 cm of concrete flooring next to the wall [549].

6.7.1.2.3 Concrete milling

The concrete shaver is an electrically driven, self-propelled system that is capable of removing contaminants from large-areas concrete floors. The mechanism contains a cutting head with a rotating milling drums with embedded diamonds. The rotating tool tips and cutters hit the surface at high speed which peel off the surface It is controlled by the operator from the handles and moves along the untreated surface. Commercially available concrete shavers are well suited for large, wide-open concrete floors and slabs. The presence of the vacuum filtration system reduces the issue of large amounts of dust contamination.

Production rates depends on the structure and the hardness of the concrete, water content, the depth setting, the cutting speed, and the type of diamond used. Heads can be used for shaving up to 2 000 m². Milling drums are equipped with various tools that depends on the material being removed. Cutters, toothed, and start-shaped milling rings can be attached to produce the finely structured surface [550]. Different automation concepts to induce the milling head on different machines have been developed in order to facilitate the decommissioning projects [551]. These include:

- The milling head gguiding and driving system is fixed on the surface to be treated.
- Milling heads are applied and moved along the surface with the aid of a fork lifter.
- Milling heads are applied using a compact heavy duty carrier.
- For the vertical movement, milling heads are fixed on a horizontal linear rail, under a scaffolding system.

6.7.1.2.4 Drilling and Spalling

This technique removes contaminated concrete surfaces without demolishing the entire structure. Holes are drilled in the concrete surface to be decontaminated in a honeycomb pattern. The spaller bit is inserted into a drilled hole. The bit expands in the hole with the help of hydraulic pump causing the spalling. Chunks of concrete resulting from the spalling are up to 5 millimeters thick and 18 to 41

---

[549] Handbook on decommissioning of nuclear installations. European commission, Luxembourg 1995.
[550] US EPA, “Technology Reference Guide for Radiologically Contaminated Surfaces”, April 2006.
[551] NEA/OECD, Radioactive Waste Management (rwm-r2011-1), “Decontamination and Dismantling of Radioactive concrete structures”, 2011.

centimeters in diameter. The spaller is provided with a metal shroud to capture these concrete chunks. A detachable shroud includes a vacuum port which will allow a hose to connect to an on-site HEPA filtration system to control the dust. Concrete spalling can be used to decontaminate interior and exterior, flat or slightly curved concrete surfaces. It is particularly used on floors and walls and can work around piping or reinforcements embedded in the concrete [552].

6.7.1.2.5 Laser ablation

This technique is based on scanning the surface of the material with a focussed laser beam, which causes ablation of the surface by micro-explosions. It takes advantages of progress made by the fiber lasers which have now a lifetime longer than 20,000 hours without maintenance and allow remote operation.

The process[553] was initially developed for painted concrete surfaces to optimise the quantity of waste produced and to avoid use of water or other abrasive, using a low average power laser which does not need any particular utility (simply 220V socket outlet). It could thus achieve the selective removal of surface coatings such as epoxy paints, adhesives, corrosion products, accumulated airborne pollutants.

The choice of the laser parameters limits the ablation to the layer of contaminated paint (which is only a few hundredths of a micrometre deep, if there is a protective paint present), without causing any damage to the substrate. All the ablated material is collected via a suction/confinement system which has a cleanable cartridge filter, HEPA filters and activated carbon filters, which prevent redeposition on the surface. But if needed more powerful sources of energy can be used in order to remove much more concrete and gain in speed. Energy sources for this application are commercially available [554].

6.7.1.2.6 Microwave scabbling

This technique uses microwaves to heat and induces volume expansion of the water contained in the concrete pores. The heat cannot dissipate as fast as the expansion proceeds. This results increased tension in the matrix structure which in turns induces spalling of the concrete. The analysis of the research on this technology highlighted that the main factors affecting scarification are the pore dimensions and the evaporable water content of the cement. It can be concluded that this is a reliable apparatus but should be further developed to improve its flexibility and ease of operation [555].

It has also been considered for the removal of coatings but also for stripping and delamination of thin mineral or an organic layer. This technique has been investigated in the recent past but showed poor efficiency and was found to be incompatible with the constraints and safety requirements of a decommissioning projects[556]

---

[552] US EPA, "Technology Reference Guide for Radiologically Contaminated Surfaces", April 2006.
[553] Aspilaser® process, CEA patent.
[554] NEA/OECD, Radioactive Waste Management and Decommissioning, "Preparing for Decommissioning During Operation and After Final Shutdown", 2018.
[555] NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.
[556] NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

6.7.1.2.7 Summary

Table 6.7-2 Summary of techniques identified for in-situ decontamination of building surface (concrete)-Surface removing techniques

Technique	What is working	What is missing	Assessment and Possibility for improvement
Grinding/Shaving	-The technology typically requires one person for operation and is considered effective for the decontamination and stripping of concrete. -Generally preferred for the radiological or hot spot decontamination of concrete surfaces.	-It is not convenient for large surfaces. -The disks are very sensitive to the presence of metallic inserts on the surface. -Not suitable for rough surfaces and their production rates are strongly reduced on uneven surfaces (concrete mould).	It usually removes only a thin layer surface contamination from concrete and produces fine dust. The costs are high in terms of consumable disks
Scarifying	-Needle scalers provide expertise in tight and hard to access areas. Moreover, the contaminated waste is easy to handle. -Scabblers are best suited for removing thin layers (up to 15 or 25 mm thick) of contaminated concrete (including concrete block) and cement.	Needle scalers are light tool and only considered for limited areas rather than large areas. -Surface after scabbling is coarsely finished which greatly reduces the accuracy of free release measurements and may even prevent the use of certain techniques.	Refined technologies for scarifiers are available which are reliable and provide the desired profile for new coating systems.

Technique	What is working	What is missing	Assessment and Possibility for improvement
Milling	-They are effective in removing radiological contaminants and paints. -Majorly used on large, open, and horizontal surfaces of concrete.	-Due to physical size and geometry, it is not appropriate for the use on small concrete floors and slabs or with significant obstructions.	The maximum working depth of a concrete shaver is 5 mm which also favours the cross contamination. This can be avoided by combining radiometric devices to enable 'smart' surface removal. It facilitates more material removed from more contaminated areas and minimises waste.
Drilling and Spalling	-It is effective in large areas and a good tool for hot-spots and in-depth decontamination of cracks in concrete. -Preferred over other technologies where rapid decontamination is required at 3 mm or greater depth.	-It results in an uneven surface which can only be demolished. -The spreading of cracks must be controlled.	High decontamination factors are possible with this technique if the contamination is in the top 6 to 8 cm layer of concrete.
Laser ablation	-process is selective and only removes the necessary volume without producing any waste other than that ejected from the surface.	Laser Ablation process is operational and effective to remove painting on flat, non-porous	Yields are very low as it covers very small surface areas. - R&D to be pursued on for: • Surface micro-cracks,

Technique	What is working	What is missing	Assessment and Possibility for improvement
	-Suitable for rough surfaces and insensitive for metal inserts in the concrete. - Totally robotic, automated operation, - Nuisance-free operation enabling other dismantling activities to be carried out at the same time.	surfaces[557],[558],[559] Several areas of R&D work are currently being carried out to extend its scope. The process was industrialised and commercialised by company ASTRANE / SDMS but it suffered lack of feedback on long term reliability and maintenance. Indeed it is not used anymore now-a-days in France although Aspigel® was developed there, because not so many projects are at this stage at CEA and because safety authority guide 14 advices to remove more concrete thickness before unconditional release[560].	• Non-flat surfaces (curves, pipes, etc.), • Greater depths of concrete • Improvement of speed - Need to convince safety authority of the interest to remove only the contaminated layer in order to minimise waste - Active area of research in UK for the treatment of fumes and particulates emerged as a waste during this technique.

6.7.2 Experiences/Case studies

Table 6.7-3 Experience/Case studies related to concrete decontamination

[557] NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

[558] [558] NEA/OECD, Radioactive Waste Management and Decommissioning, "Preparing for Decommissioning During Operation and After Final Shutdown", 2018.

[559] http://www.atsr-manifestations.fr/cariboost_files/s6-Pr_C3_A9sentation_20AspiLaser_20ATSR_202010.pdf

[560] Guide 14: This guide lays out ASN's recommendations regarding the remediation methodology to be used by licensees. http://www.french-nuclear-safety.fr/References/ASN-Guides-non-binding/ASN-Guide-No.-14

Techniques	Projects	Description	Year
Vacuuming/scrubbing	Risø (Denmark)	The concrete hot cells were remotely vacuumed before further decontamination took place [561].	1993
	Chernobyl NPP	Specially designed vacuum cleaners incorporating air filtration systems were widely used [562].	1989
Scabbling	BR3 (Belgium)	Multi-headed hand-held scabblers have been used extensively during the decontamination of the auxiliary building demineralisation cells. Production rates (machine working time) of up to 1 m²/h have been reported at a scabbling depth of 3 mm [563].	2008
	Eurochemie (Belgium)	Five to seven-headed scabblers were used for floor decontamination, while one and three-headed hand-held types were used for the decontamination of concrete walls and ceilings[564].	2004
Grinding	JPDR (Japan)	Diamond tipped hand-held grinders have been used for the decontamination of floors and walls.	1995
	Vandellos 1 (Spain)	Hand-held grinders equipped with a disk of diamond segments bonded onto the face of the disk were used for the decontamination of graphite silos. It was provided with a controlled dust extraction and produced very low vibrations[565].	2000

[561] CARLSEN, H., et al. "Decommissioning of the Riso hot cell facility", Decommissioning of Nuclear Installations (Proc. 3rd Int. Conf. Luxembourg, 1994), Office for Official Publications of the European Communities, Luxembourg (1995)135–143.
[562] IAEA, Technical report series no. 395, "STATE OF THE ART TECHNOLOGY FOR DECONTAMINATION AND DISMANTLING OF NUCLEAR FACILITIES", Vienna, 1999.
[563] NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.
[564] NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.
[565] J.L. Santiago, F. Madrid, "THE DECONTAMINATION AND DECOMMISSIONING OF THE GRAPHITE SILOS AT VANDELLOS 1", 2002.

Techniques	Projects	Description	Year
Shaving/Milling,	ATUE (France)	2 headed milling machine on fork lift making ceiling decontamination possible[566].	2012
	EL4, Brennilis (France)	Milling machine on Brokk carrier was used for wall shaving [567].	2004
	AT1, Melusine (France)	PLB milling head with WC teeth was used for ceiling shaving. The tool was heavy and provided rough finishing [568].	2001
	Brennilis (France)	Floor shaver, Multi-disc rotary head with diamond tipped rotating tip was introduced for floor shaving. It provided 3mm depth shaving[569].	2004
Abrasive blasting	BR3 (Belgium)	Sponge-Jet technique in which PU foam embedded with alumina were used to remove paint (< 1mm) [570].	2008
	ATUE (France)	Shot-blasting using steel grit as an abrasives were introduced to remove thin layer of concrete [571].	2012
	AT1, Melusine (France)	Shot-blasting using steel grit as an abrasives were introduced to remove thin layer of concrete [572].	2001
Laser ablation	Removing painting only to minimise	Ablation rate was a few micrometers per pass (between 5 and 10 µm, depending on the paint). For typical layer thicknesses (30 to 300 µm) it needs between 3 and 30 passes to	2010

566 NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

567 NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

568 NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

569 NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

570 NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

571 NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

572 NEA/OECD, Radioactive Waste Management (rwm-r2011-1), "Decontamination and Dismantling of Radioactive concrete structures", 2011.

Techniques	Projects	Description	Year
	waste at ATUE (France)	reach the substrate. The collecting system collects all the removed matter (gases and aerosols) on nuclear grade filters. The process was then industrialised and commercialised by company SDMS.	

6.7.2.1 Italy – Garigliano Stack Decontamination (2014-2017)

In Garigliano NPP the old stack didn't comply with the local seismic requirements anymore, so there was the need to demolish it and to build a new one.

To reduce the wastes produced because of the demolition, a preliminary decontamination of the inner surface of the stack was carried out.

Taking into account both the radiological and the conventional hazards to the workforce, primarily related to the difficulties to operate at heights and in a confined contaminated environment, it was decided to develop a remote decontamination system robotically operated. Scarifying was the chosen technique to remove the contaminated surface layers. The base concept tool to abrade and remove layers was taken from a commercially available scarifier.

This tool is equipped with cutters fitted on a shaft which are then placed inside a drum housing. During the rotation of the drum, the generated centrifugal forces "throws" the cutters at the concrete surface causing a mechanical cutting action.

Starting from this type of tool, the challenge was to develop and construct a completely automated robotic system able to perform an efficient cutting and milling actions on the concrete surface.

Finally, it was developed a frame housing all the circuits needed for the operation of the system. The equipped frame was called "Robotic Shuttle".

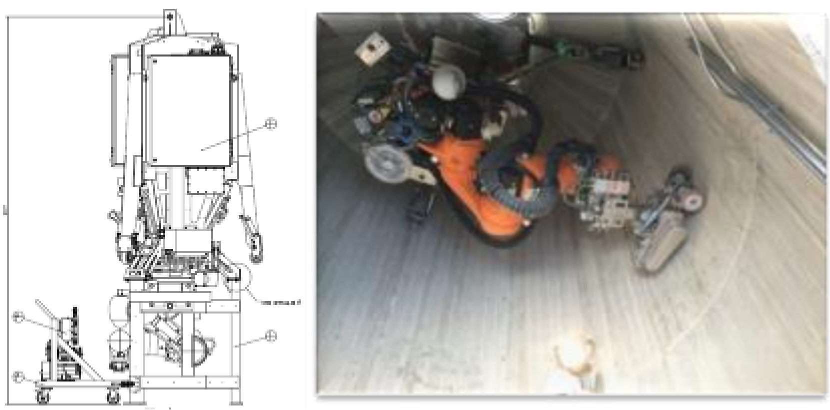


Figure 6.7-1 Garigliano robotic system for stack decontamination and sampling

The stack has been closed on the top to avoid dust dispersion. The robot going down inside the stack performed the scarification of the inner wall and was also equipped with a proper tool to take samples after the scarification. If the analysis on the sample showed that no contamination was left, then the robot continued with the scarification of the next step, otherwise it repeated the scarification of the same portion.

At the end of the activity, all the inner surface was scarified, the stack was released with no radiological constraints, and it has been demolished with conventional technique.

6.8 Management (characterisation, decontamination, removal) of radiological embedded elements

The waste of radioactive power plants is diverse and varied in nature and it encompasses a broad range of radionuclides, half-lives, activity concentrations, volumes and physical and chemical properties. Other than radionuclides, the waste may contain other hazardous elements (i.e. asbestos). Therefore, the management of the radiological embedded elements, when dismantling a radioactive power plant, is quite complex. "IAEA estimates the mass (rather than volume) of the decommissioning waste: a light-water reactor of 1 GW capacity can be expected to produce 5,000 to 6,000 tons of LILW and 1,000 tons of long-lived LILW and HLW.[573]

The management of the radiological embedded elements include the supervision of the processing steps from their detection on side, the monitoring during the dismantling process up to their removal. This Chapter overlap in parts with Chapter 8 "management of material and radioactive waste from decommissioning". For information on managing waste routes and the managing on materials and processes during decommissioning refer to this Chapter.

Chapter 4 on characterisation is closely related to this topic but does not include a specific subtopic on management aspects, there for some aspects to consider are given at this point.

6.8.1 Characterisation and classification of waste

Radioactive waste needs characterisation several times during the predisposal process. To decide on the appropriate safety procedures and organise suitable solutions for conditioning, transportation and short-term storage, the following points should be considered:

- The origin of the waste, the waste type and the physical state of the raw waste (solid, liquid or gaseous).
- The criticality risk.
- The radiological properties of the waste (e.g. half-life, activity and concentration of radionuclides, dose rates from the waste, heat generation).
- Other physical properties (e.g. size, mass, compatibility).
- Chemical properties (e.g. the composition of raw waste, water content, solubility, corrosiveness, combustibility, gas generation properties, chemical toxicity).
- Biological properties (e.g. biological hazards associated with the waste).
- Intended methods of processing, storage and disposal**Erreur ! Signet non défini.**

With a good knowledge of waste properties, it is possible to segregate the waste for treatment and conditioning. Documented procedures should be followed for uniform characterisation and segregation and waste should be designed according to the documented categories. Special attention should be given to the waste containing flammable, pyrophoric, corrosive or other hazardous materials and be stored separately.

[573] The World Nuclear Waste Report. Focus Europe. 2019. Berlin & Brussels; www.worldnuclearwastereport.org

6.8.2 International Reference Documents

Document	Discussed Topics related to Management of Radiologically Embedded Elements
IAEA-TECDOC-1817[574] Selection of Technical Solutions for the Management of Radioactive Waste	• Waste Characteristics • Treatment option • Methodologies for technical selection
IAEA Safety Standards Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors Specific Safety Guide No. SSG-40[575]	• Safety considerations • Management Systems • General considerations • Examples on Management systems for specific applications

[574] IAEA-TECDOC-1817; SELECTION OF TECHNICAL SOLUTIONS FOR THE MANAGEMENT OF RADIOACTIVE WASTE; INTERNATIONAL ATOMIC ENERGY AGENCY; VIENNA, 2017

[575] IAEA, "Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors", IAEA Safety Standards Series, Specific Safety Guide No. SSG-40, Vienna (2016).

6.9 Demolition of large, reinforced concrete structures

Concrete dismantling and demolition techniques are used whenever large quantities or deep layers of activation and contamination is needed to be removed. In nuclear reactors, the strongly reinforced concrete present in the shield structure that surrounds the reactor becomes radioactive in-depth due to neutron activation. It is also possible that radioactive contaminants penetrate deep into the pores or deep cracks in the concrete surfaces throughout the facility and cannot be removed by just using the surface decontamination techniques. The removal of these large segments of concrete is not unique to nuclear plants and many of the techniques are modified versions of those used in the non-nuclear industry. However, the volume of concrete, reinforcements present, and the radioactivity make the demolition a difficult task. Mostly, the demolition equipment for nuclear facilities is operated remotely if the dose rates are higher. In the decommissioning project, major concrete structures that need to be cut or demolished are categorised as [576];

- Heavily reinforced and massive concrete structures that are used to construct the biological shields, walls of hot cells, and foundations in a reactor.
- Large and heavy concrete (metal or magnetite aggregate) with little or no reinforcements which are also used for certain biological shields.
- Lightly reinforced or non-reinforced floors and walls.
- Pre-stressed reactor buildings and vessels.

The volume of concrete coupled with significant reinforcement represents a formidable dismantling task. Reinforced concretes typically encountered include biological shields which maybe 2 to 10 ft thick standard (140-150 lb/ft^3) or high-density concrete (magnetite or metal aggregate, 250-325 lb/ft^3). Reactor base-mats or facility footings can be as much as 25 ft thick [577]. The technologies and methodologies aligned with the cutting and demolition of reinforced concrete are already discussed in the previous chapter "Segmentation of interior concrete structures (e.g. bioshield)". These techniques are listed as:

- Diamond wire sawing;
- Flame cutting;
- Controlled blasting/ Explosive cutting;
- Abrasive water jet cutting.

For the structures that are lightly reinforced but still belongs to the biological shield walls and surroundings can be demolished using the techniques described in the following paragraph which are modified versions of those used in non-nuclear industry.

[576] Decontamination and Demolition of Concrete and Metal Structures During the Decommissioning i of Nuclear Facilities, IAEA TECHNICAL REPORTS SERIES No. 286, Vienna 1988.
[577] LaGuardia, T.S. (1980). Concrete decontamination and demolition methods (PNL-SA--8855). United States.

6.9.1 Description of techniques

6.9.1.1 Impact crushing technique

This technique has been used extensively on many decommissioning applications because of its versatility and low cost. It is preferred for the situations where the care and precision of diamond wire sawing are not required. It uses a combination of impact hammers (jackhammers or pneumatic drills) and concrete breaking jaws that are typically mounted on small excavators of Brokk demolition machines. The impact hammer usually contains a chisel point and impacts the surface to be demolished at rates of up to 600 blows per minute delivering up to 2,700 Nm (~2,000 ft.lb) force per blow [578]. It automatically monitors and adjusts its output shock energy and shock frequency characteristics according to the crushed objects. When a solid structure is broken, the single impact energy is automatically increased and the impact frequency is reduced to make it more capable of breaking. This protects the hammer and extends its service life.

Concrete breaking jaws can also be used where there is suitable access to the edge of a wall to allow the jaws to work. The issues related to noise pollution and dust generation, which lead to airborne contamination needs to be considered when using these techniques. The impact on personnel can be mitigated through the use of suitable personnel protective equipment and the use of water mist/sprays to reduce dust. The production rates depending on issues such as accessibility and radiological conditions are achievable using concrete breaking hammers and jaws. The correlation weight between the excavator and the hammer should be in the range of 15:1 and 20:1. The performance in terms of demolition rate is highly dependent on the type of hammer, the quality of the concrete, the amount of the reinforcement, and the operator skill [579].

6.9.1.2 Wrecking ball or slab

Wrecking ball or slab is one of the oldest and most commonly used methods for building demolition. A crane uses a wrecking ball, typically weighing from 1,000 lb to 13,500 [580]. The ball is either dropped onto or swung into the element to be demolished. It is typically used for demolishing non-reinforced or lightly reinforced concrete structures that are less than 3 ft thick. For the maximum control, ball -dropping method is preferred as this method gives good fragmentation after impact. The flat slab was devised to be used in the vertical drop mode, as it offers the advantage of being able to shear through steel reinforcing rods as well as concrete. The wrecking ball or slab is recommended for non-radioactive concrete structures. It would be impossible to control the release of radioactive dust during demolition

---

[578] Åke Anunti, Helena Larsson, Mathias Edelborg Westinghouse Electric Sweden AB, “Decommissioning study of Forsmark NPP”, June 2013.

[579] JingZhu, Wenzhong Zheng, Lesley H Sneed, Chonghao Xu& Yiqiang Sun,”Green Demolition of Reinforced Concrete Structures: Review of Research Findings», Global Journal of Researches in Engineering: E Civil And Structural Engineering, Volume 19, Issue 4, Version 1, 2019.

[580] JingZhu, Wenzhong Zheng, Lesley H Sneed, Chonghao Xu& Yiqiang Sun,”Green Demolition of Reinforced Concrete Structures: Review of Research Findings», Global Journal of Researches in Engineering: E Civil And Structural Engineering, Volume 19, Issue 4, Version 1, 2019.

due to the access needed for the crane to drop or swing the ball. The wrecking ball is an effective method and provides good fragmentation to expose reinforcing rods for non-radioactive structures [581].

In order to minimise the dust impact on the surrounding areas, the structure to be demolished should be pre-soaked with water before demolition. To ensure safe operation of a crane using a wrecking ball, the National Association of Demolition Contractors provides guidance on the safe operation of a crane using a wrecking ball. The ball weight should not exceed 50% of the safe load of the boom at maximum length or angle of operation, or 25% of the nominal breaking strength of the supporting line [582].

6.9.1.3 Backhoe mounted ram

Backhoe mounted rams are used for concrete structures less than 2 ft thick with light- reinforcement. The method is ideally suited for low noise, low vibration and interior demolition in confined areas. The ram is recommended for applications with limited access for heavy equipment such as a wrecking ball, and where blasting is not permitted. This technique may also be employed after controlled blasting to expose reinforcing rods so they may be cut afterwards. The equipment consists of an air operated or hydraulically operated impact ram with chisel points mounted on a backhoe arm. The ram starts impacting as soon as there is resistance to the point and stops when breakthrough occurs or when the ram head is lifted. With the ram head mounted on a backhoe, the operator has approximately a 20 to 25 ft reach, and the ability to position the ram in limited access structures.

Dust and contamination control is maintained with water fog sprays before and during breaking activities. However, the spray should be synchronized with the ram head to avoid excessive use of water [583].

6.9.1.4 Rock splitter

The rock splitter is a method used for fracturing concrete by hydraulically expanding a wedge into a pre-drilled hole until tensile stresses are large enough to cause a fracture. The splitter is ideally suited for fracturing concrete in limited access areas where large air rams cannot operate. The process is relatively quiet except for drilling holes and is used extensively for demolition near densely populated areas [584].

In order to deal with long sections of concrete, multiple splitters are used along the desired fracture line. The tool consists of a hydraulic cylinder that drives a wedge-shaped plug between two expandable guides (termed ‘feathers’) inserted into the drilled holes. The unit is powered by a hydraulic supply system which operates at 50 MPa. The hydraulic unit may be powered by air pressure, petrol engine or electric motor sources. There are units available that can develop splitting forces approaching 3.2 MN. The maximum lateral expansion of the feathers is approximately 2 cm. Further concrete may be separated at a fracture line using a backhoe mounted ram or similar equipment. The reinforcing rod in reinforced concrete must be cut before separation is possible. Additional holes and fractures would be necessary to expose the rod

---

[581] Decommissioning handbook by U.S Department of energy Office of Environmental Restoration, March 1994.
[582] JingZhu, Wenzhong Zheng, Lesley H Sneed, Chonghao Xu& Yiqiang Sun,”Green Demolition of Reinforced Concrete Structures: Review of Research Findings», Global Journal of Researches in Engineering: E Civil And Structural Engineering, Volume 19, Issue 4, Version 1, 2019.
[583] IAEA- Technical report series-401, “METHODS FOR THE MINIMISATION OF RADIOACTIVE WASTE FROM DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES”, Vienna 2001.
[584] LaGuardia, T.S. (1980). Concrete decontamination and demolition methods (PNL-SA--8855). United States.

in heavily reinforced concrete. The removal rates of up to 200 m^3/d for non-radioactive concrete have been achieved [585].

6.9.1.5 Core Drilling

Core drilling can be used to remove cylindrical sections of hardest surfaces and cutting through reinforced concrete as well. The coring unit can be used in combination with diamond wire sawing to get ideal results for segmentation of highly reinforced concrete. The core drill comprises a steel tube with diamond segments that are welded to the cutting face which make contact with the surface being drilled. During drilling, the diamond bit rotates. Water is used to both cool the face of the drill and flushes away the material being cut. The adjustment of pressure, rotary speed of the drill bit, and the water circulation depend on factors like; the drilling conditions and type of surface being drilled [586].

To carry out drilling effectively, several factors have to be considered. They include; the type of material and surface, size and diameter of the hole, structural resistance of the material, and the matching suitable diamond compounds to carry out the work efficiently. Cylindrical blocks up to 1 m in diameter can be holed out with this process. Also, the blocks can be sized to fit into a 200 L drum [587].

6.9.1.6 Bristar/ Expansive grout

Expansive grout is a material that can be used to fracture concrete. It is a chemically expanding compound which is poured into pre-drilled holes and causes tensile fractures in the concrete upon hardening. It is a compound of limestone, siliceous material, gypsum and slag. The powdered compound is mixed with water to make a paste. The paste is filled in the holes and left to cure. During curing, the grout expands and causes the concrete to crack between the holes. The factors that need to be considered include; the length and diameter of the holes, the drilling pattern designed to suit the size of the segments required, the shape of the structure, the type of concrete, and the number of steel reinforcements [588].

The compound works against the tensile strength of concrete usually between 200 to 425 psi. This non-explosive and vibration-free process can be used to crack concrete of any size, reinforced or non-reinforced provided it has a free face to which it may expand. The cracks formed propagates along the fracture line. The width of the crack ranges between 0.25 in after 10 hrs to almost 2 in after 15 hrs. The fractured burden can be further removed by demolition hammer, jackhammer, and backhoe. If reinforcing rods are encountered, they must be cut separately [589].

---

[585] IAEA- Technical report series-401, "METHODS FOR THE MINIMISATION OF RADIOACTIVE WASTE FROM DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES", Vienna 2001.
[586] https://www.robore.com/diamond-core-drilling-london-united-kingdom.html
[587] Decontamination and Demolition of Concrete and Metal Structures During the Decommissioning i of Nuclear Facilities, IAEA TECHNICAL REPORTS SERIES No. 286, Vienna 1988.
[588] OECD/NEA, RWM (CPD) project, "Decontamination and Dismantling of radioactive concrete structures", 2011.
[589] Decommissioning handbook by U.S Department of energy Office of Environmental Restoration, March 1994

6.9.1.7 Summary

Table 6.9-1 Summary of demolition techniques

Demolition technique	**What is working**	**What is missing**	**Assessment and Possibility for improvement**
Impact crushing	-When used in combination with excavators, they provide high yield and reliability. -They are insensitive to surface and metal state. -The changeable impact hammers (jackhammers or pneumatic drills) and concrete breaking jaws, typically mounted on the excavators adds to its versatility.	-The weight of the equipment makes its handling difficult. -It produces high amount of dust generation and noise pollution which can lead to airborne contamination. Moreover, vibration levels are on higher side. -Presence of reinforcements requires additional cutting techniques and treatments to reach adequate surface finishing.	It is recommended where control of debris is required.
Wrecking ball	-It is an effective method and provides good fragmentation to expose reinforcing rods for non-radioactive structures. -The safety and simplicity of the technique makes it advantageous.	-Proper control of the swing requires stringent calculations. -It produces large amounts of dust, noise and pollution. -Availability of substantial clear space high clearance.	The maximum drop height is 110 ft and maximum swing height is 50 ft. It is suitable for breaking rubble but not recommended for radioactive structures.

Demolition technique	What is working	What is missing	Assessment and Possibility for improvement
Backhoe mounted ram	It is suited for low noise, low demolition vibrations and for interior confined areas demolition.	It cannot reach the tall structures because maximum reach is 20 ft.	Dust and contamination can be controlled by fog spray. Moreover, additional cutting techniques required to cut the rebar.
Rock splitter	-The process is silent (except for hole drilling) and ideally suited for fracturing concrete in limited access areas. -It is inexpensive, provides accurate control and dismantling precision.	The process is time consuming and requires the use of breakers to expose the rebar.	It is recommended where noise and vibration must be controlled.
Core drilling	When used in combination with diamond wire sawing, it facilitates the removal of reinforced concrete structures with relatively dense steel bars.	It has low efficiency and takes much time.	Core drilling is used where surrounding material must not be disturbed, or where accessibility is limited.
Expansive grout	It can be used for massive non-reinforced concrete structures where noise, vibration, fly rock, dust or gas must be avoided.	-This method is costly and time consuming. -Further investments for cleaning the concrete rubble by backhoe and cutting of rebar is required.	-The rate of demolition depends on hole pattern, hardness of concrete, and orientation of rebar.

6.9.2 Experiences/Case studies

Table 6.9-2 Experiences related to the demolition of large reinforced concrete structures

Demolition technique	Project	Description	Year
Impact crushing	KNK (Germany)	Used successfully for the dismantling of non-radioactive concrete [590].	2004
	KKN (Germany)	The use of electrically operated hydraulic excavator (EX60) with a rock chisel, mounted drill hammer, and 8 ton of stone crusher in combination with the soft explosion was employed for the removal of the biological shield of KKN [591].	1995
	MZFR (Germany)	A remotely controlled electrohydraulic impact excavator modified with rapid-exchange coupling of a hydraulic chisel¸ a concrete mill, a crusher, a scrap cutter, and a hoe dipper was employed for concrete biological shield demolition [592].	2011
	Melusine (France)	On site demolition was done by CEA using Brokk 180 (2t). Production rates for the machine were up to 1.4 m^3/h [593].	2012
	BR3 (Belgium)	Antimissile slabs made of reinforced barite concrete was demolished by Brokk TEX 250 in the workshop. Production rates of the machine were up to 1.2 m^3/h [594].	2014
	BR3 (Belgium)	Reinforced concrete was demolished using Brokk 180 (2.4t) that provided 1750 hits/min in the workshop. Production rates of the machine were up to 2 m^3/h [595].	2014

[590] Iris Hillebrand, Forschungszentrum Karlsruhe GmbH, "Decommissioning KNK - Concept for Dismantling the Reactor Vessel and the Biological Shield"
[591] L. Valencia, E. Prechtl, "Back to the 'green field': the experience and the results gained from the decommissioning of the Niederaichbach nuclear power plant (KKN)", Nuclear Engineering and Design 170 (1997) 125-132.
[592] Beata Eisenmann, Joachim Fleisch, Erwin Prechtl, Werner Süßdorf, Manfred Urban, "Experience in Remote Demolition of the Activated Biological Shielding of the Multi-Purpose Research Reactor (MZFR)", 2012
[593] OECD/NEA, RWM (CPD) project, "Decontamination and Dismantling of radioactive concrete structures", 2011.
[594] OECD/NEA, RWM (CPD) project, "Decontamination and Dismantling of radioactive concrete structures", 2011.
[595] OECD/NEA, RWM (CPD) project, "Decontamination and Dismantling of radioactive concrete structures", 2011.

Demolition technique	**Project**	**Description**	**Year**
Wrecking ball	Elk river reactor (USA)	It was used in dismantling reactor containment building cylinder and dome after the outer insulation and steel shell were removed. This was done after all radioactive material had been removed within the structure.	1974
Backhoe mounted ram	R-1 (Sweden)	This method of chipping-off concrete with a backhoe was used for demolition of the biological shield in the R-1 research reactor in Stockholm [596].	1983
Rock splitter	Walter Reed research reactor (USA)	Rock splitter was used to demolish the heavy and dense biological shield [597].	1971
Expansive grout	BR3 (Belgium)	It has been successfully used to break up heavily reinforced concrete bases from 1 to 3 m^3. The expanding grout was left to cure overnight and the cracked concrete bases were excavated using a small back actor machine[598].	2014
	Sellafield (U.K)	Expanding grout have been successfully used to break up heavily reinforced concrete bases from 1 to 3 m^3. The expanding grout was left to cure overnight and the cracked concrete bases were excavated using a small back actor machine [599].	2009

[596] IAEA- Technical report series-401, "METHODS FOR THE MINIMISATION OF RADIOACTIVE WASTE FROM DECONTAMINATION AND DECOMMISSIONING OF NUCLEAR FACILITIES", Vienna 2001.

[597] R. I. Smith G. J. Konzek W. E. Kennedy, Jr., "TECHNOLOGY, SAFETY AND COSTS OF DECOMMISSIONING A REFERENCE PRESSURIZED WATER REACTOR POWER STATION" NUREG/CR-0130 Vol. 1, 1978

[598] OECD/NEA, RWM (CPD) project, "Decontamination and Dismantling of radioactive concrete structures", 2011.

[599] OECD/NEA, RWM (CPD) project, "Decontamination and Dismantling of radioactive concrete structures", 2011.

6.10 Robots and remote-controlled tools for dismantling

In most cases the decommissioning process utilises well-established dismantling techniques. However, the complication in the case of the decommissioning of nuclear facilities is the hazard of radiation release. The key use of robotics in decommissioning applications is to reduce the radioactive dose levels to which workers are exposed. In addition, the use of digital technologies could assist in workers' training, investigating alternative decommissioning procedures, and reduce the decommissioning project lead time and therefore the staffing costs during the decommissioning project. In particular, the removal and segregation of components of NPPs or other nuclear facilities, component inspections, conduct of radiological surveys, monitoring and sampling, mechanical and chemical decontamination, application of strippable coatings/fixatives, and work control are candidate operations for robotics and remote controlled systems[600].Some of the highlighted reasons for using remotely operated equipment besides working in hazardous environments are given in [601]:

- Due to enhanced safety, remote equipment can tolerate high radiation fields while performing the required tasks that induce high doses for more workers.
- It can reduce the cost of the decommissioning project by replacing human workers and accessibility.
- The overall workhour requirements can be lowered by a properly trained remote system operator, which in turns improves the productivity.
- Remotely operated cranes, fuel handling machines, and other equipment that are already present at the facility for normal operations can also be used in decommissioning project.
- Remote-controlled tools can provide access to work locations that are virtually impossible for a human to physically enter.

There are three different types of robotic systems: autonomous robots, supervised robots, and teleoperated robots. Primarily, work in hazardous environments in the nuclear industry is particularly dependant on 'variability and accessibility' of the work environment. Currently, autonomous robots are unable to function efficiently in many dynamic or variable environments, which require either complete human control or teleoperator (master/slave manipulation) solutions.

6.10.1 Description of techniques

6.10.1.1 Detection equipment

To perform surveys, monitoring and gathering data for analysis; detection equipment such as cameras, lights, radiation detectors to determine the most radioactive material in the area and infrared detectors for heat-sensitive materials can be used for real-time operator monitoring. Signals are transmitted to a receiver and visually displayed on large monitors for the operator's use. Different type of cameras can be used according to the operation. Characterisation techniques can be employed using remote applications. These applications may include alpha, beta, gamma or neutron radiation, checking floors for volatile

[600] Hart. J, Poley A.D, « Dose reduction and Planning, Digitization and Decommissioning », NRG (USA), 22.12.2020
[601] Decommissioning handbook by U.S Department of energy Office of Environmental Restoration, March 1994.

organics and mercury. Infrared cameras can be used to detect heat, microphones and radios to detect sound and thermometers for taking temperature and humidity measurements.

6.10.1.2 Segmenting equipment

Cutting tools including circular saws, nibblers, arc saws, plasma arc cutters, reciprocating saws, laser cutters, friction saws, grinders and rotary hammers can be applied to remote operations. Some technologies are more easily adapted for remote applications (i.e. Plasma arc cutters) while, some technologies need special fixtures, equipment, and custom-designed tools for remote operation. Many types of heavy equipment such as backhoes with ram-implemented attachment can be used for remote operated breaking concrete. Other remote equipment includes nut running tools and impact wrenches.

6.10.1.3 Decontamination equipment

There are some techniques suitable for remote operations including scabbling, vacuuming, steam cleaning and spraying. However, it is difficult to adopt some techniques for remote applications (i.e. in situ cleaning device used for electropolishing or other surface cleaning chemicals). The adaptation is made difficult but still feasible. With the advancement in technology, inputs to remote systems can be increased and other decontamination facilities will be added to remote operations.

6.10.1.4 Material handling equipment

Remote handling includes lifting, packaging and removing the material that is generated during the decommissioning project. Most of the facilities have on-site handling system which in turn reduces the operating costs and procurement delays. But, on facilities where handling equipment is not available, remote equipment should be carefully selected to minimise the recontamination of clean areas. Grapples, clamshells, or specially designed tools mounted on a remoted manipulator can be used for handling of the material generated during dismantling. Normally, the lifting capacity of a remote manipulator is limited and the physical clearance available in the material handling corridor limits the remote operations. Existing operating systems that can support handling operations include automatic guided vehicles, palletising robots, cranes, hoists, elevators and conveyors.

6.10.1.5 Sampling equipment

A core and bore drilling machine can be introduced within robots that can extract samples from within a structure. These mobile robots are designed for air, water, oil and debris sampling. Normally, this technology has been applied to disaster management.

6.10.1.6 Handled equipment

Long reach extensions to power wrenches and long reach hand triggered grapples take advantage of the distance rule and facilitates limiting radiation dose to operators. This class of equipment is utilised when dose rate limits for operators would be exceeded in contact situations, but where dose rates are manageable.

6.10.1.7 Remote system configuration

Table 6.10-1 Summary of remote system configuration

Configuration	Types	Description	Remarks/Examples
Cutting tools	Mechanical	Shears, power nibblers, saws, milling cutters, orbital cutters and abrasive cutters provide excellent remote operation.	Industrial state of development
	Thermal and similar	Plasma arc, flame cutters, and powder injection provide good remote operation.	Industrial state of development
	Electrical	EDM, MDM, consumable electrode, CAMC, and arc saw can also provide remote operations and used in many projects.	Industrial state of development

Configuration	Types	Description	Remarks/Examples
Arms	Electrical manipulators	These manipulators can be classified in 3 families depending on their payload capacity, number of axes, and their dexterity. -Power manipulators -Simple tele manipulators -Master/slave tele manipulators	-Power manipulators have 2 to 4 axes, a payload capacity of 50 to 500 kg and can be mounted on a crane. Suitable for heavy duty and low dexterity tasks -Simple tele manipulators have 6 DOF, a standard payload capacity of 20 to 250 kg and a control system providing basic operating functionalities from joystick to move manipulator with medium dexterity. -Master/slave tele manipulators have 6 to 7 DOF, a payload capacity of 20 to 60 kg, and a control system using master-slave technology to carry out tasks with high dexterity and productivity.
	Hydraulic manipulators	These manipulators can be classified in 2 families depending on their payload capacity, number of axes, and their dexterity.	Due to the great power-to-weight ratio of hydraulic manipulators, they are typically built for operations in which heavy objects are handled or tasks where large forces

Configuration	Types	Description	Remarks/Examples
		-Simple manipulators -Master/slave tele manipulators	are exerted on the physical environment.
Carriers	Bridge+lifting unit (telescopic system)	Carriers help in different tasks related to inspection, sampling, decontamination, and dismantling. These types of carriers are divided into different types: -Straight lifting unit -Suspended lifting unit	-Straight lifting units are sliding in relation to others and are employed in projects; [PIADE carrier (ELAN IIB), ATENA carrier (AT1), Remote dismantling machine (WAGR), U storage carrier (Marcoule)] -Suspended lifting units are hung by a cable under a travelling bridge. These are employed in projects, [Dual arm work platform (CP-5), Arm type (WAK)]

Configuration	Types	Description	Remarks/Examples
	Mobile carrier	These carriers are high power demolition machines. They work 6 to 8 times more efficiently than manual demolition. They are preferred for narrow openings and very restricted spaces. For lighter applications like cleaning up, decontamination, and radiological characterisation, a number of solutions have also been developed by CYBERNETIX, REMOTEC, and INTRA.	These earthly mobile carriers include BROKK 90, HUSQVARNA DXR 310, EROS, MENHIR (CYBERNETIX), REMOTIC Engine, Lifting table, and Telescopic elevator (AICHI) (MANITOU).
	Vertical supports	Support system can be provided to the manipulators to move vertically up and down.	-Circular gibbet combines a revolving motion with horizontal translation movement. -Longitudinal support gives two translation movements (one longitudinal and one vertical) -Vertical mast facilitates vertical movement and also by arm.

Configuration	Types	Description	Remarks/Examples
	Immersed carriers	These carriers are designed to work underwater. They have made considerable progress in terms of miniaturisation, pressure resistance, and leak tightness.	Examples include; VISIT II from ECA/HYTEC and H1000 from HYTEC.

6.10.2 Experiences/Case studies

Table 6.10-2 Experience of Robotics and remote-controlled tools in decommissioning projects

Dismantled components	Remote-controlled technique	Project	Year
RPV head and bottom, Pressuriser, Steam generator, and Neutron Shield Tank (NST)	Under water High pressure water jet with abrasive, carried and moved by MAESTRO arm (slave) tele manipulator, supplied by CYBERNETIX-AQUARESE [602].	BR 3 (Belgium)	2002-2008
Hot box, loop tubes, NST, Graphite core, thermal shield, lower structure, pressure vessel, and outer ventilation membrane	Remote dismantling machine (RDM) consisted of two handling system under a turntable. First system was an extendable mast suspending remotely controlled manipulator. Second contained a series of suspended crane rails [603].	WAGR (U.K)	1999-2006
RPV's upper and lower part of protection tube unit and cavity, core basket, cavity bottom	Three cutting areas were installed for the remote dismantling of the reactor components from Greifswald 1-4: a dry cutting area, a wet cutting area, and a cutting area in the reactor pit. Remote techniques used were: Band saw, disc cutter, plasma arc, and contact arc metal cutters [604].	KGR (Germany)	2004-2007

[602] IAEA-TECDOC-1602, Innovative Remote Dismantling Techniques – Final Technical Report – 2004 - 2008.
[603] NEA/OECD, Radioactive Waste Management (rwm-r2011-2), "Decontamination and Dismantling of Radioactive concrete structures", 2011.
[604] Raasch, J., and Borchardt, R., 2001, "Remote dismantling of the WWER reactors in Greifswald", ICEM'01 Report; Bruges 30.09 - 04.10.2001.

Dismantled components	Remote-controlled technique	Project	Year
Internals of RPV including cable ring carrier and annular water tank.	The cutting was carried out remotely from a central control station by using manipulators. Wire saws are used remotely as a cutting technique [605].	KKR (Germany)	2005-2006
Reflector, thermal shield, thermal shock baffle, internals, inner vessel, and outer vessel	Milling machine (ZWZ) was used to cut the reactor vessel based on a multiplicity of restrictions and circumstances (Sodium contamination). Disassembly master-slave manipulator provided with 6 axes to be able to reach all cutting positions using milling modules, hydraulic shears, oscillation brakes and magnet grip arms was installed [606].	KNK-2 (Germany)	2002-2008
Removal of graphite blocks, lead sheeting, boral, carbon steel, and aluminium reactor tank.	Remote system named Dual Arm Work Platform (DAWP) was used to perform mechanical dismantlement of radioactive reactor and bio-shield. The DAWP manipulator facilitated with commercially available tools (i.e., circular saws, jackhammers, etc.) used two Schilling Titan III hydraulic, teleoperated manipulator arms	CP-5 (USA)	2000

[605] NEA/OECD, Radioactive Waste Management (rwm-r2011-2), "Decontamination and Dismantling of Radioactive concrete structures", 2011.
[606] Iris Hillebrand, Forschungszentrum Karlsruhe GmbH, "Decommissioning KNK - Concept for Dismantling the Reactor Vessel and the Biological Shield"

Dismantled components	Remote-controlled technique	Project	Year
	controlled from a remote location for the task [607].		
Intranuclear instrumentation tubes, the vessel connection pipes and the upper vessel	Robotics were used for the cutting under water using band saw, disc saw, drilling tool and shear tools with robotic arms.	José Cabrera (Spain)	2013-2015

A significant amount of information on the experiences of the use of robotics technologies for nuclear decommissioning can be found in the IRID website, related to Fukushima Daiichi NPP.[608]

[607] EPRI Technical report, R. Reid, "Evaluation of System Automation and Robotics for Decommissioning Applications", November 2017.
[608] https://irid.or.jp/en/

7. Remediation and Site Release

The IAEA Safety Glossary[609] defines 'remediation' as: "Any measures that may be carried out to reduce the radiation exposure due to existing contamination of land areas through actions applied to the contamination itself (the source) or to the exposure pathways to humans".

Thus, remediation, if used in a narrow sense, is about reducing exposures through actions directed at the source and/or through actions directed at the exposure pathway. Environmental remediation is often considered to have the goal of returning a site to the conditions that prevailed before the contamination. In practice, however, this is often not feasible, especially if large areas are affected.

Remediation is itself a multi-phased activity consisting of identifying the environmental problems, gathering information in order to make decisions about how to solve the problems, carrying out the remediation project that will solve the problem, and verifying and documenting that the solution has in fact been achieved.[610]

Environmental remediation involves many challenges including technological and safety challenges. They are mainly related to:

- the characterisation methods and technologies needed in the problem definition phase (to determine the exact nature of the problem or if a response is even required) and in the final release phase (to verify the clearance levels)
- the technologies and methodologies applied during implementation phase to remove/reduce to acceptable level the contamination.

Site Release is the final step in the decommissioning process, granted if facilities have been removed and the site been cleaned in accordance with the cleanup levels. The Decommissioning end state, subject to national legal and regulatory requirements, encompasses partial or full decontamination and/or dismantlement, with or without restrictions on further use. Thus a site can be released for unrestricted or restricted use.

Compliance to the cleanup levels has to be demonstrated by the operator and the regulator may review, check or otherwise act on data of the operator. Methodologies and techniques for final release survey of the site and tools for statistical analysis of data play a relevant role in this final phase of the decommissioning process.

International initiatives

<u>NEA Initiatives</u>

- At its third meeting, held in Karlsruhe, Germany on 17-19 June 2002, the Working Party on Decommissioning and Dismantling (WPDD) held a Topical Session on Buildings & Sites Release and

---

[609] IAEA Safety Glossary - Terminology Used in Nuclear Safety and Radiation Protection 2018 Edition
[610] NEA - Nuclear Site Remediation and Restoration during Decommissioning of Nuclear Installations -2014

Reuse. Presentations during the topical session covered key aspects of the release of buildings and sites and provided the basis for exchange of information and experience.[611]

- The WPDD has held topical sessions covering information and experience on materials management and buildings and sites release and re-use. The background report [612]contains detailed information on the release of sites of nuclear installations from radiological control. The report is based on a questionnaire on site release distributed to relevant decommissioning projects.

- In the framework of the NEA Co-operative Programme on Decommissioning (CPD) the Task Group on Decontamination and Dismantling (D&D) of Concrete Structures undertook a comprehensive review of proven technologies and methods for decontaminating, demolishing and disposing of concrete structures.[613]

- Following a proposal submitted to the NEA Working Party on Decommissioning, a Task Group on Nuclear Site Restoration (TGNSR) was formed. The task group (between March 2012 and April 2014) gathered information at selected nuclear sites on experiences, approaches and techniques for remediation that minimise risks to workers and the environment, as well as costs and disruptions to decommissioning programmes. This was achieved using national level and project level questionnaires, detailed case studies and the experiences of task group members. The reference report [614] summarises the work carried out.

  The report highlights lessons learned from 12 case studies (in Canada, France, Germany, Korea, Spain, UK and USA) and provides observations and recommendations to consider in the development of strategies and plans, including choice of technologies.

  The group also sent 2 questionnaires to NEA members, e.g. asked the different techniques which are used or are planned to be used for soil and groundwater remediation. In many cases remediation of radioactive contamination and nonradioactive contamination is dealt with using the same technique. In a few cases, different techniques will be used (in-ground barrier, e.g. permeable reactive barrier, is a favourite method for remediation of radiological contamination).

Answer	Radiological contamination		Non-radiological contamination	
	Number of sites	Percentage (%)	Number of sites	Percentage (%)
Soil				
Dig and dispose	12	92	8	80
In-situ stabilisation	1	8	1	10
Capping	2	15	1	10
Other	3	23	3	30

[611] Topical Session on Buildings & Sites Release and Reuse, NEA/RWM/WPDD(2002)8

[612] Release of Sites of Nuclear Installations, Evaluation of a Questionnaire issued by the WPDD of the OECD/NEA and other Background Information, Final Report, NEA/RWM/WPDD(2005)10

[613] The NEA Cooperative Programme on Decommissioning, Decontamination and Demolition of Concrete Structures, NEA/RWM/R(2011)1

[614] Nuclear Site Remediation and Restoration during Decommissioning of Nuclear Installations, NEA#7192, 2014

- NEA report "R&D and Innovation Needs for Decommissioning of Nuclear Facilities"[615] identified site remediation as one of the 7 most important challenges, with needs for development of technologies:
  - Development of new ex-situ and in-situ methods:
    - Phytoremediation,
    - Bacteriological remediation,
    - Physical/chemical treatments (inc. extraction) with nanotechnology development.
  - Minimisation of waste generated, 'auto-remediation' with time.
  - Need for industrialised technologies, tested on sites, e.g. use of engineered barriers to fix contamination in the long term and prevent airborne dispersion to ground water

IAEA Initiatives

- IAEA ENVIRONET (Network on environmental remediation)[616],[617]
  - "Lessons Learned from Environmental Remediation Programmes": Lessons learned regarding technical aspects of remediation projects are reviewed. Techniques such as the application of cover systems and soil remediation (electrokinetics, phytoremediation, soil flushing, and solidification and stabilisation techniques) are analysed with respect to performance and cost.
  - "Remediation of Sites with Mixed Contamination of Radioactive and Other Hazardous Substances"
  - Review of applicable technology, with links, mainly in USA :
    - Technology Reference Guide for Radioactively Contaminated Media
    - Factors Affecting Treatment Cost and Performance
    - Excavation, Retrieval and Off-Site Disposal
    - Soil Flushing
    - Electrokinetic Separation and Chemical Oxidation
    - Solidification/Stabilisation
    - Monitored Natural Attenuation
    - Phytotechnologies

- IAEA International Project on Completion of Decommissioning (COMDEC)
  - A new IAEA-led initiative to support national authorities in the decommissioning of shutdown nuclear power plants and other nuclear installations was launched in September 2018
  - The three-year project will result in a systematic overview of the global experience in:
    - defining the desired final status of decommissioning
    - demonstrating compliance with end-state criteria
    - defining and implementing any necessary measures and controls after the end of decommissioning
  - One of COMDEC working groups deals with regulatory aspects, including release of sites and institutional controls

[615] https://www.oecd-nea.org/rwm/pubs/2014/7191-rd-innovation-needs.pdf
[616] https://nucleus.iaea.org/sites/connect/ENVIRONETpublic/Pages/default.aspx
[617] https://www.linkedin.com/groups/3256887/

- The project will provide input for the coming revision of the IAEA Safety Guide WS-G-5.1 "Release of Sites from Regulatory Control on Termination of Practices".

Other Initiatives

- Lot of references are provided in the proceedings of The ASME International Conference on Environmental Remediation and Radioactive Waste Management (ICEM) which promotes a broad global exchange of information on technologies, operations, management approaches, economics, and public policies in the critical areas of environmental remediation.

- In the USA four Federal agencies having authority and control over radioactive materials - Department of Defense (DOD), Department of Energy (DOE), Environmental Protection Agency (EPA), and Nuclear Regulatory Commission (NRC) – created a multi-agency working group aiming at providing detailed guidance for planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose or risk-based regulations or standards in radioactively contaminated sites.[618]

[618] NUREG-1575, Rev. 1 / EPA 402-R-97-016, Rev. 1 /DOE / EH-0624, Rev. 1 "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)", August 2000

7.1 Clearance of surfaces and structures (interiors and exteriors)

Clearance or release from regulatory control, where the materials meet the requirements for radiation protection of the appropriate regulatory body, is the final stage of the decommissioning process with regard to remaining surfaces and (building) structures on site. A legislative and statutory framework is required to address the objectives, principles and safety aspects relating to the release of sites from regulatory control.

National laws and regulations on the clearance of materials should be also in place. During clean-up of a site, some material may be suitable for release from regulatory control, if approved by the appropriate regulatory body.

Ensuring compliance with release criteria is the task of the operator, based on approved procedures that describe the approach to measurements and the activities and notifications for demonstrating compliance with the clearance requirements and criteria. Methodologies, characterisation principles and activity measurement methods form an indispensable part of these release strategies.

7.1.1 Standards

See also 1.2.1.3 and 1.2.1.4

There are four international documents that provide high-level guidance for relevant competent authorities when establishing clearance levels.

- Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, 2014
- RP89 – Recommended Radiological Protection Criteria for the Recycling of Metals from the Dismantling of Nuclear Installations
- RP113 – Recommended Radiological Protection Criteria for the Clearance of Buildings and Building Rubble from the Dismantling of Nuclear Installations
- RP122 – Practical Use of the Concepts of Clearance and Exemption – Part 1: Guidance on General Clearance Levels for Practices

A significant number of countries have developed national regulations that are based on these international regulations and guidance. France is an exception. Having no national regulation for clearance, France has introduced the concept of zoning, whereby wastes from a radioactive zone are considered as radioactive waste regardless of their actual radioactivity and waste from the non-radioactive zone is classed as conventional waste.

It should be noted that within the EU there are efforts to create standard criteria across all the member countries.

- Council Directive 2013/59/EURATOM laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation

Adoption of this directive by member countries is in progress. Implementation of the directive by member countries should help to harmonise the EU community and align them to the IAEA International BSS.

7.1.2 Methodologies and technologies

Methodologies for Clearance of surface and structures are defined in different national/international standards and guidance: ISO [619], DIN [620], [621], [622], ASN Guides[623] and MARSSIM[624].

In particular, MARSSIM provides information on planning, conducting, evaluating and documenting environmental radiological surveys of surface soil and building surfaces for demonstrating compliance with regulations. MARSSIM has not been updated since 2001. Revision 2 – foreseen in 2020 – updates the science, clarifies methods and implements lessons learned from over 20 years of the document's use in industry.

Moreover, there are different Application/Approach based on IAEA Safety standards [625], [626] and Safety reports [627] and on technical document provided by OECD-NEA [628] (see the international initiative at the beginning of section 7).

A study to evaluate methods and approaches used in different countries to achieve clearance of land where nuclear activities have been carried out (also called site release) has been carried out in 2013 by the Swedish Radiation Safety Authority. [629]In the EPA website[630] there are short descriptions of technologies used to characterise and/or monitor a site before, during or after remediation work.

Fiber Optic Chemical Sensors (FOCS) operate by transporting light by wavelength or intensity to provide information about analytes in the environment surrounding the sensor. The environment surrounding a FOCS is usually air or water.

Gas Chromatography (GC) is the most widely used chromatographic technique for environmental analyses, and is used onsite in field investigations and by offsite reference laboratories. Chromatography

[619] Characterisation principles for soils, buildings and infrastructures contaminated by radionuclides for remediation purposes, ISO/DIS 18557:2017
[620] Deutsches Institut für Normung e.V., DIN25457-4, Activity measurement methods for the clearance of radioactive substances and nuclear facility components – Part 4: Contaminated and activated metal scrap, 2013
[621] Deutsches Institut für Normung e.V., DIN25457-6, Activity measurement methods for the clearance of radioactive substances and nuclear facility components – Part 6: Rubble and buildings, 2017
[622] Deutsches Institut für Normung e.V., DIN25457-7, Activity measurement methods for the clearance of radioactive substances and nuclear facility components – Part 7: Ground surfaces and excavated soil, 2017
[623] Guide de l'ASN n°14: Assainissement des structures dans les installations nucléaires de base, Autorité de Sûreté Nucléaire, 2016
[624] U.S. NRC, EPA, DOE; MARSSIM, Multi-Agency Radiation Survey and Site Investigation Manual", NUREG-1575, Rev.1, ML003761445, EPA 402-R-97-016, Rev.1, DOE/EH-0624, Rev.1, 2000
[625] Application of the Concepts of Exclusion, Exemption and Clearance, IAEA Safety Standard Series No. RS-G-1.7, 2004
[626] Release of Sites from Regulatory Control on Termination of Practices, IAEA Safety Standard Series No. WS-G-5.1, 2006
[627] Monitoring for Compliance with Remediation Criteria for Sites, IAEA Safety Report Series No.72, 2012
[628] Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities, NEA#7310, 2017
[629] Approaches used for Clearance of Land from Nuclear Facilities among Several Countries, Evaluation for Regulatory Input, 2013:14, Swedish Radiation Safety Authority
[630]https://www.epa.gov/remedytech/characterization-and-monitoring-technology-descriptions-cleaning-contaminated-sites

uses a diverse group of methods to separate closely related components of complex mixtures. Field GC can provide real-time, or near real-time data, facilitating decision making and reducing the length of field mobilisation.

High-Resolution Site Characterisation (HRSC) strategies and techniques use scale-appropriate measurement and sample density to define contaminant distributions, and the physical context in which they reside, with greater certainty, supporting faster and more effective site cleanup.

Immunoassay technologies use antibodies to identify and quantify organic compounds and a limited number of metallic analytes. The technology is used widely for environmental field analysis because the antibodies can be highly specific to the target compound or group of compounds, and immunoassay kits are relatively quick and simple to use.

Infrared Spectroscopy has been an established benchtop laboratory analytical technique for many years. It identifies and quantitates compounds through the use of their infrared absorption spectra. Another use of the infrared spectra is found with video cameras that use infrared absorption to image the absorbing compounds on a video tape.

Laser-induced Fluorescence is a method for real-time, in situ field screening of residual and non-aqueous phase hydrocarbons in undisturbed vadose, capillary fringe and saturated subsurface soils and groundwater. The technology is intended to provide highly detailed, qualitative to semiquantitative information about the distribution of subsurface petroleum contamination containing polycyclic aromatic hydrocarbons (PAHs).

Mass discharge and flux estimates quantify source or plume strength at a given time and location. Consideration of the strength of a source or solute plume improves evaluation of natural attenuation and assessment of risks posed by contamination to downgradient receptors, such as supply wells or surface water bodies.

Mass Spectrometry is an established analytical technique that identifies organic compounds by measuring the mass of the compound's molecule. Although mass spectrometry can be used for the analysis of metals, non-metallic elements and radionuclides, it is most generally used for organic analysis as a field analytical technique.

Test Kits are self-contained analytical kits that generally use a chemical reaction that produces color to identify contaminants, both qualitatively and quantitatively. Numerous different kits are used in the environmental field. Test kits also can be used after an initial site characterisation phase to monitor the conditions of a remediation system or to confirm that contaminated soils have been removed.

X-Ray Fluorescence instruments are field-portable or handheld devices for simultaneously measuring metals and other elements in various media. The handheld or field-portable units use techniques that have been developed for analysis of numerous environmental contaminants in soil and sediment. They provide data in the field that can be used to identify and characterise contaminated sites and guide remedial work, among other applications.

Direct-Push Platforms use hydraulic pressure to advance sampling devices and geotechnical and analytical sensors into the subsurface. There are two sampling modes. One uses a specific tool string that either performs downhole measurements or gathers a soil or water sample at a specific depth. In the other mode, a dual tube arrangement is used to take continuous soil samples for evaluation at the surface.

Direct-Push Geotechnical Sensors can provide information about the physical properties of the subsurface environment, for example, density, competence and thickness of layers of soil or sediment. Sensors can provide information about stratigraphy, estimate depth to groundwater or approximate hydraulic conductivity.

The relatively low cost of Direct-Push Groundwater Samplers allows the collection of a larger number of samples both horizontally and vertically than could be done using conventional rigs. This density of sample taking provides a better idea of source zone locations and contaminant plume architecture, which maximises monitoring well placement efficiency and remedy design.

Direct-Push Membrane Interface Probes are semi-quantitative, field-screening devices that can detect volatile organic compounds (VOCs) in soil and sediment. They are used in conjunction with a direct-push platform to collect samples of vaporized compounds.

Direct-Push Soil and Soil-Gas Samplers have been developed to collect samples of unconsolidated material and vadose-zone gases from a range of depths, without generating large volumes of cuttings. Soil-gas sampling systems analyze vadose-zone gases at the surface or permit real-time chemical monitoring of soil gases in conjunction with direct-push analytical sensors.

Explosives behave differently than most other organic contaminants and pose an immediate safety hazard when present in large quantities or within unexploded ordnance (UXO). Energetic materials include chemicals that are used by the military as propellants, explosives and pyrotechnics. To assess the extent of explosive contamination, it is necessary to detect and identify explosives and their degradation products in soil and groundwater.

Ground Penetrating Radar (GPR) is a geophysical method that has been developed for shallow, high-resolution, subsurface investigations of the earth. GPR uses high frequency pulsed electromagnetic waves to acquire subsurface information. As with most geophysical techniques, the results should be compared with direct physical evidence.

Magnetics for Environmental Applications are used to locate subsurface iron, nickel, cobalt and their alloys which are typically referred to as ferrous materials. The technology has been widely used for quickly locating buried or subsurface cultural ferrous objects that could pose a potential threat to the environment or by assisting remediation efforts.

Open Path Technologies: Ultra Violet-Differential Optical Absorption Spectroscopy (UV-DOAS) uses the unique absorption of specific electromagnetic energy wave lengths by chemicals in the ultra violet, visible and near infrared spectrum to identify and quantify individual chemicals.

Open Path Technologies: Open Path Fourier Transform Infrared (OP-FTIR) spectroscopy is a versatile technology that can measure the presence of many chemicals in air simultaneously and achieve relatively low detection limits. FTIR open path measurements can be made using an active or passive approach.

Open Path Technologies: LIDAR operates on the same principles as radar except that it uses light rather than radio waves to collect information. There are three generic types of LIDAR:

- Range finders are used to determine the distance to a solid or hard target.
- Differential absorption LIDAR (DIAL) is used to measure chemical concentrations in the atmosphere (open air).
- Doppler LIDAR is used to measure the velocity of a moving target.

Open Path Technologies: Raman Spectroscopy sensors can identify chemicals, and provide an average concentration over the distance measured or at specified distances when a lidar configuration is used. The instrument uses an intense monochromatic light source and detectors to measure a portion of the light that is scattered inelastically from the analyte molecule.

Open Path Technologies: Tunable Diode Lasers (TDLs) are designed to focus on single absorption wavelengths specific to a compound of concern in the gaseous form. They are capable of achieving low detection limits and are virtually interferent-free. Open path TDLs are used in atmospheric pollutant studies, fenceline monitoring, process line/tank leak detection, industrial gas-purity applications and monitoring and control of combustion processes.

Passive (no purge) Samplers use methods based on the free flow of contaminant molecules from the sampled media to a receiving phase in a sampling device. Depending upon the sampler, the receiving phase can be a solvent, chemical reagent or porous adsorbent. They are deployed down a well to the desired depth within the screened interval or open borehole to obtain a discrete sample without using pumping or a purging technique.

7.1.3 Experiences/Case studies

See also 1.2.2

Over 180 commercial, experimental or prototype reactors, over 500 research reactors, and several fuel cycle facilities have been retired from operation. Around 20 of these reactors had the full decommissioning process completed by the end of 2016, of which the majority in the US. After the decommissioning process was completed, the sites were released as greenfield or industrial facilities for unrestricted use.

The most recent experiences in completed commercial NPP decommissioning projects are all located in the US, namely Yankee Rowe, Trojan, Rancho Seco and Zion.

In Europe on-going reactor decommissioning projects which are well advanced (building decontamination and dismantling phase) can be found in numerous countries:

- Sweden (Barseback 1&2)
- Spain (José Cabrera)

- Italy (Caorso, Trino, Garigliano)
- Germany (Stade, Obrigheim, Würgassen)
- France (Chooz A)
- Belgium (BR3)
- UK (Berkeley)

In Japan Tokai-1 reactor decommissioning nears its completion. Decommissioning examples of non-reactor type nuclear facilities related to clearance:

- France (CEA + Areva sites)
- US (DOE sites)
- Belgium (Eurochemic, Belgonucleaire, FBFC)

A review of best practices in terms of clearance of structures and surfaces should therefore focus on the above projects.

7.2 Characterisation methods and technologies to identify subsurface contamination

Globally the nuclear industry has an ongoing requirement to understand how contamination spreads within the subsurface to enable risks to key receptors; groundwater and surface waters, people and the broader environment, to be quantified. Operation of nuclear facilities inevitably generates waste materials and the disposal of these wastes, often by burial, along with the release of liquid effluents, deposition of aerial emissions and unplanned leaks and accidents can all lead to an increase in radioactivity within the subsurface environment.

Measurement of radionuclides in the subsurface is important if the contamination levels are to be established, rates and directions of movement determined and the risk to receptors modelled and quantified. The knowledge of how contamination moves in the subsurface forms a key component in the design of new nuclear sites, both for operational power stations and waste disposal. It is also crucial in the successful design of any works to remediate existing inground contamination.

A large amount of literature is available detailing the need for the measurement of radionuclides and some non-radionuclides in the subsurface environment for a whole range of purposes. International best practices include a range of International Atomic Energy Authority (IAEA) Publications including (as examples):

- Safety Report Series No.35 Surveillance and monitoring of Near Surface Disposal Facilities for Radioactive Waste (2004).
- IAEA Safety Standards No SSG-31 Monitoring and Surveillance of Radioactive Waste Disposal Facilities (2014).
- Technical Report Series No.445 Applicability of Monitored Natural Attenuation at Radioactively Contaminated Sites. (2006).

The principle requirement for subsequent understanding of the movement and behaviour of contaminants in the subsurface is that data collected to quantify them is of high quality and robust. It is essential that the characterisation undertaken can be relied upon to underpin decision making, whether that is a remediation project, design of a new facility or the quantification of a risk to (for example) human health.

This global requirement can be traced down into national requirements through the legislative framework countries have in place. Within the UK for example, nuclear licenced sites have a legal requirement to demonstrate control of contamination that has previously been accidentally released or otherwise lost into the subsurface. Such loss of containment can be considered a breach of the site licence conditions under which UK sites are operated, specifically Licence Condition 34: Leakage and escape of radioactive material and radioactive waste. Site Licence Condition 34 states “that so far as is reasonably practicable, that radioactive material and radioactive waste on the site is at all times adequately controlled or contained so that it cannot leak or otherwise escape from such control or containment”

This requirement has driven research and the development that has resulted in a range of technologies that are now available to potentially detect and track subsurface contamination comprising radioactive and other hazardous materials.

7.2.1 Technologies

The requirement to track subsurface contamination is not just restricted to the UK. An in-depth review incorporated both UK, European and international data and information from the nuclear industry, academia, non-nuclear industry, technology providers etc. revealed a wide range of technologies that could be deployed to identify sub-surface contamination;

- Fixed Radiometric Monitoring (above ground)
- Health Physics Monitoring (above ground)
- Below Ground Radiometric Techniques
- Intrusive soil investigation methods e.g. CPT
- Groundwater monitoring/sampling and laboratory analysis methods
- Remote/In-situ Groundwater Monitoring (loggers, sensors etc.)
- Geophysical Techniques
- Tracers
- Acoustic Emission
- Soil Gas Monitoring
- Sensing cables
- Real Time Soil Moisture Measurement (Neutron Probes)
- Thermal Methods

Some of the key technologies which have been developed more significantly in recent years both within and outside the nuclear industry include the following;

Downhole Logging

Dataloggers for level and conductivity measurement have been deployed widely within the groundwater monitoring field for many years. More intermittent use and trials of multi-parameter probes including ISE's have also taken place with promising results in both groundwater and plant environments. Loggers have also been utilised as part of tracer tests. Examples of these include tracer testing at the Low Level Waste Repository located in Cumbria in the UK and also as part of the pre-disposal work undertaken at the Radiana Low and Short Lived Intermediate Waste disposal site currently under construction in Bulgaria (http://dprao.bg/en/about-us/). This testing is used to ascertain flowpaths and investigate flow rate of groundwater and contaminant plumes.

Many contaminant plumes have the effect of altering basic parameters of the groundwater they are contained within when compared to uncontaminated groundwater e.g. temperature, pH, conductivity, dissolved oxygen etc.

A range of commercially available groundwater and surface water monitoring devices have been developed over the past 10-15 years measuring a range of parameters. Some like dissolved oxygen, pH, ORP, conductivity, salinity, total dissolved solids, temperature, turbidity and level monitoring are well developed. More recently more complex devices have become available including sensors for the dye tracer Rhodamine and ion selective electrodes such as nitrate, chloride, and ammonium.

The current research focus is on the field deployment of different monitoring 'loggers' within boreholes and rivers to ascertain their effectiveness in detection contaminant plumes which are radiologically dominated. In addition, the application of remote access logger systems is also being extensively developed and deployed. The deployment of data loggers with telemetry which allows for the downloading of data remotely has significantly improved the data sets available to underpin modelling of contaminant and ground water movement. Whereas prior to these development, individual readings were taken manually by dipping each well, it is now possible to collect time series data every few minutes or seconds over periods of months or years providing a granularity of data that to this point was not available.

Below Ground Gamma Monitoring

The detection of radioactivity in the sub surface is of particular importance on a number of nuclear sites especially given their decommissioning programmes where the decommissioning of redundant buildings significantly increases the risk of the release of contamination into the sub surface environment. A range of commercially available detectors, particularly for gamma radiation are currently on the market and widely used for below ground monitoring and these generally comprise below ground gamma detectors (for example LRGS (Na-I), IRGS (Cd-Zn-Te), HRGS and Geiger Muller) combined with a data collection system. In the same way data collection using down hole loggers can be used to detect groundwater data, gamma systems can be deployed to monitor gamma radiation insitu but in reality, given the cost of the equipment and the calibration and maintenance requirements, it is unlikely that such systems would be permanently installed in a borehole. More likely, an access tube would be installed within the borehole and regular gamma profiling using portable equipment would be undertaken to build up a time series data set for the subsurface gamma radiation levels.

Such a system is installed at the Sellafield site to monitor inground radioactivity around the Magnox Swarf Storage Silo facility. In this example, a number of steel tubes with sealed ends have been driven into the ground adjacent to the facility and gamma detectors are lowered to obtain data from a range of depths which in turn is used to model the movement of activity from historic leaks in the sub surface. Ultimately it is aimed to develop a permanent down hole monitoring array for this location but currently costs and technology availability preclude this from being available.

Electrical Resistivity Tomography

There are a range of geophysical methods and techniques available for leak or plume detection including:

- Electrical resistivity tomography (ERT)
- High resolution resistivity-steel cased resistivity tomography (HRR-LDM)
- Cross-borehole radar
- Cross-borehole seismic, and

- Cross-borehole electromagnetic induction

Of these the methods based on direct current resistivity, i.e., ERT and HRR-LDM, are better suited for the detection of contamination within the unsaturated zone for example an ongoing or future leak from a facility.

Passive Flux Meters

Passive flux meters (PFM) are passive sampling devices developed over the past ~ 10 years that provide simultaneous in situ point measurements of a time-averaged contaminant mass flux and water flux. The device, with a preloaded suite of tracers, and sorbent is placed in a monitoring well or borehole for a known exposure period, where it intercepts the groundwater flow and captures a target contaminant by sorption onto a specific media. The measurements of the contaminant and the remaining resident tracer can then be used to estimate groundwater and contaminant flux. The resulting contaminant mass flux profiles indirectly reflect the distribution of the soil contamination and directly reflect the mobility of the contaminants present. Prototypes are also in development which will attempt to measure flow direction with a PFM. A number of commercially available, non-radiological applications are now available. Research work to develop a passive flux meter for the detection and tracking of Strontium-90 within groundwater and/or surface water is ongoing with the aim to undertake a field trial of this technology at the Sellafield site in the near future.

Carbon-14 Gas

Soil gas monitoring for He-3/He-4 is a dating/monitoring methodology for tritium plumes with possible application for leak detection. Other gases used in other studies include sulphur hexafluoride, noble gases, helium, krypton and deuterated organic compounds however, their presence in nuclear wastes is not constrained. One current area of research is C-14 within the inground vapour phase. It has been identified that C-14 becomes volatile at neutral and acidic pH, and as such it is possible that on leakage into the ground, C-14 may exchange with the unsaturated zone vapours and therefore be detectable within boreholes monitoring this zone.

Currently the implementation of a field trial is being considered. This would initially ascertain if the C-14 exchange phenomenon is occurring from historical contamination in the field. Manual samples from a boreholes screened within the unsaturated zone above a contaminant plume containing C-14 can be taken and analysed in a laboratory.

7.3 Modelling and statistical tools to analyse contaminant transport in subsurface soil and groundwater

This summary reviews the use of modelling and statistical tools to simulate below ground radionuclide contaminate transport on legacy nuclear sites. The link between decontamination and decommissioning (D&D) and contaminant transport models is the production of safety cases on complex nuclear sites. Therefore, this document is structured in terms of modelling to support the risk based management of below ground contamination on nuclear sites.

7.3.1.1 Land management during D&D

Much of the practical focus of the D&D of nuclear sites concerns characterisation, dismantling/decommissioning, treatment and waste management of above ground facilities. However, after over 50 years of civil nuclear activities in the UK and internationally, on many sites there is a legacy of below ground contamination.

Defining the end state of nuclear licenced sites requires (a) an assessment of the risks posed by below ground contamination and (b) a plan of how any risks posed by below ground contamination will be managed to achieve the defined end state. To provide regulatory assurance, nuclear sites must demonstrate that the site end state will pose acceptable risks to key receptor groups and the environment[631]. Laboratory experiments cannot run for the same duration as a nuclear sites, therefore modelling the transport of residual contaminants and associated risk calculations is one of the key ways to demonstrate that long term risks are acceptable.

In the UK, the Nuclear Decommissioning Authority (NDA) has published summaries of how the legacy nuclear sites that fall under the remit of the NDA will plan to reach their desired end states[632] . A key activity during land management will be the quantification and management of below ground contamination.

The UK's Environment Agency (EA) have issued CRL 11 which provides guidelines on how modelling should be used during the planning and implementation of remediation strategies[633]. While this guidance is not aimed at operational nuclear sites, the nuclear industry stakeholder engagement forum SAFEGROUNDS considers that the guidance is a source of best practice for land management on nuclear licenced sites[634] . The rest of this document is structured along the lines of a tiered risk based modelling approach as outlined in CRL 11.

---

[631] ONR, EA, SEPA, NRW. Management of radioactive waste from Decommissioning of Nuclear Sites: Guidance on Requirements for Release from Radioactive Substances Regulation, Version 1.0 July 2018

[632] R. Griffin. (2005). Guidance For Site Stakeholder Groups: A Route Map To Determine Site End States And End Points. Richard Griffin. NDA. 10 November 2005

[633] Environment Agency (2004). CLR11: Model Procedures for the Management of Land Contamination. Bristol

[634] P. Towler et. al., (2009). Safegrounds LMG V2 W29 Good practice guidance for the management of contaminated land on nuclear licensed and defence sites CIRIA London 2009

7.3.1.2 Tiered modelling approach

CRL 11 describes assessing risk using a pollutant linkage approach. This includes identification of a contaminant “source”, a “pathway” for movement of the contaminant and “receptor(s)” that are harmed in a defined way. The combination of source, pathway and receptor is commonly called a “scenario”. CRL 11 recommends a tiered approach to model calculations. This includes;

Tier 1. A simple generic assessment using “model” or idealised source-pathway-receptors using existing site data to:

a) Determine if risks are below regulator concern or justify further investigation
b) Provide a screening level levels of radionuclides (Bq/g soil or Bq/L in water) to further support more intensive site investigations (SI).

Tier 2. A site specific assessment is undertaken if risks from tier 1 are above a threshold level. It is common for detailed site investigations at this point to provide the information to construct a site specific assessment.

Tier 3. If threshold levels are exceeded at tier 2 then this can trigger land management options including remediation. Many sites have undertaken below ground contaminant modelling to support a detailed risk assessment and construct Safety Cases.

7.3.2 The use of models in the tiered approach.

7.3.2.1 Tier 1 Simple generic assessments

The EA developed CLEA[635] for tier one assessments of non-radioactive contaminants and RCLEA[636] for radioactive contaminants. In the US, the RESRAD suite of codes can consider fixed US regulatory based exposure scenarios[637]. BNFL developed the RADCONTAB spreadsheet tool for its UK sites[638]. These tools provide instantaneous risks, so effects that would change contaminant concentrations with time such as groundwater advection, diffusion and mixing are not considered. Exposure pathways tend to be fixed and use Regulatory defined scenarios.

If tier 1 exceeds a threshold value for a contaminant in soil or water, then site investigations are usually undertaken to provide the parameter values for a tiered 2 (and beyond) assessment. This is where statistical tools and models are most often used.

---

[635] Defra and Environment Agency (2002). The Contaminated Land Exposure Assessment Model (CLEA): Technical basis and algorithms. R & D Publication CLR 10, March 2002
[636] J. S. Penfold, Robinson PC, Walke RC and Watson CE (2011). RCLEA: The Radioactively Contaminated Land Exposure Assessment Methodology - Technical Report. CLR14, Version 1.2, March 2011
[637] Halliburton NUS Corporation, 1994, Verification of RESRAD. A Code for Implementing Residual Radioactive Material Guidelines, Version 5.03, HNUS-ARPD-94-174, Gaithersburg. USA
[638] S. M. Willans 1, H. G. Richards (2006). RADCONTAB 1.0: A Look-Up Tables Tool for Radiological Assessment of Contaminated Land on Nuclear Licensed Sites J Radiol Prot . 2006 Mar;26(1):105-10

The EA has issued guidance on how screening levels should be used in tiered assessments, including use of statistical techniques[639]. An increasing trend is for SI studies to be designed using the Data Quality Objectives (DQO) approach which has a requirement for statistically based sampling designs[640].

Normally SI studies will use spatial relational software in support of statistically based sampling. Geographic Information Systems (GIS) are used to map the spatial distribution of contamination, and many GIS suites include inbuilt statistical tools. A common GIS based statistical method is to plot variograms, which result in 3-D statistically based plots that identify contaminant plumes and hotspots. For example, CEA plot their SI data on 3D maps using, for example, the KARTOTRACK™ GIS tool with variogram analysis to identify areas of contaminant hotspots[641].

7.3.2.2 Tier 2 Simple site-specific calculations

Tools at this level use similar methods to Tier 1 methods but this time include site specific scenarios and parameters. Available UK modelling tools that can do this include CLEA and RCLEA. NNL developed the tool ReCLAIM which was designed specifically for application to UK nuclear licensed site scenarios [642], [643]. RESRAD has a large number of alternative exposure scenarios. Increasingly, the generic modelling tool GOLDSIM is being applied at this tier[644].

7.3.2.3 Tier 3 Detailed site-specific modelling

Tier three is normally applied to large and complex sites. They normally form part of a risk-based safety case that supports ongoing and end state land management plans. It is common for risks to be calculated using simplified compartment models such as GOLDSIM. The MONDRIAN suite of tools had that role in the 2002 LLWR Safety Case[645]. The key processes and parameters needed to configure and populate such models are derived from detailed contaminant transport modelling, using codes such as those discussed below.

Use of complex below ground contaminant transport modelling has been documented on NDA sites such as Sellafield and the LLWR, and USDOE sites such as Savannah River, Oak Ridge and Hanford

---

639 DEFRA and EA. (2002). CLR 7 Assessment of Risks to Human Health from Land Contamination: An Overview of the Development of Soil Guideline Values and Related Research. Department for Environment, Food and Rural Affairs (DEFRA) and the Environment Agency (EA), London

640 USEPA. (2006). Guidance on Systematic Planning Using the Data Quality Objectives Process. EPA QA/G-4, EPA/240/B-06/001, U.S. Environmental Protection Agency, Office of Environmental Information, Washington DC

641 Y. Desnoyers and D. Dubot (2012). Data analysis for radiological analysis: Geostatistical and statistical complementarity. Paper presented at: Workshop for radiological characterisation for decommissioning. 18019 April 2012. Sweden, Access in June 2020 from https://www.oecd-nea.org/rwm/wpdd/rcd-workshop/

642 S. M. Willans, N. Galais, C.P. Lennon and D.P. Trivedi, (2007). ReCLAIM v2.0: Comparison of calculated doses with other assessment tools when emulating contaminated land scenarios ICEM07-7309 paper, September 2-6, Bruges, Belgium

643 S. M. Willans, N. Galais, C.P. Lennon and D.P. Trivedi. (2007). ReCLAIM v2.0: A spreadsheet tool for calculating doses and soil/water screening levels for assessment of radioactively contaminated land. J. Radiol. Prot. 27: 87-93

644 GTG (2005). GoldSim Contaminant Transport Module User's Guide [includes Radionuclide Transport Module Description], Version 3.0 (May 2005), GoldSim Technology Group, Issaquah, WA USA

645 BNFL, Drigg Post-closure Safety Case: Overview Report, September 2002

In the 1990's – early 2000's BNFL developed bespoke tools such as the DRINK[646] code to perform complex contaminant transport calculations. However, code development especially in the US has led to the release of tools many tools, well known ones being TOUGHREACT[647], Modflow-MT3D[648], PHAST[649] , PFLOWTRAN[650] etc. In France the HP1 and HYTEC codes perform a similar role[651].

These codes can be configured using a 2D or 3D "mesh" of nodes or pipe connectors to represent localised groundwater flow. Contaminant migration between connectors is affected by chemical and physical processes such as sorption (several using advanced chemical models such as surface completion), precipitation and dissolution, ion-exchange etc.

A summary of subsurface contaminant modelling tools referenced above is provided in Table 7.3-1.

Moreover in the NRC report [652] an overview of the state of environmental modelling within the government sector is reported. It is important to note that these systems are multiple-environmental-media modelling systems, not single-media modelling systems. These systems models include representations of various media (e.g., air, ground, and surface water). A concise description of each system is given in this document.

---

[646] S Manton, T. Johnstone, D. Trivedi, A. Hoffman and P. Humphreys (1995). Modelling Radionuclide Migration in the Near Surface Environment with the Coupled Geochemical/Microbiological Code DRINK. Fourth International Conference on the Chemistry and Migration Behaviour of the Actinides and Fission Products in the Geosphere - Migration '93. Charleston SC, USA, December 12-17, 1993 Proceedings of the International Topical Meeting: Published in Radiochimica Acta 68 pp75(1995)

[647] T. Xu, Sonnenthal E, Spycher N., and Pruess K. (2003). TOUGHREACT User's Guide: A Simulation Program for Non-isothermal Multiphase Reactive Geochemical Transport in Variably Saturated Geologic Media. Earth Sciences Division, Lawrence Berkeley National Laboratory University of California, Berkeley, CA 94720. April 2003

[648] H. Prommer, Barry D.A. Zheng C. (2003) MODFLOW/MT3DMS based reactive multicomponent transport modeling. Ground Water 41(2):247–257

[649] D. L . Parkhurst, Kipp K.L, Engesgaard P, Charlton S.C. (2004) PHAST – a program for simulating ground-water flow, solute transport and multicomponent geochemical reactions. USGS Tech Methods 6-A8: p154

[650] G. E. Hammond, P.C. Lichtner, C. Lu, and R.T. Mills. (2011). PFLOTRAN: Reactive Flow & Transport Code for Use on Laptops to Leadership-Class Supercomputers, Editors: Zhang, F., G. T. Yeh, and J. C. Parker, Ground Water Reactive Transport Models, Bentham Science Publishers. ISBN 978-1-60805-029-1.71

[651] J. van der Lee, L. De Windt, V. Lagneau, P. Goblet. (2002), Presentation and application of the reactive transport code HYTEC, Computational methods in water resources, 1, 599–606

[652] NUREG/CP-0177 - PNNL-13654 «Proceedings of the Environmental Software Systems Compatibility and Linkage Workshop» 2000

Table 7.3-1 Summary of contaminant transport modelling codes referenced in this document

Code	Assessment Tier (1-3)	Output D – Dose/Risk C - Concentration	Configurable (C) or Fixed (F) geometry	Groundwater flow	Retardation processes	Aqueous chemistry	Bio-geochemistry	Comments
RCLEA	1, 2	D	F	N	N	N	N	End state scenarios not operational nuclear licensed sites
RADCON TAB	1	D	F	N	N	N	N	Operational UK sites
ReCLAIM	1, 2	D	F	N	N	N	N	Used on operational UK sites
RESRAD	1-3	D, C	C	Y	Y	SEE COM MENT	N	Risk code that can also model time dependency.
GOLDSIM	1-3	D, C,	C	Y	Y	SEE COM MENT	N	General purpose modelling platform with limited flow capabilities
PHAST	3	C	C	Y	Y	Y	N	
MODFLO W-MT3D	3	C	C	Y	Y	Y	N	
PFLOWT RAM	3	C	C	Y	Y	Y	N	
TOUGHR EACT	3	C	C	Y	Y	Y	N	
HYTEC	3	C	C	Y	Y	Y	N	
DRINK	3	C	F	Y	Y	Y	Y	Fixed to LLWR flow mesh
GRM	3	C	C	Y	Y	Y	Y	Very detailed biogeochemis try and gas generation

7.3.3 Experiences/Case studies

Some situations require processes that can't be readily included in the types of codes listed above. The bespoke BNFL developed GRM code is one of the very few codes that have been developed to model biogeochemical aspects of waste degradation and contaminant transport. It has been used in support of waste and contaminant evolution calculations for the UK's LLWR and also Finnish long term waste degradation studies[653].

Examples of the outcome from use of complex contaminant modelling tools are provided in Figure 7.3-1, Figure 7.3-2 and Figure 7.3-3, associated with Sellafield[654], US DOE Savannah River[655] and US DOE Hanford sites[656] respectively.

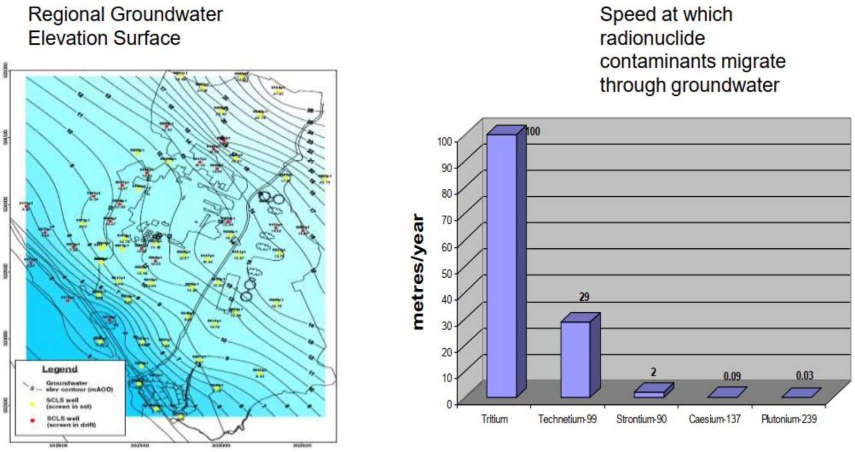


Figure 7.3-1 Output from groundwater and contaminant modelling at Sellafield

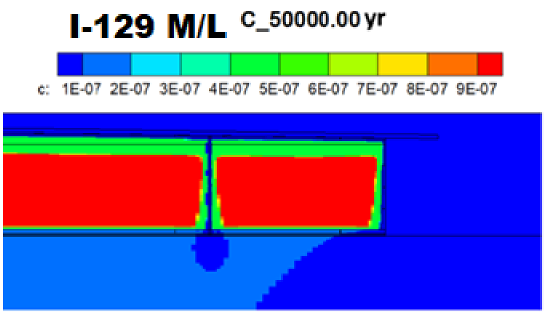


Figure 7.3-2 Modelling of I-129 release from disposed Saltstone at the USDOE Savannah River Site

---

653 J. S. Small, Humphreys, P. N., Johnstone, T. L., Plant, R., Randall, M. G. and Trivedi, D. P. (2000). Results of an Aqueous Source Term Model for a Radiological Risk Assessment of the Drigg LLW Site, UK. In: Scientific Basis for Nuclear Waste Management XXIII. Edited by R. W. Smith and D. W. Shoesmith. Materials Research Society Proceedings 608, pp 129-134

654 J McCord (2020). Presentation slides the GeolSoc. www.cms.geolsoc.org.uk\groups\specialist\ engineering. Accessed June 2020

655 G. P. Flach, Hang T. (2018). PORFLOW Simulations Supporting the Saltstone Performance Assessment December 13, 2018 SRNL-STI-2018-00652, Revision 0

656 Interra. (2020). Flow and Transport Modeling to Support a Remedial Investigation/Feasibility Study (RI/FS) for the 100-N Area at the U.S. Department of Energy's Hanford Site. Website interra.com accessed June 2020.

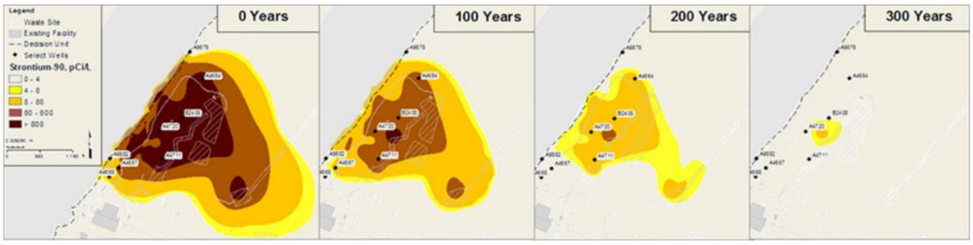


Figure 7.3-3 Modelling of contaminant transport in the vadose zone at USDOE Hanford Site

7.3.4 Further development needs

1. A key issue is that while high level codes such as GOLDSIM are designed to take into account parameter uncertainties (stochastic), all of the contaminant transport codes have fixed parameter values (deterministic). Being able to quantify the uncertainties in the stated risks is an important consideration, especially in terms of explaining the safety case outcome to key stakeholder. To provide the stochastic calculations requires to bridge the gap between assessment and contaminant transport models requires repetitive running of the complex radionuclide mobility codes. This can only be achieved over practical timescales using at least petascale (1015 FLOPS - floating point operations per second) calculations, which is the limit of supercomputing at the time of writing.

2. On nuclear sites some contaminants such as ^{14}C have an appreciable gaseous phase chemistry. Integrating aqueous and gaseous transport is rarely achieved (partially represented in GRM) and remains a gap.

3. The majority of codes cannot easily represent microbial processes, which can be important in controlling solution redox and also involved in several soil and groundwater remediation schemes. GRM represents key naturally occurring microbial processes but induced microbial processes during clean-up are difficult to model in any code.

4. The impact of climate change over long timescales is very difficult to simulate without beginning a new model representation at each new climate state. Having the required variable boundary conditions (and variable model meshes) is challenging.

5. Several codes, including some coupled GOLDSIM configurations, TOUGHREACT, GRM, PFLOWTRAN etc., are coupled to the chemical modelling code PHREEQC to perform chemistry calculations. This tool is no longer supported by the US Geological Survey (USGS) which presents a long-term threat to code development.

7.4 Soil remediation technologies

Technologies for soil remediation depend on the nature of contamination and on the nature of soils. For example, the radioactive ^{137}Cs released in the environment (nuclear accident for example) is known to be durably fixed into the soils (half-life 30.2 years) as Cs is mainly sorbed on negatively charged soil particles, preferably on phyllosilicates microparticles (2-3 µm) and aggregates (<10-12 µm). The strong Cs affinity for phyllosilicates leads to its very low release in solution as referred for illite and vermicullite which are present in several soils (as Fukushima area) and prevents the use of washing processes[657].

7.4.1 Description of technologies already implemented on sites

7.4.1.1 Leaching

Numerous articles, patents have been published concerning the leaching of soil, solid waste (such as tiles, asbestos, ashes…) and contaminated rubble from a nuclear site being dismantled [658, 659, 660]). The variability of the leaching solutions and the nature of the solid matrix to be treated (matrix structure, porosity, contamination, etc) makes any direct comparison difficult. Some representative examples are described below.

The decontamination of soil/concrete rubble may be carried out by leaching processes, which allow radionuclides extraction while preserving the integrity of the solid matrix to be treated. In order to predict whether a metal can be extracted, the study of speciation should be considered. pH, oxidation state or the presence of organic species can influence this speciation and therefore metals mobility [661]. Cations may be classified from the most to the least mobile as follows: monovalent exchangeable cations in organic or inorganic structure, then multivalent exchangeable cations and then chelated cations with organic moieties or anionic species (such as sulfates or nitrates). Metallic ions as part of the crystalline structure of mineral particles are only mobile after decomposition or weathering and precipitated metals are mobile under dissolution conditions. Hence, leaching processes can be used to extract contaminant from solid matrices by combining both attack of the solid (physically, by using for example temperature/pressure increase or microwave irradiation or ultrasounds ; or chemically using acidic solution) and ion exchange processes, and can be course adapted in the case of radioactive contamination removal. The presence of exchangeable cations (K^+, NH_4^+) in the leaching solution allows to improve extraction from solid matrix and to prevent contaminant readsorption by ion exchange.

---

[657] Ishii, K., A. Terakawa, S. Matsuyama, A. Hasegawa, K. Nagakubo, T. Sakurada, Y. Kikuchi, M. Fujiwara, H. Yamazaki, H. Yuhki, S. Kim, I. Satoh, Measures against Radioactive Contamination Due to Fukushima First Nuclear Power Plant Accidents Part I: Removing and Decontamination of Contaminated Soil, International Journal of PIXE 22 (2012) 13-19

[658] T. Wang, M. Li, S. Teng, "Desorption of cesium from granite under various aqueous conditions", Appl. Radiat. Isot. 68 (2010) 2140-2146.

[659] D. Parajuli, H. Tanaka, Y. Hakuta, K. Minami, S. Fukuda, K. Umeoka, R. Kamimura, Y. Hayashi, M. Ouchi, T. Kawamoto, "Dealing with the aftermath of Fukushima Daiichi nuclear accident: decontamination of radioactive cesium enriched ash", Environ. Sci. Technol. 47 (2013) 3800-3806.

[660] P. Samuleev, W. Andrews, K. Creber, P. Azmi, D. Velicogna, W. Kuang, K. Volchek, "Decontamination of radionuclides on construction materials", J. Radioanal. Nucl. Chem.296 (2013) 811-815.

[661] C. Mulligan, R. Yong, B. Gibbs, "An evaluation of technologies for the heavy metal remediation of dredged sediments" J. Hazard. Mater, 85(2001) 145-163.

In most cases, leaching or scrubbing processes are combined with physical separation (by screening, classification, flotation, etc).

7.4.1.2 Particulate foam flotation

In this Particulate foam flotation process[662, 663], the soil is dispersed in water (5 to 10% by mass) with a small amount of cationic surfactant (<3 kg for a ton of earth) which acts as a "collector": it absorbs on the surface of contaminated phyllosilicates particles to increase their hydrophobicity and their attachment to the surface of air bubbles injected from the bottom of the flotation column. These air bubbles, covered with these small functionalised particles (containing ^{137}Cs), rise to the top of the column to form a stable foam enriched in ^{137}Cs. The foam is collected at the top of the column and the foam residue dries at room temperature in a few hours and constitutes the ultimate waste containing the major part of fine particles containing ^{137}Cs contamination of the treated soil. The bottom of the column contains the decontaminated suspension of soil which can then be recovered by a solid-liquid separation (for example by decantation) which also makes it possible to recycle the aqueous phase of the process.

7.4.1.3 Other Processes complementary to leaching processes

Reference [664] evaluated agglomeration leaching process using columns for TRIGA reactors soil decontamination. These soils could be decontaminated down to 100 Bq/kg as potential release criteria.

Reference [665] described spraying treatment for the removal of radionuclides of a contaminated surface. Secondary waste arising would be aqueous effluents, which can subsequently be reprocessed to reduce volume of radioactive waste prior to disposal.

Finally, coupling with ultrasound [666] or microwave irradiation [667] may also be considered.

Another way to improve the decontamination yields is to perform thermal desorption under sub-critical hydrothermal conditions (KAKENHI Projects n°18H03398 and n°17J07598). Reference [668] implemented a hydrothermal treatment process using seawater or ionic leaching solution to desorb Cs from clays. Increasing the temperature between 100 °C and the critical water temperature (374 °C) with autogenous pressure, improves the rate of desorption by promoting ion exchange.

---

[662] S. Faure, M. Messalier, Method for the radioactive decontamination of soil by dispersed air flotation foam and said foam, Patent WO2013167728, 2013-11-14
[663] Julie C. M. Chapelain, SFaure Davide Beneventi, “Clay Flotation: Effect of TTAB Cationic Surfactant on Foaming and Stability of Illite Clay Microaggregates Foams” Ind. Eng. Chem. Res. 2016, 55, 2191−2201
[664] J. Lee, J. Moon, G. Kim, K. Lee, “Decontamination of radioactive soil wastes using an agglomeration-leaching process” Korean. J. Chem. Eng. 27(2010) 639-644.
[665] F. Sandalls, “Removal of radiocaesium from urban surfaces”, Radiat. Protect. Dosimetry, 21 (1987) 137-140.
[666] H. Itabashi, K. Mori, M. Uemura, “Elimination extraction method of cesium in soil by ultrasonic utilization”. Patent JP 2016 004036A, (2016).
[667] H. Sato, A. Yamagishi “Decontamination method and decontamination device of radioactive cesium contaminated soil”. Patent JP 2017 072512A, (2017).
[668] X. Yin, L. Zhang, M. Harigai, X. Wang, S. Ning, M. Nakase, Y. Koma, Y. Inaba, K. Takeshita, “Hydrothermal-treatment desorption of cesium from clay minerals: The roles of organic acids and implications for soil decontamination”, Water Res. 177 (2020) 115804

Table 7.4-1 Chemical/Hydrothermal Leaching technology

Techno.	What is working	What is missing
Chemical / Hydro-thermal Leaching	Non-destructive - preserve the integrity of the solid matrix. Applicable to all radionuclides even if more difficult for Cs. Allows to reach potential release criteria with promising decontamination yields and volume reduction factors. Existing commercial full scale transportable system (up to 350 kg/h).	No single technology is sufficient – trains of technologies must be considered. Coupling with physical separation (screening, classification, flotation, etc). Coupling with ultrasound or microwave irradiation to enhance desorption at pilot scale. Preliminary studies mandatory concerning site and contamination context. High energy cost for hydrothermal treatment.

7.4.2 Experiences/Case Studies

7.4.2.1 FRTR

The Federal Remediation Technologies Roundtable (FRTR) provides a tool for screening potentially applicable technologies for a remediation project. The matrix allows you to screen 49 in situ and ex situ technologies for either soil or groundwater remediation. Variables used in screening include contaminants, development status, overall cost, and cleanup time. In-depth information on each technology is also available, including direct links to the database of cost and performance reports written by FRTR members.

7.4.2.2 Commercial systems deployed on DOE nuclear sites

Reference [669] gives information concerning US commercial systems deployed on DOE nuclear sites (Oak Ridge, Rocky Flats, Hanfor) with soil radionuclides contamination including Cs, Sr and also U, Pu and Am. Amongst quoted companies, Bergmann has built full-scale transportable systems allowing to process up to 350 tons of soil per hour. Authors conclude that in the case of remediation of complex sites, no single technology is sufficient and trains of technologies must be considered.

7.4.2.3 Radioactive soil excavated by KAERI around TRIGA reactors

Reference [670] described leaching system for ^{60}Co, ^{137}Cs radioactive soil excavated by KAERI around TRIGA reactors.

[669] J. Devgun, M. Natsis, N. Beskid, J. Walker, « Soil washing as a potential remediation technology for contaminated DOE sites » for presentation at Waste Management 93 Tucson (US) (1993)

[670] G. Kim, W. Choi, C. Jung, J. Moon, "Development of a washing system for soil contaminated with radionuclides around TRIGA reactors" J. Ind. Eng. Chem. 13(2007) 406-413.

7.4.2.4 Experience in UK

In the UK, a pilot-scale leaching facility treating 800 kg of soil contaminated with ^{137}Cs [671] and [672] were aiming a restoration target of 4Bq/g in treated soil; 73% of soil mass was recovered after separating particles of size greater than 1mm.

7.4.2.5 Contaminated soils around Fukushima-Daiichi damaged site

In Fukushima Prefecture, large quantities of removed/excavated contaminated agricultural and residential soils (22 million cubic meters) have been generated from decontamination works conducted by Japan MOE. It is planned to store these contaminated soils temporary contained in big bags in an Interim Storage Facility (ISF). Japan MOE studies how to reduce as low as possible the volume of soil to be stored. Then it is necessary to concentrate the radioactivity into a small volume of soil and decontaminate it by categorising below and above 8,000 Bq/kg (respectively reuse and store). There is a need for environmentally friendly processes operating in continuous mode that limit the production of secondary wastes and allow reducing the high volumes of contaminated soils.

Leaching processes have been considered for application to the contaminated soils from Fukushima-Daiichi with an upsurge of patents pending between 2011 and 2017. Ebara patented technology for decontamination of solid materials contaminated with Cs by leaching and eluting Cs in aqueous phase [673]. Although obtained decontamination yields around 95% are promising, the use of leaching solution raises a critical issue as regards outflows. In this way, organic biodegradable leachant as used by [674] and applied on real soils from Fukushima may be considered.

Japanese Ministry of the Environment (MOE) conducted in 2012 and 2017 two “technology demonstration plans” in order to test the efficiency of different processes to reduce the final volume of ultimate soils wastes for Interim Storage Facility.

MOE Demonstration plan in 2012 at Fukushima:

In 2012, eight techniques were first tested in Japan on samples of actual top soil removed and stored in big-bags. All the tests were evaluated by the Japanese Ministry of the Environment[675]:

- a thermal process (process n ° 1) which was destructive (not reported here),
- six separation techniques (processes n ° 2 to 7, Table 7.4-2)
- and a chemical treatment (process n ° 8).

The chemical washing n°8 by Toshiba Corporation using oxalic acid was not successful to extract Cs from

[671] N. Beresford, “Land contaminated by radioactive materials” Soil Use Manag. 21(2006) 468-474.
[672] G. Stonnell, M. Pearl, “Characterisation and restoration of contaminated land on the Dounreay and Harwell sites of UKAEA” In “Site characterization techniques used in environmental restoration activities”. IAEA-TECDOC-1148. International Atomic Energy Agency (Vienna) pp187-190.
[673] T. Sekine, T. Shimomura, T. Miama, D. Sakashita, K. Futami, “Decontamination method and apparatus for solid state material contaminated by radiocesium”. Patent EP 2 600 353 A2, (2013).
[674] H. Sawai, I. Rahman, C. Lu, Z. Begum, M. Saito, H. Hasegawa “Extractive decontamination of cesium-containing soil using a biodegradable aminopolycarboxylate chelator” Microchem. J., 134(2017) 230-236.
[675] Japan Atomic Energy Agency Report, “Decontamination Technology Demonstration Test Project”, June 26, 2012.

phyllosilicates particles.

Table 7.4-2 Summary of decontamination technology R&D projects based on separation techniques for Fukushima soils

N° Technique	Corporation	Process	Soil initial activity (Bq/kg)	Cs extraction%	volume reduction %
2	ROHTO Pharmaceutical Co., Ltd.	Small-scale separation system using a special pump and a sieving machine	15300	97%	90%
3	Takenaka Corporation	Ball mill/drum washer	12500	87-91%	48-60%
4	**Kumagai Gumi Co., Ltd**	**Crusher and washer system**	**19700 à 125000**	**89-99%**	**91%**
5	Hitachi Plant Technologies, Ltd.	Separation (sieving) followed by heating at 700 °C	10584	58%	12%
6	Konoike Construction Co., Ltd	Crusher and washer system, cavitation washing	3970	74-91%	66-75%
7	Sato Kogyo Co., Ltd.	Washing and separation using micro-bubbles, high pressure water jet	6600	85%	65%

A particularly promising approach was carried by process no. 4 company Kumagai Gumi Co., Ltd. using a 3 steps process: mechanical prewash, grinding of the soil and vibrating sieving to 5 mm, then treatment with hydrocyclone. At the end of the treatment, a fine portion of soil less than 75 µm is isolated, containing most of the caesium (clays + silts). Factors of 90% caesium removal and 90% volume reduction were obtained. Processing flow rates are not specified.

MOE Demonstration plan in 2017 at Fukushima

Two other and non-destructive technologies in order to reduce the soil volume for Interim Storage Facility were tested in 2017 on actual contaminated soil in big bags (second call of the MOE). Both are based on separation of fine particles that are assumed to contain the entire (or major) part of radioactivity, which is the case in Fukushima for Cs (but cannot be transposed as it in case of other radioactive contamination). One was proposed by Kajima Corporation based on the separation of fine particles containing radioactive Cs using magnetic field and micro bubble flotation, useful only for agricultural soils with a high clay content (80%). The second process was the "Particulate foam flotation" process (see Section 7.4.1.2), proposed to treat both residential soils and agricultural soils. It was tested by CEA/ORANO and VEOLIA to directly separate and concentrate contaminated phyllosilicates using a flotation foam[676].

Particulate Foam Flotation Demonstration campaign in OKUMA/Japan (November 2017)

Flotation tests on actual Japanese contaminated soils were successfully conducted in OKUMA (FUKUSHIMA prefecture) by the CEA / AREVA / VEOLIA teams in collaboration with ANADEC/ATOX with an integrated continuous pilot (24 kg of soil/hour).

For two kinds of soils where phyllosilicates particle proportion was the lowest, volume reduction factor from 3.5 to 7.4 depending on testing conditions was obtained on the pilot as expected and 40 to 45% of Cs extraction rate were achieved.

Cs extraction ratio can be improved if the amount of Cs particles that can be floated is increased. Before flotation, an optimised dispersion system of the soil in water is recommended: for instance, ultrasonic systems on lab scale experiments were tested to separate phyllosilicates from soil that could be stick in water to sand particles. Thanks to these improvements on soil dispersion, 80% of the initial Cs dispersed on fine particles below 75 µm are expected to be floated.

7.4.2.6 Decontamination of soils at JOSE CABRERA (Spain)

Remediation of potentially contaminated soils is currently being carried out within the scope of the Jose Cabrera NPP decommissioning project, among other activities. In order to reduce the volume of radioactive waste to be generated during restoration activities, a plant has been implemented to treat excavated contaminated soils by the technique of granulometric separation and aqueous washing of coarse fractions.

The soil washing plant is located on the east side of the José Cabrera site. Prior to the installation of the plant, the feasibility of the soil washing technique was demonstrated in a pilot test conducted in a laboratory. Once implemented at the site, tests with conventional material were accomplished to establish the operating parameters of the process. Subsequently 500 tons of contaminated soil were treated to verify the effectiveness of the decontamination process.

Description of the process

[676] Anouar Ben Said, Fabien Frances, Agnès Grandjean, Christelle Latrille, Sylvain Faure. Study of a foam flotation process assisted by cationic surfactant for the separation of soil clay particles: processing parameters and scaling-up sensitivity. Chemical Engineering and Processing: Process Intensification, Elsevier, 2019, 142, pp.107547

Soil decontamination is carried out by aqueous washing. Initially, contaminated soils from the excavation are processed in the plant separating the thin fraction (clay), which is the most contaminated, from thicker fractions (sands and gravel), which are cleaner and whose contamination is generally detachable. The thick fractions are then subjected to a physical washing process and eventually managed as releasable material. The aim is its conventional management. Several radiological determinations are made to ensure the compliance with the contamination concentrates in the fine fraction then this small volume will be managed as very low level waste. In this way, the total volume of radioactive waste is reduced with the consequent lower cost associated with the restoration of the site.

The washing plant treats soils classified as very low-level waste (VLLW), which because of its activity levels and physicochemical characteristics (proportion of fines less than 30%) it is anticipated that, after subjecting them to the process of washing, can be managed as releasable material. The gravel and sands washed are placed in containers that are sent to clearance area where gamma spectrometry measurements check if clearance levels are met.

As part of the installation, a water treatment plant that is no longer reusable in the process, with the aim of ensuring compliance with established criteria for discharge.

The facility has 4 storage areas, in order to temporarily store the materials that are generated during treatment.

The treatment capacity of the plant is 30-50 tons/ day, and its operation is performed in batches of about 500 tons.

Throughout the process, different radiological controls are carried out on the storage areas and in the water recovery line, in order to measure the effectiveness of the different processes performed. In the case of water, is necessary to decide the reuse or the need for treatment if there is incorporation of isotopes due to the washing process.

The main processes are:

- Material feed. Previously, the material larger than 100 mm are separated, in the area of entrance collection. Subsequently, the material is provided in the hopper of the loading platform.
- Granulometric classification and separation of gravel. From the hopper, materials are moved, using the tape conveyor feed, up to the first washing equipment: washing cylinder ("tromel"). The mixed soil and washing water are driven by gravity to a vibrant inclined screen that allows granulometric sorting, separating the gravel from the mixture.
- Washing of gravel. The gravel is washed on a ramp and go through a vibrant drainer. From this point, gravels are driven by a conveyor belt to the collection of releasable gravel.
- Separation of fine and sand washing. Separation between sands and fine takes place through two phases of hydrocycling, before and after washing process called attrition clearance levels.
  Attrition consists of a process of friction by which attached contamination is released. The sands obtained, once drained, are driven by means of a conveyor belt to the collection of releasable sands.
- Water recovery and fine dehydration. The material thinner and after a decanting process, is sent to the filtration stage. The filtration stage is performed through a filter-press in which all fine material

are compacted into appropriate conditions ("cakes"). The generated waste (the "cakes") are stored in big bag suitable for further management by Enresa.

– Water treatment. Water from dehydration rejoins the process or is sent to the water treatment plant. This treatment includes filtration and ion exchange (if necessary), that will ensure compliance with the radiological criteria and environmental discharge.

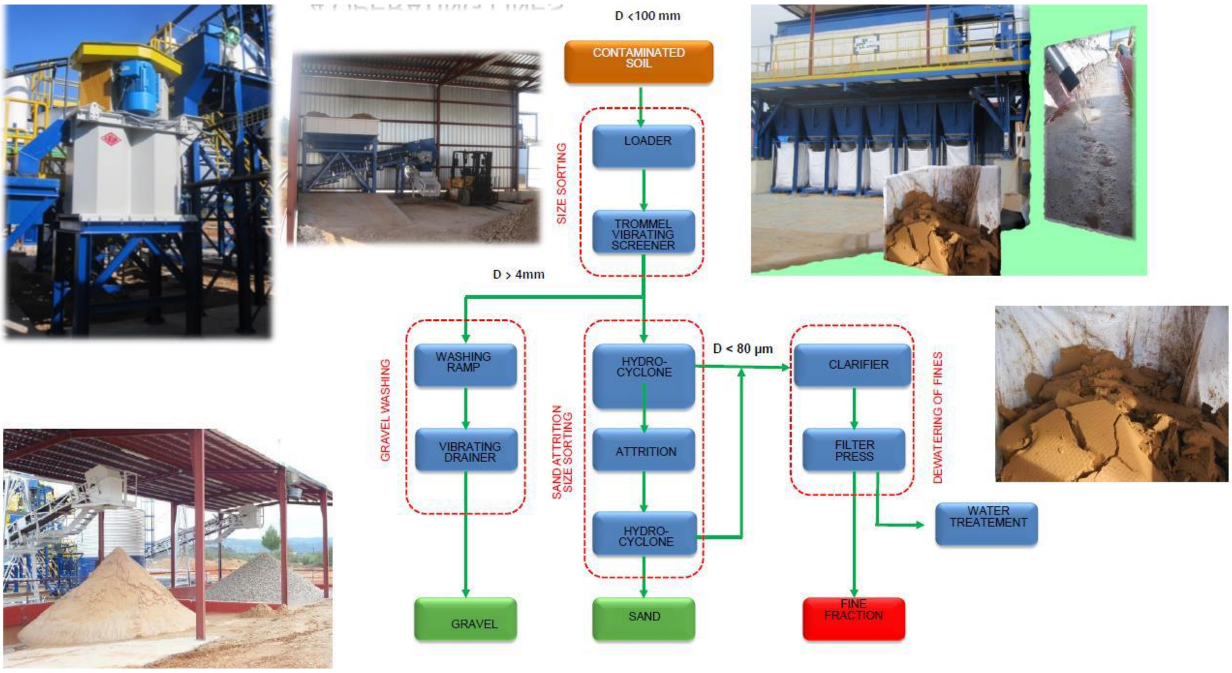


Figure 7.4-1 Decontamination Process

The physicochemical nature of gravels influences the effectiveness washing. Gravel of a limestone nature are porous and more difficult to wash when deposited contamination in the pores of the material. However, gravel of a silica nature whose surface is smooth, allow easily to eliminate contamination on its surface. At the José Cabrera site, the proportion of porous gravel is low. However, due to the high sensitivity of the clearance measurement system, a small amount of material of this nature can provoke the rejection of the container with the gravel washed.

In order to optimise the clearance process eliminating those materials whose washing is not completely effective, an automated system of continuous measurement for segregation has been installed at the exit of gravel collection area.

This system consists of a transporter that carries the gravel towards a detection set consisting of two scintillation detectors. The conveyor belt is divided into sections that carry an amount material, and once done the characterisation of gravels, overturns the contents of each bucket to the clearance measurement system or in a container to be treated as very low level waste.

The implementation of this system has made possible to improve the performance of the clearance of washed gravel, reducing the number of containers rejected through the process.

The soil washing plant obtained the approval for operation by the Nuclear Security Council in June 2018. Since then it has been in operation adjusted to the progress of restoration activities. During excavations, a significant part of soil is not contaminated, do not require washing treatment and can be sent directly to the clearance process. There are also certain fraction of the soil with higher contamination and granulometry composed mainly of fine material. In these case soil washing is not effective and are classified directly as radioactive waste. By all this, during excavations in situ radiological control are conducted to preclassify excavated soil in order to define their destination:

- Clearance
- Treatment at the soil washing plant.
- Radioactive Waste (VLLW).

At the end of 2019, the amounts of excavated soil and their destinations were as follows:

- Total excavated soil: 15.961 tons.
    - Releasable material: 12.231 tons.
    - Radioactive waste: 309 tons.
    - Material treated in the soil washing plant: 3.421 tons.

However, the amount of soil according to the destination may vary throughout dismantling depending on radiological characteristics of the restored area.

The results of the washing plant, indicate 74% of the amount of the washed material (29% sands+ 45% gravels), is eventually released, while 26% is managed as very low radioactive waste (3% gravels, 4% sands and 19% fine). These results highlight the optimisation of the volume of radioactive waste to be managed by concentrating the activity in the thin fractions of the soil and separate it from the rest of fractions (gravels and sands).

Conclusions

The soil washing plant installed in the José Cabrera nuclear power plant allows the project to optimise the volume of radioactive waste to be managed from the excavation of potentially contaminated areas. The percentage of clearance once the soils are treated in the plant, is currently around 74%. This percentage may vary depending on the characteristics of the area to be excavated, and is expected to remain between the 65 and 75% of released materials.

The soil washing plant for decontamination is a pioneer process in Spain and its applicability in other projects will depend on the granulometric composition and physicochemistry of the site terrains. Conducting lab-scale tests is essential for deciding the feasibility of such a treatment.

An additional segregation system has been implemented to separate the small amounts of gravel whose washing is not effective due to its physicochemical nature. Testing and operation of this measurement system has been a pilot experience, which may be applied to the in situ soil segregation in an automated manner, in future decommissioning projects.

7.5 Remediation of contaminated groundwater (radiological)

Remediation has usually been considered as the last step in a sequence of decommissioning steps. The remediation refers to actions taken to reduce the impact from contamination in land areas and in the associated groundwater in order to leave the site in a state that is suitable for its next use. [677]

Environmental remediation is a multi-phased activity consisting of identifying environmental problems, gathering information in order to develop a range of solutions to solve problems, evaluating the options and selecting the preferred solution, carrying out the remediation project that will resolve the problem, and then verifying and documenting that the solution was successful.

The presence of radioactive contaminants in groundwater may be caused by natural contaminants or could come from human activities. The natural occurring radioactive materials found in groundwater are mainly uranium, radium, and radon. Levels of anthropogenic radioactivity in groundwater come from nuclear weapons testing fallout, nuclear accidents (e.g. Fukushima and Chernobyl) and routine authorised releases to the environment. Most of the sources for groundwater contaminated with higher levels of radionuclides are from the production of nuclear weapons or nuclear power plants. The materials from these activities are uranium, plutonium, thorium, caesium, strontium, technetium, and tritium.

In principle remediation should only be considered if there is a source-pathway-receptor linkage, an approach that is used for both radioactive and other hazardous substances. Should remediation be necessary it could be carried out at different times at different parts of the site (partial site remediation or phased site remediation) or as one single project, depending on what is the optimum approach for the site.

For most environmental remediation projects, a maintenance and monitoring period may be required after site closeout. The sites that are released without restrictions will not require any long-term monitoring. However, the sites released with conditions, or released for restricted use will require long-term monitoring and post-remediation activities, at least to check that the land quality evolution is as expected.

7.5.1 Description of the methodologies

This chapter describe the methodologies already implemented according to the reference documents [678], [679], [680].

Some remediation groundwater methodologies are applicable to "in situ" treatment while others are more suitable for "ex situ" approaches. The in-situ remediation approach involves cleaning the water where it is presently situated, the ex-situ remediation involves having the excavated contaminated water and then treatment off-site.

---

[677] NEA - Nuclear Site Remediation and Restoration during Decommissioning of Nuclear Installations - 2014
[678] EPA - 402-R-07-004 - Technology Reference Guide for Radioactively Contaminated Media - October 2007
[679] IAEA-TECDOC-1086- Technologies for remediation of radioactively contaminated sites - June 1999
[680] IAEA - Technical Reports Series No.424 - Remediation of sites with dispersed radioactive contamination - June 2004

The most common form of groundwater remediation is known as "pump-and-treat", contaminated groundwater is extracted from the subsurface by pumping, the water is treated and then it is discharged, if water treated is reinjected to the subsurface, the method is "pump and re-inject". These methodologies are often associated with treatment technologies which falls into a few broad categories:

- **Chemical separation technologies** separate and concentrate radioactive contaminants from groundwater, surface, or waste water. Extractability rates of the different chemical separation technologies vary considerably based on the types and concentrations of contaminants, as well as differences in methodology. Chemical separation technologies can be in-situ or ex-situ. For ex-situ treatment of groundwater, the construction and operation of a groundwater extraction and delivery system is required. All ex situ chemical separation technologies generate a treated effluent and a contaminated residual that requires further treatment or disposal. The principle technologies are *ion exchange* and *chemical precipitation*, which are ex-situ treatments, and *permeable reactive barriers*, which are in situ treatments.
- **Physical separation technologies** separate contaminated media into clean and contaminated fractions by taking advantage of the contaminants' physical properties. The physical separation of the radionuclides from the liquid media results in "clean" liquid and a contaminated residue that requires further handling, treatment, and/or disposal. Physical separation technologies are ex-situ processes and require the construction and operation of a ground-water extraction and delivery system. They generate a treated effluent waste stream of which the volume and type depend on the technology. The principle technologies are: membrane filtration (reverse osmosis and microfiltration), carbon adsorption, and aeration.
- **Biological treatment** of radioactively-contaminated groundwater, surface water, and wastewater involves removal of the contaminants via plant root systems in a hydroponic or wetlands setting, uptake by root systems and transpiration to the air (for tritium), or control of the groundwater plume through significant uptake of groundwater by plants. The use of plant systems for treatment of contaminated groundwater, surface water, and wastewater is called phytoremediation.

Ion exchange separates and replaces radionuclides in a waste stream with relatively harmless ions from a synthetic resin or natural zeolite (for strontium and caesium). Resins consist of an insoluble structure with many ion transfer sites and an affinity for particular kinds of ions. "Exchangeable" ions are bound to the resin with a weak ionic bond. If the electrochemical potential of the ion to be recovered (contaminant) is greater than that of the exchangeable ion, the exchange ion goes into solution and the ionic contaminant binds to the resin. Resins must be periodically regenerated by exposure to a concentrated solution of the original exchange ion. A typical ion exchange unit uses columns or beds containing the exchange resin and various pumps and piping to carry the waste streams and potentially new and spent resin. Resins are either acid-cationic (for removing positively charged ions) or base-anionic (for removing negatively charged ions); resins used for radioactive liquid waste are often either hydrogen or hydroxyl. Typically, four operations are carried out in a complete ion exchange cycle: service, backwash, regeneration, and rinse. Media with more than one radioactive contaminant can require more than one treatment process.

Chemical precipitation converts soluble radionuclides to an insoluble form through a chemical reaction or by changing the solvent's composition to reduce radionuclide solubility. Precipitation adds a chemical precipitant to the radionuclide-containing aqueous waste in a stirred reaction vessel. Solids are separated from the liquids by settling in a clarifier and/or by filtration. Flocculation, with or without a chemical coagulant or settling aid, can be used to enhance solids removal. Commonly used precipitants include carbonates, sulfates, sulfides, phosphates, polymers, lime and other hydroxides. The amounts of radionuclides that can be removed from a solution depend on the precipitant and dosage used, the concentration of radionuclides present in the aqueous waste, and the pH of the solution. Reagents and filters must be selected on a site-specific basis for the particular radionuclides present. Multiple radionuclides could impact the technology's effectiveness, multiple treatment processes might be required.

Permeable reactive barriers are installed in the subsurface across the flow path of a radionuclide-contaminated groundwater plume, allowing the groundwater to passively flow through the wall while prohibiting the movement of the radionuclides. This is accomplished by employing treatment agents within the wall such as chelators and reactive minerals. A permeable reactive barrier is built by excavating a trench perpendicular to the groundwater flow path and backfilling it with the reactive materials, which can be mixed with sand to increase permeability. In some applications, the permeable reactive barrier is made the focal point of laterally connected, impermeable subsurface barriers or permeable conduits so that the groundwater is collected and funnelled through the reactive material.

Membrane filtration uses a semi-permeable membrane to separate dissolved radionuclides or solid radionuclide particles in liquid media (e.g., groundwater, surface water) from the liquid media itself. Generally, some form of pretreatment (such as filtration of suspended solids) is required in order to protect the membrane's integrity. Water flow rate and pH should be controlled to ensure optimum conditions. Two types of membrane processes used for treatment of radionuclides in liquids are micro or ultrafiltration and reverse osmosis. Micro and ultrafiltration rely on the pore size of the membrane, which can be varied to remove particles and molecules of various sizes. Reverse osmosis uses a selectively permeable membrane that allows water to pass through it, but which traps radionuclide ions on the concentrated, contaminated liquid side of the membrane.

Adsorption involves pumping groundwater through a series of vessels containing granular activated carbon. Dissolved contaminants in the groundwater are adsorbed by sticking to the surface and within the pores of the carbon granules. Although granular activated carbon is the most common adsorbent used, other adsorbents include activated alumina, Forager Sponge, lignin adsorption/sorptive clay, and synthetic resins.

Aeration is a mass transfer process that enhances the volatilisation of compounds from water by passing air through water to improve the transfer between air and water phases. The process can be performed using packed towers, tray aeration, spray systems, or diffused bubble aeration. In packed tower aeration, a counter-current flow of water and air are passed through a packing material. The packing, which typically consists of plastic shapes that have a high surface-to-volume ratio, provides a high surface area for the radon transfer from the water to the air. The groundwater is pumped to the top of the packed tower and distributed evenly over the packing while an air stream is blown into the bottom of the tower. Aeration requires a groundwater extraction and delivery system and adequate power to maintain the treatment system. Also, adequate venting and/or an air treatment system are required for aeration.

Phytoremediation is a process that uses plants to remove, transfer, stabilise, or destroy contaminants in groundwater, surface water, or wastewater. It applies to all biological, chemical, and physical processes that are influenced by plants and that aid in the cleanup of contaminated media. Phytoremediation can be applied in-situ or ex-situ (e.g. hydroponically) to groundwater or surface water. Rhizofiltration uses hydroponically grown plants that are exposed to contaminated water in their water supply resulting in uptake of contaminants by the plant roots and the translocation/accumulation of contaminants into plant shoots and leaves. Phytoremediation hydraulic control involves the use of deep-rooted plants to control the migration of contaminants in groundwater. Depending on the type of plants, climate, and season, plants can act as organic pumps when their roots reach down to the water table and establish a dense root mass that takes up large quantities of groundwater.

Table 7.5-1 Summary of remediation groundwater methodologies

Technology / Methodology	What is working	What is missing
Ion Exchange	This technology has been applied to waste streams contaminated with radionuclides and metals. Particularly groundwater, surface water, waste water, liquid waste. Ion exchange reduces the contaminant mobility. It is expected to remove 65 to 97% radium and 65 to 99% uranium. The range of removal of beta emitters such as Cs^{137} and Sr^{89} is 95 to 99%.	Ion exchange does not affect the radiotoxicity of contaminants. The performance and selectivity of adsorbents in applications are affected by a variety of environmental factors (i.e. composition of the groundwater).
Chemical precipitation	This technology effectively reduces high levels of radionuclides, especially radium and uranium, and dissolved metals from groundwater, surface water and wastewater. Chemical precipitation reduces the volume of contaminants and the toxicity of the liquid medium. It achieved 80% uranium removal using ferric sulfate, 92 to 93% uranium removal using ferrous sulfate, and 95% uranium removal using alum. Precipitation through lime softening can achieve 75 to 95% removal of radium.	Chemical precipitation does not reduce the mobility of the contaminants. The performance can be affected by the physical and chemical properties (e.g. temperature, pH, flow rate) of the waste material.
Permeable reactive barriers	This technology can effectively reduce the concentrations of radionuclide-contaminated groundwater. Costs become significant for depths of more than 80 feet (24.4 m). Reduction of uranium by as much as 99.9%; reduction of Sr^{90} by as much as 99%; reduction of Tc^{99} by as much as 51.6%.	This technology is not a rapid remediation. It is not suitable to sites with high levels of dissolved oxygen and/or high levels of dissolved minerals, with significant contrast in permeability, with numerous underground utilities or numerous large rocks.

Technology / Methodology	What is working	What is missing
Membrane filtration	This technology can treat a variety of waste, including metals and organics, and effectively remove most radionuclides from water. It is applicated to groundwater, surface water, waste water and leachate. Removal efficiencies for membrane filtration is greater than 99% for uranium, plutonium, and americium with initial concentrations of 35, 30 and 30 pCi/L, respectively. Removal efficiency was 43% for radium that had an initial concentration of 30 pCi/L.	Pre-treatment can be required to remove film-forming materials such as oxidants, iron and magnesium salts, particulates, and oils and greases. This will reduce fouling of the membrane and ensure the treatment's effectiveness.
Adsorption	This technology can be used to treat organics, certain inorganics, and radionuclides (uranium, Co60, Ru106, Ra226, and Po210, radon) from groundwater. Other applicable media are pretreated surface water, waste water and leachate. Effectively removes contaminants at low concentrations (less than 10 mg/L) from water at nearly any flow rate, and removes higher concentrations of contaminants from water at low flow rates (2-4 L/min). It has been used to adsorb radon and neutral forms of Co^{60} and Ru^{106}. Radon has been removed with efficiencies of 90 to 99.9%.	Activated carbon for the removal of inorganic contaminants has not been as widespread due to the low capacity and the difficulty in regenerating spent carbon, which subsequently require treatment and disposal. Also, the presence of iron can promote fouling of the carbon.
Aeration	Aeration effectively removes volatile organics and radon from groundwater, surface water and wastewater. A literature review of over sixty aeration systems showed radon removal efficiencies ranging from 78.6 to over 99% for packed tower aeration, 93 to 95% for diffuse bubble aerators, 71 to 100% for multi-stage bubble aerators, 35 to 99% for spray aerators, and 70 to 99% for tray aeration.	Pretreatment might be required to prevent fouling of the packing material and ensure the treatment's effectiveness.
Phytoremediation	Applicable media are groundwater and surface water. This methodology is typically implemented at low costs. Rhizofiltration has been shown in bench-scale testing to reduce water concentrations of europium and in field demonstrations to reduce water concentrations of caesium, strontium, and uranium.	Phytoremediation might be limited to lower levels of contamination due to plant toxicity effects. Climatic or seasonal conditions affect the growth of plants. It is not a rapid method to reach the remediation goal. Phytoremediation is limited to shallow groundwater and requires a large surface area of land.

7.5.2 Experiences/Case Studies

In the NEA report [681] there is a focus on the experiences of NEA member countries in nuclear site remediation. The report was prepared by the Task Group on Nuclear Site Restoration (TGNSR) which was formed through nominations from members participating in the Co-operative Programme for the Exchange of Scientific and Technical Information Concerning Nuclear Installation Decommissioning Projects (CPD), following a proposal submitted to the NEA Working Party on Decommissioning and Dismantling (WPDD). The task group gathered information at selected nuclear sites on experiences, approaches and techniques for remediation that minimise risks to workers and the environment, as well as costs and disruptions to decommissioning programmes. This was achieved using national level and project level questionnaires, detailed case studies and the experiences of task group members.

There were 28 site and project questionnaires to evaluate from 12 countries. The answers to the questionnaires revealed that the most important radionuclides detected in the groundwater are Sr-90 and U. C-14, Co-60, Tc-99, I-129, Ra-226 and TRU (Pu, Am-241) are mentioned only for one or two sites. It can further be observed that elements which are very mobile, like C, Sr, Tc and I, are observed both in ground and groundwater, as is expected, while the very mobile H-3 directly moves into the groundwater. U is observed in both media, indicating that it is rather mobile in certain chemical environments.

The questionnaire shows the different techniques which are used or are planned to be used for groundwater remediation. The answers show that pumping and treating the groundwater is one of the preferred methods for dealing with both radiological and non-radiological contamination. Groundwater monitoring takes place in at least one third of the projects. In a few cases, different techniques will be used (in-ground barrier, e.g. permeable reactive barrier, is a favourite method for remediation of radiological contamination). A few other techniques have been mentioned in the answers: pump and dispose for radiological contamination; bacteria addition and in-situ chemical transformation for non-radiological contamination. The methodologies mentioned in the answers are summarised in the following Table 7.5-2.

Table 7.5-2 Answers related to the different techniques in the NEA report [682]

Answer	Radiological contamination		Non-radiological contamin.	
	Number	Percentage	Number	Percentage
Pump and treat	4	33%	3	27%
In-ground barrier	4	33%	1	9%
Pump and re-inject	1	8%	3	27%
Monitoring	4	33%	4	36%
Other	2	17%	2	18%

[681] NEA - Nuclear Site Remediation and Restoration during Decommissioning of Nuclear Installations - 2014
[682] NEA - Nuclear Site Remediation and Restoration during Decommissioning of Nuclear Installations - 2014

7.6 Methodologies and techniques for final release survey of the Site

The survey for the final release of the site, also called final status survey, is the last step of a series of surveys designed to demonstrate compliance with a dose- or risk-based regulation for sites with radioactive contamination. There are four phases in the final status survey: planning, implementation, assessment and decision-making.

In the implementation phase, direct measurements are combined with scanning surveys and sampling. The level of survey effort is determined by the potential for contamination as indicated by the survey unit classification.

Instrumentation or measurement techniques should be selected based on detection sensitivity to provide technically defensible results that meet the objectives of the survey.

Many reference documents and standards are available to provide guidance on the release of sites or parts of sites from regulatory control after a practice has been terminated. [683, 684, 685, 686, 687,688]

7.6.1 Field Measurement Methods and Instrumentation

This chapter describe the methods and instrumentation already implemented according to the reference documents [689, 690, 691, 692, 693, 694, 695]

Three methods are available for collecting radiation data while performing a survey—direct measurements, scanning, and sampling.

Total surface activities, removable surface activities, and radionuclide concentrations in various environmental media (e.g., soil, water, air) are the radiological parameters typically determined using field measurements and laboratory analyses. Certain radionuclides or radionuclide mixtures may

---

[683] NUREG-1757, Vol.2 Rev.1, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria", September 2006

[684] IAEA Safety Guide No. WS-G-3.1, Remediation Process for Area Affected by Past Activities and Accidents, 2007

[685] IAEA Safety Guide No. WS-G-5.1, Release of Sites from Regulatory Control on Termination of Practices, 2006

[686] IAEA Technical Reports Series No. 424, Remediation of Sites with Dispersed Radioactive Contamination, 2004

[687] NEA-OECD No. 7290, Strategic Considerations for the Sustainable Remediation of Nuclear Installations, 2016

[688] ISO 18557:2017 Characterisation principles for soils, buildings and infrastructures contaminated by radionuclides for remediation purposes

[689] NUREG-1575, Rev. 1 / EPA 402-R-97-016, Rev. 1 /DOE / EH-0624, Rev. 1 "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)", August 2000

[690] NUREG-1507 "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions", June 1998

[691] NUREG-1575, Supp.1 / EPA 402-R-09-00 / DOE/HS-000 "Multi-Agency Radiation Survey and Assessment of Materials and Equipment Manual (MARSAME)", January 2009

[692] NUREG-1576 / EPA 402-B-04-001 / NTIS PB2004-105421 "Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP)", July 2004

[693] EUR 17624 - P.H. Burgess "Handbook on measurement methods and strategies at very low levels and activities", February 1998

[694] EPRI Technical Report, Groundwater and Soil Remediation Guidelines for NPP, 2011

[695] IAEA Safety Reports Series No. 72, Monitoring for Compliance with Remediation Criteria for Sites, 2012

necessitate the measurement of alpha, beta, and gamma radiations. In addition to assessing each survey unit as a whole, any small areas of high activity should be identified and their extent and activities determined. Due to numerous detector requirements, no single instrument (detector and readout combination) is generally capable of adequately measuring all of the parameters required to satisfy the release criterion or meet all the objectives of a survey.

The instrument and measurement method should be able to detect the type of radiation of interest, and should, in relation to the survey or analytical technique, be capable of measuring levels that are less than the derived concentration level (DCL). Numerous commercial firms offer a wide variety of instruments appropriate for the radiation measurements described hereafter. These firms can provide thorough information regarding capabilities, operating characteristics, limitations, etc., for specific equipment.

If the field instruments and measurement methods cannot detect radiation levels below the DCLs, laboratory methods are typically used.

There are certain radionuclides that will be essentially impossible to measure at the DCLs in situ using current state-of-the-art instrumentation and techniques because of the types, energies, and abundances of their radiations. Examples of such radionuclides include very low energy, pure beta emitters such as ^{3}H and ^{63}Ni and low-energy photon emitters such as ^{55}Fe and ^{125}I. Pure alpha emitters dispersed in soil or covered with some absorbing layer may not be detectable because alpha radiation will not penetrate through the media or covering to reach the detector.

7.6.1.1 Measurement Methods

Measurement methods used to generate field data can be classified into two categories commonly known as scanning surveys and direct measurements. The decision to use a measurement method as part of the survey design is determined by the survey objectives and the survey unit classification. Scanning is performed to identify areas of high activity that may not be detected by other measurement methods. Direct measurements are analogous to collecting and analyzing samples to determine the average activity in a survey unit. Scans and direct measurements can be combined in an integrated survey design.

7.6.1.1.1 Direct Measurements

To conduct direct measurements of alpha, beta, and photon surface activity, instruments and techniques providing the required detection sensitivity are selected. The type of instrument and method of performing the direct measurement are selected as dictated by the type of potential contamination present, the measurement sensitivity requirements, and the objectives of the radiological survey.

If the equipment and methodology used for scanning is capable of providing data of the same quality required for direct measurement (e.g., detection limit, location of measurements, ability to record and document results), then scanning may be used in place of direct measurements.

The following sections briefly describe methods used to perform direct measurements in the field.

Direct Measurements for Photon Emitting Radionuclides

There are a wide variety of instruments available for measuring photons in the field, but all of them are used in essentially the same way. The detector is set up at a specified distance from the surface being measured and data are collected for a specified period of time. A collimator may be used in areas where activity from adjacent or nearby areas might interfere with the direct measurement.

The measurements are typically performed using a germanium or a sodium iodide detector with a multichannel analyser. Germanium detectors have better resolution and can identify radionuclides at lower concentrations.

Sodium iodide detectors often have a higher efficiency and are significantly less expensive than germanium detectors. Low-energy photons (i.e., x-rays and gamma rays below 50 keV) can be measured using specially designed detectors with an entrance window made from a very light metal, typically beryllium.

Direct Measurements for Alpha Emitting Radionuclides

Direct measurements for alpha-emitting radionuclides are generally performed by placing the detector on or near the surface to be measured. The limited range of alpha particles (e.g., about 1 cm or 0.4 in. in air, less in denser material) means that these measurements are generally restricted to relatively smooth, impermeable surfaces such as concrete, metal, or drywall where the activity is present as surface contamination. In most cases, direct measurements of porous (e.g., wood) and volumetric (e.g., soil, water) material cannot meet the objectives of the survey.

However, special instruments such as the long range alpha detector have been developed to measure the concentration of alpha emitting radionuclides in soil under certain conditions.

Direct Measurements for Beta Emitting Radionuclides

Direct measurements for beta emitting radionuclides are generally performed by placing the detector on or near the surface to be measured, similar to measurements for alpha emitting radionuclides.

However, special instruments such as large area gas-flow proportional counters and arrays of beta scintillators have been developed to measure the concentration of beta emitting radionuclides in soil under certain conditions.

7.6.1.1.2 Scanning Surveys

Scanning is the process by which the operator uses portable radiation detection instruments to detect the presence of radionuclides on a specific surface (i.e., ground, wall, floor, equipment).

Small areas of high activity typically represent a small portion of the site or survey unit. Thus, random or systematic direct measurements or sampling on the commonly used grid spacing may have a low probability of identifying such small areas. Scanning surveys are often relatively quick and inexpensive to perform. For these reasons, scanning surveys are typically performed before direct measurements or sampling.

The following sections briefly describe techniques used to perform scanning surveys for different types of radiation.

Scanning for Photon Emitting Radionuclides

Sodium iodide survey meters (NaI(Tl) detectors) are normally used for scanning areas for gamma emitters because they are very sensitive to gamma radiation, easily portable and relatively inexpensive. Sodium iodide survey meters are also used for scanning to detect areas with elevated areas of low-energy gamma and x-ray emitting radionuclides such as ^{241}Am and ^{239}Pu.

Specially designed detectors, such as the FIDLER (field instrument for the detection of low energy radiation) probe with survey meter, are typically used to detect these types of radionuclides.

Scanning for Alpha Emitting Radionuclides

Alpha scintillation survey meters and thin window gas-flow proportional counters are typically used for performing alpha surveys.

In most cases, porous and volumetric contamination cannot be detected by scanning for alpha activity and meet the objectives of the survey because of high detection sensitivities. Under these circumstances, samples of the material are usually collected and analysed as discussed in sections 4.8 and 4.9.

Scanning for Beta Emitting Radionuclides

Thin window gas-flow proportional counters are normally used when surveying for beta emitters, although solid scintillators designed for this purpose are also available.

Low-energy (<100 keV) beta emitters are subject to the same interferences and self-absorption problems found with alpha emitting radionuclides, and scans for these radionuclides are performed under similar circumstances.

7.6.1.2 Sampling

Sampling is the process of collecting a portion of an environmental medium as representative of the locally remaining medium. The collected portion of the medium is then analysed to determine the radionuclide concentration.

Laboratory methods often involve combinations of both chemical and instrument techniques to quantify the low levels expected in the samples.

It is important to develop appropriate sample collection procedures for surveys to demonstrate compliance with a dose- or risk-based regulation. Sample collection procedures are concerned mainly with ensuring that a sample is representative of the sample media, is large enough to provide sufficient material to achieve the desired detection limit, and is consistent with assumptions used to develop the conceptual site model and the DCLs.

Surface Soil

The purpose of surface soil sampling is to collect samples that accurately and precisely represent the radionuclides and their concentrations at the location being sampled.

Building Surfaces

Because building surfaces tend to be relatively smooth and the radioactivity is assumed to be on or near the surface, direct measurements are typically used to provide information on contaminant concentrations.

7.6.1.2.1 Analytical Procedures

The selection of the appropriate radio analytical methods is normally made prior to the procurement of analytical services and is included in the statement-of-work of the request for proposal.

Specific equipment and procedures have to be used once the sample is prepared for analysis.

The decision maker and survey planning team should decide whether routine methods will be used at the site or if non-routine methods may be acceptable.

Photon Emitting Radionuclides

There is no special sample preparation required for counting samples using a germanium detector or a sodium iodide detector beyond placing the sample in a known geometry for which the detector has been calibrated.

The samples are typically counted using a germanium detector with a multichannel analyser or a sodium iodide detector with a multichannel analyser.

Data reduction is usually the critical step in measuring photon emitting radionuclides. There are often several hundred individual gamma ray energies detected within a single sample. Computer software is usually used to identify the peaks, associate them with the proper energy, associate the energy with one or more radionuclides, correct for the efficiency of the detector and the geometry of the sample, and provide results in terms of concentrations with the associated uncertainty.

Beta Emitting Radionuclides

Laboratory sample preparation is an important step in the analysis of surface soil and other solid samples for beta emitting radionuclides. The laboratory will typically have a sample preparation procedure for any kind of sample.

Measurements of solid samples are typically performed using a gas-flow proportional counter. Liquid samples are usually diluted using a liquid scintillation cocktail and counted using a liquid scintillation spectrometer. Liquid scintillation spectrometers can be used for low-energy beta emitting radionuclides, such as ^{3}H and ^{63}Ni. They also have high counting efficiencies, but often have a high instrument background as well. Gas flow proportional counters have a very low background.

Data reduction for beta emitting radionuclides is less complicated than that for photon emitting radionuclides. Since the beta detectors report total beta activity, the calculation to determine the concentration for the radionuclide of interest is straightforward.

Alpha Emitting Radionuclides

Laboratory sample preparation for alpha emitting radionuclides is similar to that for beta emitting radionuclides.

Because of the limited penetrating power of alpha particles, the preparation for counting is often a critical step. Gross alpha measurements can be made using small sample sizes with a gas-flow proportional counter, but self-absorption of the alpha particles results in a relatively high detection limit for this technique. Liquid scintillation spectrometers can also be used to measure alpha emitting radionuclides but the resolution limits the usefulness of this technique. Most alpha emitting radionuclides are measured in a vacuum (to limit absorption by air) using alpha spectroscopy. This method requires that the sample be prepared as a virtually weightless mount in a specific geometry. Electrodeposition is the traditional method for preparing samples for counting. This technique provides the highest resolution, but it requires a significant amount of training and expertise on the part of the analyst to produce a high quality sample. Precipitation of the radionuclide of interest on the surface of a substrate is often used to prepare samples for alpha spectroscopy.

Alpha emitting radionuclides are typically measured using alpha spectroscopy. The data reduction requirements for alpha spectroscopy are greater than those for beta emitting radionuclides, and similar to those for photon emitting radionuclides. Alpha spectroscopy produces a spectrum of alpha particles detected at different energies, but because the sample is purified prior to counting, all of the alpha particles come from radionuclides of a single element.

7.6.1.3 Instrument Selection

It is highly unlikely that any single instrument (detector and readout combination) will be capable of adequately measuring all of the radiological parameters necessary to demonstrate that criteria for release have been satisfied. It is usually necessary to select multiple instruments to perform the variety of measurements required.

Selection of instruments will require an evaluation of a number of situations and conditions.

Instruments must be stable and reliable under the environmental and physical conditions where they will be used, and their physical characteristics (size and weight) should be compatible with the intended application. The instrument must be able to detect the type of radiation of interest, and the measurement system should be capable of measuring levels that are less than the DCL.

For gamma radiation scanning, a scintillation detector/ratemeter combination is the usual instrument of choice. A large-area proportional detector with a ratemeter is recommended for scanning for alpha and beta radiations where surface conditions and locations permit; otherwise, an alpha scintillation or thin-window GM detector (for beta surveys) may be used.

For direct gamma measurements, a pressurized ionization chamber or in-situ gamma spectroscopy system is recommended. As an option, a NaI(Tl) scintillation detector may be used if cross-calibrated to a pressurized ion chamber or calibrated for the specific energy of interest.

The same alpha and beta detectors identified above for scanning surveys are also recommended for use in direct measurements.

There are certain radionuclides that, because of the types, energies, and abundances of their radiations, will be essentially impossible to measure at the guideline levels, under field conditions, using state-of-the-art instrumentation and techniques. Examples of such radionuclides include very low energy pure beta emitters, such as ^{3}H and ^{63}Ni, and low energy photon emitters, such as ^{55}Fe and ^{125}I. Pure alpha emitters dispersed in soil or covered with some absorbing layer will not be detectable because the alpha radiation will not penetrate through the media or covering to reach the detector. In such circumstances, sampling and laboratory analysis would be required to measure the residual activity levels unless surrogate radionuclides are present.

The number of possible design and operating schemes for each of the different types of detectors is too large to discuss in detail within the context of this document. For a general overview, lists of common radiation detectors along with their usual applications during surveys are provided in the following Table 7.6-1, Table 7.6-2 and Table 7.6-3.

Table 7.6-1 Radiation Detectors with Applications to Alpha Surveys

Detector Type	**Detector Description**	**Application**	**Remarks**
Gas Proportional	<1 mg/cm^2 window; probe area 50 to 1000 cm^2 <0.1 mg/cm^2 window; probe area 10 to 20 cm^2 No window (internal proportional)	Surface scanning; surface contamination measurement Laboratory measurement of water, air, and smear samples	Requires a supply of appropriate fill gas

Detector Type	Detector Description	Application	Remarks
		Laboratory measurement of water, air, and smear samples	
Air Proportional	<1 mg/cm^2 window; probe area ~50 cm^2	Useful in low humidity conditions	
Scintillation	ZnS(Ag) scintillator; probe area 50 to 100 cm^2 ZnS(Ag) scintillator; probe area 10 to 20 cm^2 Liquid scintillation cocktail containing sample	Surface contamination measurements, smears Laboratory measurement of water, air, and smear samples Laboratory analysis, spectrometry capabilities	
Solid State	Silicon surface barrier detector	Laboratory analysis by alpha spectrometry	
Passive, integrating electret ion chamber	<0.8 mg/cm^2 window, also window-less, window area 50-180 cm^2, chamber volume 50-1,000 ml	Contamination on surfaces, in pipes and in soils	Useable in high humidity and temperature

Table 7.6-2 Radiation Detectors with Applications to Beta Surveys

Detector Type	Detector Description	Application	Remarks
Gas Proportional	<1 mg/cm^2 window; probe area 50 to 1000 cm^2 <0.1 mg/cm^2 window; probe area 10 to 20 cm^2 No window (internal proportional)	Surface scanning; surface contamination measurement Laboratory measurement of water, air, and smear samples Laboratory measurement of water, air, and smear samples	Requires a supply of appropriate fill gas Can be used for measuring very low-energy betas
Ionization (non-pressurized)	1-7 mg/cm^2 window	Contamination measurements; skin dose rate estimates	
Geiger-Mueller	<2 mg/cm^2 window; probe area 10 to 100 cm^2 Various window thickness; few cm^2 probe face	Surface scanning; contamination measurements; laboratory analyses Special scanning applications	
Scintillation	Liquid scintillation cocktail containing sample Plastic scintillator	Laboratory analysis; spectrometry capabilities Contamination measurements	
Passive, integrating	7 mg/cm^2 window, also window-less, window area 50- 180 cm^2, chamber volume 50- 1,000 ml	Low energy beta including H-3	Useable in high humidity and temperature

electret ion chamber		contamination on surfaces and in pipes	

Table 7.6-3 Radiation Detectors with Applications to Gamma Surveys

Detector Type	Detector Description	Application	Remarks
Gas Ionization	Pressurized ionization chamber; Non-pressurized ionization chamber	Exposure rate measurements	
Geiger-Mueller	Pancake (<2 mg/cm^2 window) or side window (~30 mg/cm^2)	Surface scanning; exposure rate correlation (side window in closed position)	Low relative sensitivity togamma radiation
Scintillation	NaI(Tl) scintillator; up to 5 cm by 5 cm NaI(Tl) scintillator; large volume and “well” configurations CsI or NaI(Tl) scintillator; thin crystal Organic tissue equivalent (plastics)	Surface scanning; exposure rate correlation Laboratory gamma Spectrometry Scanning; low-energy gamma and x-rays Dose equivalent rate measurements	High sensitivity; Cross calibrate with PIC (or equivalent) or for specific site gamma energy mixture for exposure rate measurements. Detection of low-energy radiation

Detector Type	Detector Description	Application	Remarks
Solid State	Germanium semiconductor	Laboratory and field gamma spectrometry and spectroscopy	
Passive, integrating electret ion chamber	7 mg/cm^2 window, also window-less, window area 50-180 cm^2, chamber volume 50-1,000 ml		Useable in high humidity and temperature

7.6.2 Experiences/Case studies

This section describes the key issues and experiences that were derived from the twelve case studies considered in the NEA report[696] and summarised in the following Table 7.6-4 and from the international remediation experience and understanding.

Table 7.6-4 Remediation case studies

Case Study Number	Case Study Title	Country	Brief Description
1	CEA's Grenoble STED facility	France	Remediation of contaminated soil around and under redundant solid and liquid waste processing buildings.
2	Monts d'Arree, Brennilis	France	Clean-up of a waste water channel on the Brennilis site.
3	PIMIC rehabilitation project, CIEMAT	Spain	Remediation and waste management activities following decommissioning of a nuclear research facility.
4	Windscale Trenches, Sellafield	United Kingdom	Remediation of historical unlined low level waste disposal trenches. Enhanced capping selected as the remedial option for interim management.
5	Uranium Conversion Facility, Daejeon	Republic of Korea	Remediation following decommissioning of a uranium conversion facility.

[696] NEA-OECD No. 7192, Nuclear Site Remediation and Restoration during decommissioning of Nuclear Installations, 2014

Case Study Number	Case Study Title	Country	Brief Description
6	Fuel Assembly Plant, Hanau	Germany	Uranium contaminated soil and sediment under a fuel assembly plant was excavated.
7	618-10 Burial Ground, Hanford	United States	Removal of contaminated soil and debris from waste trenches is currently underway.
8	Site groundwater, Hanford	United States	A pump-and-treat system and natural attenuation are being used to treat contaminated groundwater at Hanford.
9	In-Situ Permeable Treatment Wall, West Valley	United States	A permeable treatment wall system replaced a pump-and-treat system that was not adequately treating 90Sr at a former fuel reprocessing plant at West Valley.
10	Laboratory building decommissioning at Chalk River Laboratories	Canada	Unplanned contamination in soil found under building during decommissioning.
11	“Lenteja” area remediation (PIMIC decommissioning project), CIEMAT	Spain	Site remediation of a contaminated area in CIEMAT.
12	Caustic cells, Chalk River Laboratories	Canada	Retrieval of historical buried radioactive wastes in waste management area.

Occasionally, characterisation methods that are commonly used may not include the appropriate technology to be used under the particular circumstances. Therefore, other methods must be found to obtain contamination data. As it has been observed from site characterisation efforts in the United Kingdom, geophysical surveys can only be useful if the contamination or contaminated wastes are distinct from the surrounding soil. If wastes or contaminated soil are too compact, and neighbouring soils are quite consolidated, the wastes may be geophysically indistinguishable from the neighbouring soils. Alternatively, in loose sandy soil, disturbed areas such as waste trenches may also be indistinguishable.

Geophysical surveys may also produce inconclusive results that may be misleading in other ways. Such surveying of waste trenches at Hanford in the United States led to assumptions that the trenches contained metal drums. These drums, of course, could potentially contain all types of radioactive materials. Upon excavation, however, the source of the significant metal anomalies turned out to be “page wire” fencing.

Even test pits can have limitations. An important consideration in deciding whether to carry out further characterisation is the potential level of benefit that could be realised were intrusive characterisation to be undertaken. At Sellafield, the contents of the waste trenches are believed to be very heterogeneous and their boundaries are not clearly known. Consequently, any single investigation will only build confidence in the understanding of the trench contents in the immediate vicinity of the test pit. Little benefit from such investigation can be anticipated unless significant excavation is involved. Moreover, as the wastes are often highly compacted, test pit intrusions might actually create contaminant migration pathways in the subsurface.

A test pit sampling plan was developed at Hanford in the United States using a biased, non-statistical design, based on previous geophysical studies and historical records. The results of trench sampling had similar limitations. The results of the sampling were valuable to note “hot spots”, but without knowledge of densities and potential interferences, the data did not provide further useful information. Beyond a radius of approximately one metre, even high activity waste was difficult to detect.

At Fernald, also in the United States, the same lesson was learnt when using Geoprobe sampling of subsurface soil contamination. Point sampling can provide very useful information, but cannot be expected to identify all the hot spots of contamination.

7.7 Tools for statistical analysis and management of survey data for site release

The survey for the final release of the site, also called final status survey, is the last step of a series of surveys designed to demonstrate compliance with a dose- or risk-based regulation for sites with radioactive contamination. There are four phases in the final status survey: planning, implementation, assessment and decision-making.

The assessment phase includes verification and validation of the survey results combined with an assessment of the quantity and quality of the data. Both the average level of contamination in the survey unit and the distribution of the contamination within the survey unit are considered during area classification. For this reason, the assessment phase includes a graphical review of the data to provide a visual representation of the radionuclide distribution, an appropriate statistical test to demonstrate compliance for the average concentration of a uniformly distributed radionuclide, and the high measurement comparison (HMC) to demonstrate compliance for small areas of high activity.

As already mentioned in Section 7.6, many reference documents and standards are available to provide guidance on the release of sites or parts of sites from regulatory control after a practice has been terminated [697, 698, 699, 700, 701,702]

7.7.1 Description of tools

This chapter describe the tools already available and implemented according to the reference documents [703, 704, 705, 706, 707, 708]

7.7.1.1 <u>Interpretation of Survey Results</u>

Interpreting a survey's results is most straightforward when measurement data are entirely higher or lower than the DCL. In such cases, the decision that a survey unit meets or exceeds the release

---

[697] NUREG-1757, Vol.2 Rev.1, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria", September 2006

[698] IAEA Safety Guide No. WS-G-3.1, Remediation Process for Area Affected by Past Activities and Accidents, 2007

[699] IAEA Safety Guide No. WS-G-5.1, Release of Sites from Regulatory Control on Termination of Practices, 2006

[700] IAEA Technical Reports Series No. 424, Remediation of Sites with Dispersed Radioactive Contamination, 2004

[701] NEA-OECD No. 7290, Strategic Considerations for the Sustainable Remediation of Nuclear Installations, 2016

[702] ISO 18557:2017 Characterisation principles for soils, buildings and infrastructures contaminated by radionuclides for remediation purposes

[703] NUREG-1505 "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys", Rev. 1, June 1998

[704] NUREG-1575, Rev. 1 / EPA 402-R-97-016, Rev. 1 /DOE / EH-0624, Rev. 1 "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)", August 2000

[705] NUREG-1757, Vol.2 Rev.1, "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria", September 2006

[706] EUR 17624 - P.H. Burgess "Handbook on measurement methods and strategies at very low levels and activities", February 1998

[707] Draft Regulatory Guide 4006 "Demonstrating Compliance with the Radiological Criteria for License Termination", August 1998

[708] Brownlee, K.A. "Statistical theory and Methodology in Science and Engineering", New York, Wiley, 1960

criterion requires little in terms of data analysis. However, formal statistical tests provide a valuable tool when a survey unit's measurements are neither clearly above nor entirely below the DCL. Nevertheless, the survey design always makes use of the statistical tests in helping to assure that the number of sampling points and the measurement sensitivity are adequate, but not excessive, for the decision to be made.

7.7.1.2 The Decision to Use Statistical Tests

The objective of compliance demonstration is to provide some level of confidence that the release criterion is not exceeded. 100% confidence in a decision cannot be proven because the data always contain some uncertainty. The use of statistical methods is necessary to provide a quantitative estimate of the probability that the release criterion is not exceeded at a particular site. Statistical methods provide for specifying (controlling) the probability of making decision errors and for extrapolating from a set of measurements to the entire site in a scientifically valid fashion.

The information needed to perform a statistical test is determined by the assumptions used to develop the test. It is recommended the use of nonparametric statistical tests because these tests use fewer assumptions, and consequently require less information to verify these assumptions. The tests described later in the document are relatively easy to implement compared to other statistical tests.

Two potential DCLs based on the area of contamination are defined:

- If the residual radioactivity is evenly distributed over a large area, it is looked at the average activity over the entire area. The DCL_W (the DCL used for the statistical tests) is derived based on an average concentration over a large area.
- If the residual radioactivity appears as small areas of high activity within a larger area, typically smaller than the area between measurement locations, it is considered the results of individual measurements. The DCL_{HMC} (the DCL used for the high measurement comparison – HMC) is derived separately for these small areas and generally from different exposure assumptions than those used for larger areas.

The "W" in DCL_W stands for Wilcoxon Rank Sum (WRS) test, which is the statistical test recommended for demonstrating compliance when the contaminant is present in background. The Sign test recommended for demonstrating compliance when the contaminant is not present in background also uses the DCLw.

The WRS and Sign tests are designed to determine whether or not the level of residual activity uniformly distributed throughout the survey unit exceeds the DCL_W. Since these methods are based on ranks, the results are generally expressed in terms of the median. When the underlying distribution is not symmetric, i.e. the mean is not equal to the median, these tests are still true tests of the median but only approximate tests of the mean. However, numerous studies show that this is a fairly good approximation (Hardin and Gilbert, 1993). The assumption of symmetry is less restrictive than that of normality because the normal distribution is itself symmetric. If, however, the measurement distribution is skewed to the right, the average will generally be greater than the median. In severe cases, the average may exceed the DCL_W while the median does not. For this reason, it is recommended comparing the arithmetic mean of the survey unit data to the DCL_W as a first step in the interpretation of the data.

7.7.1.3 Data Quality Assessment

Data Quality Assessment (DQA) is a scientific and statistical evaluation that determines if the data are of the right type, quality, and quantity to support their intended use. There are five steps in the DQA process:

- Review the Data Quality Objectives (DQOs) and Survey Design
- Conduct a Preliminary Data Review
- Select the Statistical Test
- Verify the Assumptions of the Statistical Test
- Draw Conclusions from the Data

The effort expended during the DQA evaluation should be consistent with the graded approach used in developing the survey design.

The Data Quality Objectives (DQO) Process is a powerful tool in developing appropriate survey designs to ensure that the survey results are of sufficient quality and quantity to support the final decision.

Review the Data Quality Objectives and Survey Design

The first step in the DQA evaluation is a review of the DQO outputs to ensure that they are still applicable.

The sampling design and data collection documentation should be reviewed for consistency with the DQOs. For example, the review should check that the appropriate number of samples were taken in the correct locations and that they were analyzed with measurement systems with appropriate sensitivity.

Determining that the sampling design provides adequate power is important to decision making, particularly in cases where the levels of residual radioactivity are near the DCL_W. This can be done both prospectively, during survey design to test the efficacy of a proposed design, and retrospectively, during interpretation of survey results to determine that the objectives of the design are met.

After the data are analyzed, a sample estimate of the data variability, namely the sample standard deviation(s) and the actual number of valid measurements will be known. The consequence of inadequate power is that a survey unit that actually meets the release criterion has a higher probability of being incorrectly deemed not to meet the release criterion.

Conduct a Preliminary Data Review

To learn about the structure of the data—identifying patterns, relationships, or potential anomalies—one can review quality assurance (QA) and quality control (QC) reports, prepare graphs of the data, and calculate basic statistical quantities.

Basic statistical quantities that should be calculated for the sample data set are the:

- mean
- standard deviation
- median

The average of the data can be compared to the reference area average and the DCL_W to get a preliminary indication of the survey unit status. Where remediation is inadequate, this comparison may readily reveal that a survey unit contains excess residual radioactivity—even before applying statistical tests.

The value of the sample standard deviation is especially important. If too large compared to that assumed during the survey design, this may indicate an insufficient number of samples were collected to achieve the desired power of the statistical test. Again, inadequate power can lead to unnecessary remediation.

50% of the data points are above the median, and 50% are below the median. Large differences between the mean and the median would be an early indication of skewness in the data. This would also be evident in a histogram of the data.

Examining the minimum, maximum, and range of the data may provide additional useful information.

Graphical Data Review

At a minimum, a graphical data review should consist of a posting plot and a histogram. Quantile plots are also useful diagnostic tools, particularly in the two-sample case, to compare the survey unit and reference area.

A posting plot is simply a map of the survey unit with the data values entered at the measurement locations. This potentially reveals heterogeneities in the data—especially possible patches of high residual radioactivity. Even in a reference area, a posting plot can reveal spatial trends in background data that might affect the results of the two-sample statistical tests.

If the posting plot reveals systematic spatial trends in the survey unit, the cause of the trends would need to be investigated. In some cases, such trends could be due to residual radioactivity, but may also be due to inhomogeneities in the survey unit background.

A frequency plot (or a histogram) is a useful tool for examining the general shape of a data distribution. This plot is a bar chart of the number of data points within a certain range of values.

The frequency plot will reveal any obvious departures from symmetry, such as skewness or bimodality (two peaks), in the data distributions for the survey unit or reference area. The presence of two peaks in the survey unit frequency plot may indicate the existence of isolated areas of residual radioactivity. In some cases it may be possible to determine an appropriate background for the survey unit using this information. The interpretation of the data for this purpose will generally be highly dependent on site-specific considerations and should only be pursued after a consultation with the responsible regulatory agency.

The presence of two peaks in the background reference area or survey unit frequency plot may indicate a mixture of background concentration distributions due to different soil types, construction materials, etc. The greater variability in the data due to the presence of such a mixture will reduce the power of the statistical tests to detect an adequately remediated survey unit. These situations should be avoided whenever possible by carefully matching the background reference areas to the survey units, and choosing survey units with homogeneous backgrounds.

Skewness or other asymmetry can impact the accuracy of the statistical tests. A data transformation (e.g., taking the logarithms of the data) can sometimes be used to make the distribution more symmetric.

Select the Tests

The most appropriate procedure for summarising and analyzing the data is chosen based on the preliminary data review. The parameter of interest is the mean concentration in the survey unit. The nonparametric tests, in their most general form, are tests of the median. If one assumes that the data are from a symmetric distribution—where the median and the mean are effectively equal—these are also tests of the mean. If the assumption of symmetry is violated, then nonparametric tests of the median approximately test the mean. Computer simulations have shown that the approximation is a good one. That is, the correct decision will be made about whether or not the mean concentration exceeds the DCL, even when the data come from a skewed distribution. In this regard, the nonparametric tests are found to be correct more often than the commonly used Student's t test. The robust performance of the Sign and WRS tests over a wide range of conditions is the reason that they are recommended.

When a given set of assumptions is true, a parametric test designed for exactly that set of conditions will have the highest power. It should be noted that for large number of measurements, the Student's t test is not a great deal more powerful than the nonparametric tests. On the other hand, when the assumption of normality is violated, the nonparametric tests can be very much more powerful than the t test.

Therefore, any statistical test may be used provided that the data are consistent with the assumptions underlying their use. When these assumptions are violated, the prudent approach is to use the nonparametric tests which generally involve fewer assumptions than their parametric equivalents.

Verify the Assumptions of the Tests

An evaluation to determine that the data are consistent with the underlying assumptions made for the statistical procedures helps to validate the use of a test. One may also determine that certain departures from these assumptions are acceptable when given the actual data and other information about the study. The nonparametric tests described in this chapter assume that the data from the reference area or survey unit consist of independent samples from each distribution. Spatial dependencies that potentially affect the assumptions can be assessed using posting plots. Asymmetry in the data can be diagnosed with a stem and leaf display, a histogram, or a Quantile plot. Data transformations can sometimes be used to minimise the effects of asymmetry.

One of the primary advantages of the nonparametric tests is that they involve fewer assumptions about the data than their parametric counterparts. If parametric tests are used, (e.g., Student's t test), then any additional assumptions made in using them should be verified (e.g., testing for normality).

One of the more important assumptions made in the survey design is that the sample sizes determined for the tests are sufficient to achieve the data quality objectives set for the Type I (α) and Type II (β) error rates. Verification of the power of the tests (1-β) to detect adequate remediation may be of particular interest.

If the hypothesis that the survey unit residual radioactivity exceeds the release criterion is accepted, there should be reasonable assurance that the test is equally effective in determining that a survey unit has residual contamination less than the DCL_W.

Otherwise, unnecessary remediation may result. For this reason, it is better to plan the surveys cautiously—even to the point of:

- overestimating the potential data variability
- taking too many samples
- overestimating minimum detectable concentrations (MDCs)

If one is unable to show that the objectives were met with reasonable assurance, a resurvey may be needed. Examples of assumptions and possible methods for their assessment are summarised in the following Table 7.7-1.

Table 7.7-1 Methods for Checking the Assumptions of Statistical Tests

Assumption	**Diagnostic**
Spatial Independence	Posting Plot
Symmetry	Histogram, Quantile Plot
Data Variance	Sample Standard Deviation
Power is Adequate	Retrospective Power Chart

Draw Conclusions from the Data

The types of measurements that can be made in a survey unit are:

1) direct measurements at discrete locations,
2) samples collected at discrete locations, and
3) scans.

The statistical tests are only applied to measurements made at discrete locations. When the data clearly show that a survey unit meets or exceeds the release criterion, the result is often obvious without performing the formal statistical analysis. The following Table 7.7-2 and Table 7.7-3 describe examples of circumstances leading to specific conclusions based on a simple examination of the data.

Table 7.7-2 Summary of Statistical Tests - Radionuclide not in background and radionuclide-specific measurements made

Survey Result	**Conclusion**
All measurements less than DCL_W	Survey unit meets release criterion
Average greater than DCL_W	Survey unit does not meet release criterion
Any measurement greater than DCL_W and the average less than DCL_W	Conduct Sign test and high measurement comparison

Table 7.7-3 Summary of Statistical Tests - Radionuclide in background or radionuclide non-specific (gross) measurements made

Survey Result	**Conclusion**
Difference between largest survey unit measurement and smallest reference area measurement is less than DCL_W	Survey unit meets release criterion
Difference of survey unit average and reference area average is greater than DCL_W	Survey unit does not meet release criterion
Difference between any survey unit measurement and any reference area measurement greater than DCL_W and the difference of survey unit average and reference area average is less than DCL_W	Conduct WRS test and high measurement Comparison

Both the measurements at discrete locations and the scans are subject to the high measurement comparison (HMC). The result of the HMC is not conclusive as to whether the survey unit meets or exceeds the release criterion, but is a flag or trigger for further investigation.

The investigation may involve taking further measurements to determine that the area and level of the high residual radioactivity are such that the resulting dose or risk meets the release criterion. The investigation should also provide adequate assurance that there are no other undiscovered areas of high residual radioactivity in the survey unit that might otherwise result in a dose or risk exceeding the release criterion. In some cases, this may lead to re-classifying all or part of a survey unit—unless the results of the investigation indicate that reclassification is not necessary.

7.7.1.4 Contaminant Not Present in Background

The statistical test discussed in this section is used to compare each survey unit directly with the applicable release criterion. A reference area is not included because the measurement technique is radionuclide-specific and the radionuclide of concern is not present in background. In this case the contaminant levels are compared directly with the DCL_W. The method in this section should only be used if the contaminant is not present in background or is present at such a small fraction of the DCL_W value as to be considered insignificant. In addition, one sample tests are applicable only if radionuclide-specific measurements are made to determine the concentrations. Otherwise, the method in Section 7.7.1.5 is recommended.

Reference areas and reference samples are not needed when there is sufficient information to indicate there is essentially no background concentration for the radionuclide being considered.

With only a single set of survey unit samples, the statistical test used here is called a one-sample test.

<u>**One-Sample Statistical Test**</u>

The Sign test is designed to detect uniform failure of remedial action throughout the survey unit.

This test does not assume that the data follow any particular distribution, such as normal or log-normal. In addition to the Sign Test, the DCL_{HMC} is compared to each measurement to ensure none exceeds the DCL_{HMC}. If a measurement exceeds this DCL, then additional investigation is recommended, at least locally, to determine the actual areal extent of the elevated concentration.

The hypothesis tested by the Sign test is:

Null Hypothesis

H0: The median concentration of residual radioactivity in the survey unit is greater than the DCL_W

Versus

Alternative Hypothesis

Ha: The median concentration of residual radioactivity in the survey unit is less than the DCL_W

The null hypothesis is assumed to be true unless the statistical test indicates that it should be rejected in favor of the alternative. The null hypothesis states that the probability of a measurement less than the DCL_W is less than one-half, i.e., the 50^{th} percentile (or median) is greater than the DCL_W. Note that some individual survey unit measurements may exceed the DCL_W even when the survey unit as a whole meets the release criterion. Such a survey unit may still not exceed the release criterion.

The assumption is that the survey unit measurements are independent random samples from a symmetric distribution. If the distribution of measurements is symmetric, the median and the mean are the same.

The hypothesis specifies a release criterion in terms of a DCL_W. The test should have sufficient power ($1-\beta$) to detect residual radioactivity concentrations at the Lower Boundary of the Gray Region (LBGR). If σ is the standard deviation of the measurements in the survey unit, then Δ/σ expresses the size of the shift (i.e., Δ = DCGLW - LBGR) as the number of standard deviations that would be considered "large" for the distribution of measurements in the survey unit.

7.7.1.5 Contaminant Present in Background

The statistical tests discussed in this section is intended to be used to compare each survey unit with an appropriately chosen, site-specific reference area. Each reference area should be selected on the basis of its similarity to the survey unit.

Two-Sample Statistical Test

The comparison of measurements from the reference area and survey unit is made using the Wilcoxon Rank Sum (WRS) test (also called the Mann-Whitney test). The WRS test should be conducted for each survey unit. In addition, the HMC is performed against each measurement to ensure that it does not exceed a specified investigation level. If any measurement in the remediated survey unit exceeds the specified investigation level, then additional investigation is recommended, at least locally, regardless of the outcome of the WRS test.

The WRS test is most effective when residual radioactivity is uniformly present throughout a survey unit. The test is designed to detect whether or not this activity exceeds the DCL_W. The advantage of the nonparametric WRS test is that it does not assume that the data are normally or log-normally distributed. The WRS test also allows for "less than" measurements to be present in the reference area and the survey units. As a general rule, the WRS test can be used with up to 40 percent "less than" measurements in either the reference area or the survey unit. However, the use of "less than" values in data reporting is not recommended.

When possible, report the actual result of a measurement together with its uncertainty.

The hypothesis tested by the WRS test is

Null Hypothesis

H0: The median concentration in the survey unit exceeds that in the reference area by more than the DCL_W

versus

Alternative Hypothesis

Ha: The median concentration in the survey unit exceeds that in the reference area by less than the DCL_W

The null hypothesis is assumed to be true unless the statistical test indicates that it should be rejected in favour of the alternative. One assumes that any difference between the reference area and survey unit concentration distributions is due to a shift in the survey unit concentrations to higher values (i.e., due to the presence of residual radioactivity in addition to background).

Note that some or all of the survey unit measurements may be larger than some reference area measurements, while still meeting the release criterion. Indeed, some survey unit measurements may exceed some reference area measurements by more than the DCL_W. The result of the hypothesis test determines whether or not the survey unit as a whole is deemed to meet the release criterion. The HMC is used to screen individual measurements.

Two assumptions underlying this test are:

1. samples from the reference area and survey unit are independent, identically distributed random samples, and
2. each measurement is independent of every other measurement, regardless of the set of samples from which it came.

7.7.1.6 Evaluating the Results: The Decision

Once the data and the results of the tests are obtained, the specific steps required to achieve site release depend on the procedures instituted by the governing regulatory agency and site-specific ALARA considerations. The following suggested considerations are for the interpretation of the test results with respect to the release limit established for the site or survey unit. Note that the tests need not be performed in any particular order.

High Measurement Comparison

The High Measurement Comparison (HMC) consists of comparing each measurement from the survey unit with the investigation levels. The HMC is performed for both measurements obtained on the systematic-sampling grid and for locations flagged by scanning measurements. Any measurement from the survey unit that is equal to or greater than an investigation level indicates an area of relatively high concentrations that should be investigated—regardless of the outcome of the nonparametric statistical tests.

The statistical tests may not reject H0 when only a very few high measurements are obtained in the survey unit. The use of the HMC against the investigation levels may be viewed as assurance that unusually large measurements will receive proper attention regardless of the outcome of those tests and that any area having the potential for significant dose contributions will be identified. The HMC is intended to flag potential failures in the remediation process. This should not be considered the primary means to identify whether or not a site meets the release criterion.

The derived concentration guideline level for the HMC is:

$DCL_{HMC} = Am \times DCL_{W}$

where Am is the area factor for the area of the systematic grid area. Note that $DCGL_{HMC}$ is an *a priori* limit, established both by the DCL_W and by the survey design (i.e., grid spacing and scanning MDC). The true extent of an area of high activity can only be determined after performing the survey and taking additional measurements. Upon the completion of further investigation, the *a posteriori* limit, $DCGL_{HMC} = Am \times DCL_W$, can be established using the value of Am appropriate for the actual area of high concentration. The area of high activity is generally bordered by concentration measurements below the DCL_W. An individual high measurement on a systematic grid could conceivably represent an area four times as large as the systematic grid area used to define the $DCGL_{HMC}$. This is the area bounded by the nearest neighbors of the high measurement location. The results of the investigation should show that the appropriate $DCGL_{HMC}$ is not exceeded.

If measurements above the stated scanning MDC are found by sampling or by direct measurement at locations that were not flagged by the scanning survey, this may indicate that the scanning method did not meet the objectives.

The preceding discussion primarily concerns Class 1 survey units. Measurements exceeding $DCGL_W$ in Class 2 or Class 3 areas may indicate survey unit mis-classification.

Interpretation of Statistical Test Results

The result of the statistical test is the decision to reject or not to reject the null hypothesis.

Provided that the results of investigations triggered by the HMC were resolved, a rejection of the null hypothesis leads to the decision that the survey unit meets the release criterion. However, estimating the average residual radioactivity in the survey unit may also be necessary so that dose or risk calculations can be made. This estimate is designated δ. The average concentration is generally the best estimator for δ. However, only the unbiased measurements from the statistically designed survey should be used in the calculation of δ.

If residual radioactivity is found in an isolated area of high activity—in addition to residual radioactivity distributed relatively uniformly across the survey unit—the unity rule can be used to ensure that the total dose is within the release criterion:

$$\frac{\delta}{DCGL_W} + \frac{(\text{average concentration in high area} - \delta)}{(\text{area factor for high area})(\text{DCGL}_\text{W})} < 1$$

If there is more than one high area, a separate term should be included for each. When calculating δ for use in this inequality, measurements falling within the high area may be excluded providing the overall average in the survey unit is less than the $DCGL_W$. As an alternative to the unity rule, the dose or risk due to the actual residual radioactivity distribution can be calculated if there is an appropriate exposure pathway model available. Note that these considerations generally apply only to Class 1 survey units, since areas of high activity should not exist in Class 2 or Class 3 survey units.

A retrospective power analysis for the test will often be useful, especially when the null hypothesis is not rejected. When the null hypothesis is not rejected, it may be because it is in fact true, or it may be because the test did not have sufficient power to detect that it is not true. The power of the test will be primarily affected by changes in the actual number of measurements obtained and their standard deviation. An effective survey design will slightly overestimate both the number of measurements and the standard deviation to ensure adequate power. This insures that a survey unit is not subjected to additional remediation simply because the final status survey is not sensitive enough to detect that residual radioactivity is below the guideline level. When the null hypothesis is rejected, the power of the test becomes a somewhat moot question. Nonetheless, even in this case, a retrospective power curve can be a useful diagnostic tool and an aid to designing future surveys.

7.7.1.7 If the Survey Unit Fails

What if at any point the survey unit should fail? This is primarily because there are many different ways that a survey unit may fail the final status survey. The overall level of residual radioactivity may not pass the nonparametric statistical tests. Further investigation following the high measurement comparison may show that there is a large enough area with a concentration too high to meet the release criterion. Investigation levels may have caused locations to be flagged during scanning that indicate unexpected levels of residual radioactivity for the survey unit classification. Site-specific information is needed to fully evaluate all of the possible reasons for failure, their causes, and their remedies.

When a survey unit fails to demonstrate compliance with the release criterion, the first step is to review and confirm the data that led to the decision. Once this is done, it is possible to identify and evaluate potential solutions to the problem. The level of residual radioactivity in the survey unit should be determined to help define the problem.

Once the problem has been stated the decision concerning the survey unit should be developed into a decision rule. Next, determine the additional data, if any, needed to document that the survey unit demonstrates compliance with the release criterion. Alternatives to resolving the decision statement should be developed for each survey unit that fails the tests. These alternatives are evaluated against the Objectives, and a survey design that meets the objectives of the project is selected.

7.7.1.8 Alternate Statistical Methods

The use of statistics to provide a quantitative estimate of the probability that the release criterion is not exceeded at a site is encouraged. While it is unlikely that any site will be able to demonstrate compliance with a dose- or risk-based regulation without at least considering the use of statistics, it is recognized that the use of statistical tests may not always provide the most effective method for demonstrating compliance. For example, it is recommended a simple comparison to an investigation level to evaluate the presence of small areas of high activity in place of complicated statistical tests. At some sites a simple comparison of each measurement result to the DCL_W, to demonstrate that all the measurement results are below the release criterion, may be more effective than statistical tests for the overall demonstration of compliance with the regulation provided an adequate number of measurements are performed.

Moreover, it is recommended that nonparametric statistical tests are used for evaluating environmental data. There are two reasons for this recommendation:

1. environmental data is usually not normally distributed, and
2. there are often a significant number of qualitative survey results (e.g., less than MDC).

Either one of these conditions means that parametric statistical tests may not be appropriate. If one can demonstrate that the data are normally distributed and that there are a sufficient number of results to support a decision concerning the survey unit, parametric tests will generally provide higher power (or require fewer measurements to support a decision concerning the survey unit). The tests to demonstrate that the data are normally distributed generally require more measurements than the nonparametric tests.

There are a wide variety of statistical tests designed for use in specific situations. These tests may be preferable to the generic statistical tests recommended in this document when the underlying assumptions for these tests can be verified. The following Table 7.7-4 lists several examples of statistical tests that may be considered for use at individual sites or survey units. A brief description of the tests and references for obtaining additional information on these tests are also listed in the Table 7.7-4. Applying these tests may require consultation with a statistician.

Table 7.7-4 Examples of Alternate Statistical Test

Alternate Tests	Probability Model Assumed	Type of Test	Reference	Advantages	Disadvantages
Alternate 1-Sample Tests (no reference area measurements)					
Student's t Test	Normal	Parametric test for Ho: Mean < L	*Guidance for Data Quality Assessment,* EPA QA/G-9, p. 3.2-2.	Appropriate if data appears to be normally distributed and symmetric.	Relies on a non-robust estimator for μ and 1. Sensitive to outliers and departures from normality.
t Test Applied To Logarithms	Lognormal	Parametric test for Ho: Median < L	*Guidance for Data Quality Assessment,* EPA QA/G-9, p. 3.2-2	This is a well- known and easy-to-apply test. Useful for a quick summary of the situation if the data is skewed to right.	Relies on a non-robust estimator for 1. Sensitive to outliers and departures from lognormality.
Minimum Variance Unbiased Estimator For Lognormal Mean	Lognormal	Parametric estimates for mean and variance of lognormal distribution	Gilbert, *Statistical Methods for Environmental Pollution Monitoring,* p. 164, 1987.	A good parametric test to use if the data is lognormal.	Inappropriate if the data is not lognormal.
Chen Test	Skewed to right, including Lognormal	Parametric test for Ho: Mean > 0	*Journal of the American Statistical Association (90),* p.767, 1995.	A good parametric test to use if the data is lognormal.	Applicable only for testing Ho: "survey unit is clean." Survey unit must be significantly greater than 0 to fail. Inappropriate if the data is not skewed to the right.
Alternate 1-Samples Tests (no reference area measurements)					
Bayesian Approaches	Varies, but a family of probability distribution must be selected.	Parametric test for Ho: Mean < L	DeGroot, *Optimal Statistical Decisions*, p. 157, 1970.	Permits use of subjective "expert judgment" in interpretation of data.	Decisions based on expert judgment may be difficult to explain and defend.

Alternate Tests	Probability Model Assumed	Type of Test	Reference	Advantages	Disadvantages
Bootstrap	No restriction	Nonparametric. Uses resampling methods to estimate sampling variance.	Hall, *Annals of Statistics (22)*, p. 2011-2030, 1994.	Avoids assumptions concerning the type of distribution.	Computer intensive analysis required. Accuracy of the results can be difficult to assess.
Lognormal Confidence Intervals Using Bootstrap	Lognormal	Uses resampling methods to estimate one-sided confidence interval for lognormal mean.	Angus, *The Statistician,* p. 395, 1994.	Nonparametric method applied within a parametric lognormal model.	Computer intensive analysis required. Accuracy of the results can be difficult to assess.
Alternate 2-Sample Tests (reference area measurements are required)					
Student's t Test	Symmetric, normal	Parametric test for difference in means Ho: μx < μy	*Guidance for Data Quality Assessment,* EPA QA/G-9, p. 3.3-2	Easy to apply. Performance for non-normal data is acceptable.	Relies on a non-robust estimator for 1, therefore test results are sensitive to outliers.
Mann-Whitney Test	No restrictions	Nonparametric test difference in location Ho: μx < μy	Hollander and Wolfe, *Nonparametric Statistical Methods,* p. 71, 1973.	Equivalent to the WRS test, but used less often. Similar to resampling, because test is based on set of all possible differences between the two data sets.	Assumes that the only difference between the test and reference areas is a shift in location.
Kolmogorov-Smirnov	No restrictions	Nonparametric test for any difference between	Hollander and Wolfe, *Nonparametric Statistical*	A robust test for equality of two sample distributions against all alternatives.	May reject because variance is high, although mean is in compliance.

Alternate Tests	Probability Model Assumed	Type of Test	Reference	Advantages	Disadvantages
		the 2 distributions	*Methods,* p. 219, 1973.		
Bayesian Approaches	Varies, but a family of probability distributions must be selected	Parametric tests for difference in means or difference in variance.	Box and Tiao, *Bayesian Inference in Statistical Analysis*, Chapter 2, 1973.	Permits use of "expert judgment" in the interpretation of data.	Decisions based on expert judgement may be difficult to explain and defend.
Alternate 2-Sample Tests (reference area measurements are required)					
2-Sample Quantile Test	No restrictions	Nonparametric test for difference in shape and location.	EPA, *Methods for Evaluating the Attainment of Cleanup Standards, Vol. 3*, p. 7.1, 1992.	Will detect if survey unit distribution exceeds reference distribution in the upper quantiles.	Applicable only for testing Ho: "survey unit is clean." Survey unit must be significantly greater than 0 to fail.
Simultaneous WRS and Quantile Test	No restrictions	Nonparametric test for difference in shape and location.	EPA, *Methods for Evaluating the Attainment of Cleanup Standards, Vol. 3*, p. 7.17, 1992.	Additional level of protection provided by using two tests. Has advantages of both tests.	Cannot be combined with the WRS test that uses Ho: "survey unit is not clean." Should only be combined with WRS test for Ho: "survey unit is clean."
Bootstrap and Other Resampling Methods	No restrictions	Nonparametric. Uses resampling methods to estimate	Hall, *Annals of Statistics (22)*, p. 2011, 1994.	Avoids assumptions concerning the type of distribution.	Generates informative resampling distributions for graphing. Computer intensive analysis required.

Alternate Tests	Probability Model Assumed	Type of Test	Reference	Advantages	Disadvantages
		sampling variance.			
Alternate to Statistical Tests					
Decision Theory	No restrictions	Incorporates loss function in the decision theory approach.	DOE, *Statistical and Cost-Benefit Enhancements to the DQO Process for Characterisation Decisions*, 1996.	Combines elements of cost-benefit analysis and risk assessment into the planning process.	Limited experience in applying the method to compliance demonstration and decommissioning. Computer intensive analysis required.

7.7.1.9 Accelerated Cleanup Models

There are a number of approaches designed to expedite site cleanups. These approaches can save time and resources by reducing sampling, preventing duplication of effort, and reducing inactive time periods between steps in a cleanup process. Although this document describes the final status survey as one step of a series of surveys, there are many examples of accelerated cleanup approaches.

The Superfund Accelerated Cleanup Model (SACM), which includes a module called integrated site assessment, has as its objectives increased efficiency and shorter response times.

7.7.2 Experiences/Case Studies [709]

7.7.2.1 Sandia National Laboratories

Sandia National Laboratories (SNL) uses the Observational Approach. This approach uses an iterative process of sample collection and real-time data evaluation to characterise a site. This process allows early field results to guide later data collection in the field. Data collection is limited to only that required for selecting a unique remedy for a site.

[709] NUREG-1575, Rev. 1 / EPA 402-R-97-016, Rev. 1 /DOE / EH-0624, Rev. 1 “Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)”, August 2000

7.7.2.2 DOE's Hanford Site

At DOE's Hanford Site, the Hanford Past Practice Strategy (HPPS) was developed in early 90s to accelerate decision-making and initiation of remediation through activities that include maximising the use of existing data consistent with data quality objectives.

7.7.2.3 Guidance for Management of contaminated soils in France

French safety authority published a Guide called "Guide 24: Management of soils contaminated by the activities of a basic nuclear installations in France"[710]. This guide is intended for basic nuclear installation (BNI) operators at sites where soil contamination leading to the undertaking of a remediation or soil management procedure has been detected. The guide outlines the procedure for managing and cleaning contaminated soils, including classification, excavation, and disposal of the soil. The guide was developed in conjunction with IRSN and ASND to clarify and harmonise the guidance relating to soil remediation in documents issued by several organisations

[710] http://www.french-nuclear-safety.fr/References/ASN-Guides-non-binding/ASN-Guide-No.-24

8. Management of material and radioactive waste from decommissioning

This chapter, complementary to chapter 6.8 is addressing predisposal management of radioactive waste, once it is removed during dismantling operation; this includes a long-term task with several different activities like processing, storage and transportation from the generation to its disposal, but not including the final waste disposal.

The overall goal of this set of tasks is the minimisation of the radioactive waste in terms of type, volume and activity but also not forgetting the safety of the workers. Therefore, an effective integrated management system applied to all steps of the predisposal waste treatment and conditioning is needed.

International initiatives

A number of different initiatives [711], [712], [713]have been promoted in the past years by IAEA and NEA. More recent initiatives are described in the following.

IAEA Initiatives

- IAEA's Specific Safety Guide No. SSG-40[714] on Predisposal Management states in specific the following steps in this task:
  - Pretreatment, which may include waste assay and characterisation, waste collection, waste segregation, chemical adjustment and decontamination.
  - Treatment, which may include volume reduction, removal of radionuclides and changing the composition of the waste.
  - Conditioning, which involves those operations that transform radioactive waste into a suitable form for subsequent activities such as handling, transport, storage and disposal; conditioning may include immobilisation of the waste, placing of the waste into containers and provision of additional packaging.
  - Storage, which refers to the temporary placement of radioactive waste in a facility where appropriate isolation and monitoring is provided. Storage is an interim activity performed with the intent to retrieve the waste later for clearance from regulatory control, for authorized use (e.g. after a decay period), for processing and/or for disposal, or in the case of effluent, for authorized discharge. (Pub1719web-85295…)."

  This guide gives a list of key points to consider when establishing the management system:

  - The preservation of technology, knowledge, and the transfer of such knowledge to individuals joining the operating organisation in the future.

[711] Policies and Strategies for the Decommissioning of Nuclear and Radiological Facilities, NW-G-2.1, IAEA, 2011
[712] IAEA (2007), Disposal Aspects of Low and Intermediate Level Decommissioning Waste – Results of a Co-ordinated Research Project 2002-2006, IAEA-TECDOC-1572, IAEA, Vienna.
[713] NEA (2008), Regulating the Decommissioning of Nuclear Facilities: Relevant Issues and Emerging Practices, OECD, Paris
[714] IAEA, "Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors", IAEA Safety Standards Series, Specific Safety Guide No. SSG-40, Vienna (2016).

- The retention or transfer of ownership of the radioactive waste and the waste management facility.
- Succession planning for technical human resources and managerial human resources.
- The continuation of arrangements for interacting with interested parties.
- The provision of adequate financial resources (the adequacy of resources for maintenance and decommissioning of facilities and equipment may need to be reviewed periodically over operational periods that may extend over decades).
- The preservation and quality of records and information (e.g. details of radioactive waste inventories; records relating to siting, design, commissioning, operation and decommissioning of the facility; and records relating to the development of the safety case).
- Provision for review to ensure that the goals of the management system can still be met.

The guide points out that "With a good knowledge of waste properties, it is possible to segregate the waste for treatment and conditioning. Documented procedures should be followed for uniform characterisation and segregation and waste should be designed according to the documented categories. Special attention should be given to the waste containing flammable, pyrophoric, corrosive or other hazardous materials and be stored separately."

It also gives basic principles for the segregation of waste after production:

- Material containing predominantly short-lived radionuclides should not be mixed with waste containing long-lived radionuclides.
- Is clearance from regulatory control or recycling or discharge, either directly or after a period of storage for radioactive decay possible?
- Mechanical, chemical or electrochemical decontamination methods are used to remove the surface of the material.
- The possibility of chemical reactions and acceptance criteria has to be taken into account and special care should be taken when mixing of waste is allowed.

Considering treatment, the guide considers that the treatment of radioactive waste may include the following:

- Reduction in the volume of the waste (e.g. by incineration of combustible waste, compaction of solid waste and segmentation or disassembly of bulky waste components or equipment).
- Removal of radionuclides (e.g. by evaporation or ion exchange for liquid waste streams and filtration of gaseous waste streams).
- Change of the form or composition of the waste (e.g. using chemical processes such as precipitation, flocculation and acid digestion, as well as by chemical or thermal oxidation).
- Change of the form or properties of the waste (e.g. solidification, sorption or encapsulation; common immobilisation matrices include cement, bitumen and glass)."

The guide also describes the process of packing of the waste material into suitable packages for safe handling, transport and storage. These packages should meet the respective acceptance criteria. Shielding of the container might be needed depending on waste characteristics and the handling methods. It is also possible to store a container in a second one to meet acceptance criteria and ensure ease of decontamination.

- IAEA-TECDOC-1817, “Selection of Technical Solutions for the Management of Radioactive Waste”, 2017. This document has the objective to identify and critically review the criteria to be considered while selecting waste management technologies; summarise, evaluate, rank and compare the different technical solutions; and offer a systematic approach for selecting the best matching solution.

- IAEA-TECDOC-1130 "Recycle and Reuse of Materials and Components from Waste Streams of Nuclear Fuel Cycle Facilities," 2000. This report analyses the existing options, approaches and developments in recycle and reuse in nuclear industry.

NEA Initiatives

- The Nuclear Energy Agency (NEA) Working Party on Decommissioning and Dismantling (WPDD) established an expert group in 2016 – the Task Group on Optimising Management of Low-Level Radioactive Materials and Waste from Decommissioning (TGOM) – to examine how different countries manage their (very) low-level radioactive waste and materials arising from decommissioning. The expert group considered all the steps of the waste management life cycle, from generation during dismantling to the final destination, whether it involved clearance, recycling or disposal to a landfill or to a repository.

  The final report explores the elements contributing to optimisation in national approaches at the strategic level, describing the main factors involved and the relationship between them. It also identifies constraints in the practical implementation of optimisation, based on experience in NEA member countries.

- NEA Co-operative Programme on Decommissioning (CPD) commissioned a Task Group on Recycling and Reuse of Materials (TGRRM)

  The final report [715] is a review of the 20 years of decommissioning experience since the initial 1996 report on Recycling and Reuse of Metals (NEA, 1996) was prepared.

European Commission Initiatives:

The European Commission organised a number of initiatives:

- H2020 PREDIS “Predisposal Management of Radioactive Waste”

  The PREDIS project targets innovation and break-through technologies for safer, more efficient, more economic, and more environmentally-friendly handling of ILW/LLW radioactive wastes. The focus is on conditioning of metallic materials, liquid organic wastes and solid organic wastes arising from nuclear plant operations, decommissioning and other

[715] NEA OECD, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017

industrial processes. The project also addresses digitalisation solutions for improvements in handling and assessing cemented-waste packages in extended interim surface storage.

PREDIS consists of four technical R&D work packages (WP4-7), a strategic work package (WP2) and a knowledge management work package (WP3).

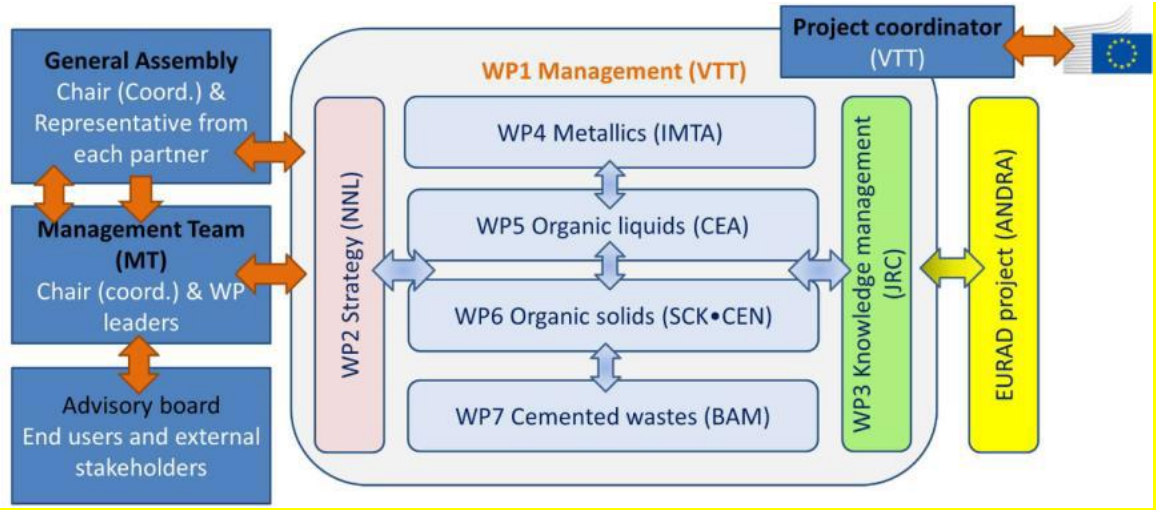


Figure 8-1 PREDIS project organisation

Regarding the technical Work Packages here are the summarised the main objectives.

Work Package 4 - Innovations in metallic treatment and conditioning:

- minimise through new and / or optimised treatment and decontamination processes, and more efficient characterisation of the amount of metallic waste to be disposed of in disposal facilities, by allowing more efficient clearance and recycling;
- contribute to the development of a new reference, stable and safe solution for the storage and final disposal of metallic wastes, including reactive metals, such as aluminium and beryllium;
- estimate of the potential scale of the opportunity to optimise the management of European metallic wastes, including quantification of the benefits in economic terms and application of the waste hierarchy.

Work Package 5 - Innovations in liquid organic waste treatment and conditioning:

- implementing geopolymers and related alkali-activated materials as mineral binders;
- fulfilling technical and economic requirements related to Radioactive Liquid Organic Waste (RLOW) (robustness regarding waste variability;
- ease of implementation and operation, possibly mobile; capacity ranging from low (a few liters) to large volumes (a few tens of m^3) of waste batches; limitation of secondary waste; reduction of disposal cost by minimisation of volumes allowed by pre-treatment and optimisation of waste incorporation rates);
- leading to a final wasteform showing properties and performances compatible with safety and technical requirements related to disposal but also prolonged storage and transport. Disposability assessment and demonstration is a key issue and challenge of

WP5. In particular, applicability of the direct conditioning route according to RLOW radiological categories (VLLW, LLW/ILW-SL, ILW-LL) and according to disposal facilities features (near-surface and/or intermediate-depth and/or geological) which has to be investigated and analysed.

Work Package 6 - Innovations in solid organic waste treatment and conditioning:

- Demonstrate the reliability of alkaline binders for conditioning of residues and secondary wastes stemming from treatment of Radioactive Solid Organic Waste (RSOW). The alkaline binders investigated in PREDIS include both (common) cementitious materials and (novel) materials like geopolymers.
- Verify the matrix performance of conditioned final/ultimate waste according to a set of uniform Waste Acceptance Criteria (WAC).
- Apart from these objectives, PREDIS will also build further on the achievements and deliverables of the THERAMIN project, which is scheduled to end in May 2020. Additional objectives related to this project include:
  - Improve understanding of materials inventory before the thermal treatment and of the reconditioned wastes once the conversion and immobilisation has been achieved.
  - Demonstrate thermal treatment methods leading to a significant volume reduction and to safe reconditioned waste packages.
  - Deploy results for safe utilisation by end users for mathematical calculations, avoiding systematic experimental studies of the reconditioned wastes.

Work Package 7 - Innovations in cemented waste handling and pre-disposal storage:

- Compile information about the state of the art of current methods and procedures for cemented waste management with specific focus on monitoring/long-term storage;
- Identify, evaluate and demonstrate store and package quality assurance (mainly NDE) and monitoring technologies;
- Adapt and demonstrate digital twin technology;
- Develop and demonstrate methods for data handling;
- Develop and demonstrate a digital decision framework;
- Identify opportunities for increased store automation, reducing human exposure to radiation;
- Identify options for post treatment of packages and potential approaches to improve package design, construction and maintenance.

The state of the art for all the technical Work Packages will be addressed in the 1st year of the project with formalisation of stakeholders needs.

- H2020 THERAMIN Thermal treatment for radioactive waste minimisation and hazard reduction

  The aim of the THERAMIN project is to provide improved safe long-term storage and disposal of intermediate and low level radioactive waste streams (ILW and LLW), suitable for thermal processing. The work programme provides a vehicle for coordinated EU-wide research and technology demonstration designed to provide improved understanding and optimisation of the application of thermal treatment in radioactive waste management programmes across Europe, improving the technology readiness level to accelerate industrial implementation. Technologies demonstrated in the project included direct Joule heating, plasma melting, thermal gasification, etc.

- H2020 CHANCE "Characterisation of Conditioned Nuclear Waste for its safe Disposal in Europe"
  See International initiatives in Chapter 4

- H2020- MICADO "Measurement and Instrumentation for Cleaning And Decommissioning Operations"
  This project is addressing characterisation of packaged waste for the in-field Waste Management (historical waste retrieval operations and waste from decommissioning). Thus, complementary information concerning methodology and technologies is given in Chapter 4.

- ELINDER
  See International initiatives in Chapter 4

Other Initiatives

- In 2015, the Waste Management & Decommissioning Working Group of World Nuclear Association decided to produce a report that would bring together this knowledge and expertise, to provide guidance to those facing new decommissioning challenges.

  During the plenary sessions of the Waste Management & Decommissioning Working Group and the Annual Symposium of World Nuclear Association, the status of the report was regularly presented to the nuclear community. This allowed the authors to continuously develop and update the report by taking into account new processes, improvements and events. In addition, the authors cooperated with other international organisations (such as the International Atomic Energy Agency, OECD Nuclear Energy Agency and the European Commission) to ensure that the report would complement the findings and objectives of these organisations.

  It is intended that Methodology to Manage Material and Waste from Nuclear Decommissioning [716] will serve as a practical guide to decommissioning nuclear plants, allowing both established nuclear stakeholders and those new to the industry to learn from past experience. It outlines international good practice and gives details on potential methodologies for decommissioning and dismantling waste management programmes.

[716] World Nuclear Association "Methodology to Manage Material and Waste from Nuclear Decommissioning" 2019

8.1 Management routes for materials including radioactive waste streams

Nuclear decommissioning projects are largely driven by the application of the waste hierarchy (Figure 8.1-1). The waste hierarchy provides a tool to which methods of waste management has the most beneficial impact on the environment. Prevention is at the top of the diagram (most preferable) followed by reuse, recycling, recovery and then disposal at the bottom (least preferable).

Figure 8.1-1 Waste Hierarchy

Waste is appropriately categorised based on its hazard and origin. The classification is to ensure that the radioactive waste is managed appropriately. The three categories are: low-level waste (LLW), intermediate-level waste (ILW), and high-level waste (HLW). Very low-level waste (VLLW) is a sub-category of LLW with specific activity limits. LLW contains relatively low levels of radioactivity, not exceeding 4 gigabecquerel (GBq) per tonne of alpha activity, or 12 GBq per tonne of beta/gamma activity. ILW exceeds the upper boundaries for Low Level Waste but does not generate a significant amount of heat. Whilst, HLW is waste where the temperature may rise significantly because of their radioactivity. Further descriptions on the classification and treatment processes of waste can be found in Section 8.3.

Decommissioning and waste management activities are highly reliant on one another. Hence, integrating the strategy for both activities strengthens the approach. The availability of waste management routes is essential to any decommissioning project. However, decommissioning wastes may not meet existing waste management routes (and their specified criteria/specification), which means that conditioning treatment and/or storage will be required.

In the absence of a final disposal route, on-site waste treatment and storage facilities are required to allow the progression of dismantling operations. An analysis of policies and procedures for radioactive waste reduction, such as recycling or clearance, should be conducted when setting the strategy for decommissioning and the documentation should be regularly reviewed. The waste management plan should consider the presence of potentially recoverable material or equipment (reuse or recycle within or outside the nuclear sector). In most cases, the dismantling of nuclear facilities produces a high volume of material, that subject to clearance, can be classified as conventional material. However, material which is both contaminated and hazardous, can be considered as problematic, as no waste route is available for disposal. The management of hazardous and toxic materials is discussed in further detail in Section 8.8.

Figure 8.1-2 Disposal facility of El Cabril, Spain conditioning facility (left) and Very low level (VLL) waste disposal area (right)

A key tool that can be used in the management of materials from nuclear facilities is a materials' management strategy. A successful materials' management strategy should incorporate the overall lifecycle, from the waste generated during development and design up to disposal. The strategy should outline the type of material produced, whether the waste will fit into any pre-existing disposal routes as outlined within the Waste Acceptance Criteria (WAC), the clearance criteria, and how characterisation, decontamination, dismantling and waste treatment will be implemented. The waste management strategy should include:

- An inventory of solid, aqueous and gaseous materials/waste (in nuclear and conventional areas of the facility);
- A waste management plan for the radioactive and potentially radioactive waste, to ensure that the conditioned waste meets the WAC;
- The auxiliary systems and infrastructure for handling materials, interim storage and transportation capabilities including reviewing the existing facilities for on-site treatment;
- The selected technologies which will be used for treatment;
- The material and waste disposal routes (and their availability).

The high-level decision for on-site waste treatment is closely associated to the country's waste acceptance criteria for storage and disposal, clearance criteria (if applicable), and transportation limitations.

Figure 8.1-3 Auxiliary facilities for material treatment and temporary storage on site (Jose Cabrera Nuclear Power Station, Spain)

Prior to undertaking dismantling activities, the necessary waste routes and possible interim storage facilities should ideally already be available. This approach allows time between the process of generating waste via dismantling the facility, from the material's transfer to a long-term disposal site. This reduces the risk of delays being caused by materials and waste management logistics and provides the opportunity for more time to establish the waste routes for problematic waste streams.

The waste is required to undergo detailed characterisation to ensure that the material complies with the waste stream's specification. There are many logistical challenges with managing the materials and radioactive waste in a decommissioning project.

The main purpose of a logistics plan is to avoid resourcing bottlenecks during the decommissioning project and thus prevent delays. The logistics plan should be developed in accordance with the waste management strategy, the dismantling strategy and the facility modification design. The planning of the dismantling logistics should be considered in the early planning of decommissioning.

One of the key challenges in waste management is the diversity of radioactive waste. The range of components that may be considered could include:

- Large components (for example from the primary circuit);
- Highly radioactive systems (for example activated/contaminated components from a reactor's core);
- Large volume of materials with low/minimal contamination (for example contaminated soils).

Decommissioning and waste management have not been historically considered during the design and construction of current facilities. This has caused a growth in cost and timescales during dismantling, which could have been mitigated if the challenges regarding waste had been identified earlier. The logistics plan should consider:

- The location of waste and materials treatment (within existing facilities or in a new facility on-site, sending off-site to a specialised facility, fleet or programmatic approach to optimise the location and capacity of treatment facilities);
- The decoupling of material and waste treatment from the dismantling activities;
- Ensuring that there is sufficient interim storage capacity;
- Measures to comply with the transport regulations for radioactive waste;
- Measures to comply with the treatment of waste (in compliance with regulatory framework and the waste acceptance criteria of a repository).

8.2 Radioactive material decontamination

The key driver for the decontamination of radioactive material is to reduce the hazardous inventories of materials which need to be managed and stored during the decommissioning process. [717,718] The reduction for the radiological risk and cost of storage of radioactive materials can be achieved using decontamination technologies.

Whilst the section 5.3 'Systems Decontamination (Internal)' considers less intrusive techniques for decontamination, the scope of this section considers techniques that could be applicable throughout the remit of nuclear remediation tasks.

8.2.1 Chemical

8.2.1.1 Chemical Dissolution

See Section 5.3.1

In the context of general decontamination (rather than internal surfaces), the technique can additionally, be used to immerse smaller items into the decontamination liquor, this has additional usefulness to decontaminate items that can be disassembled in the process of decommissioning activities.

8.2.1.2 Electrochemical

See Section 5.3.2

8.2.2 Mechanical / Physical

8.2.2.1 Abrasive Methods

Dry abrasive blasting is a rapid, readily available, surface cleaning and preparatory method employed in wide variety of industries. The technique involves pneumatically firing an insoluble abrasive at the surface to be decontaminated. The technique is suitable for use on a wide variety of surfaces including metals, non-metallic and masonry-based surfaces.[719]

A wide variety of media can be used as the abrasive, which defines the terminology used for each of the technique. For example, metallic media is termed shot blasting, non-metallic media is termed grit blasting, soft media is termed sponge blasting, and cryogenic blasting would typically use CO_2 pellets.

[717] IAEA, "Decontamination and Decommissioning of Nuclear Facilities," IAEA-TECDOC-511, Vienna, 1989.
[718] L. Noynaert, "13 - Decontamination Processes and Technologies in Nuclear Decommissioning Projects," in Nuclear Decommissioning; Planning, execution and International Experience, Woodhead Publishing Series in Energy, 2012, pp. 319-345
[719] E. Feltcorn, "Technology Reference Guide for Radiologically Contaminated Surfaces," US Environment Protection Agency, Washington DC, 2006.

Vibratory abrasion or vibratory polishing uses a vibrating vessel containing an abrasive media (ceramic chips or stainless-steel balls)[720] that mechanically dislodges the contamination from a surface. The process is typically relatively light touch in terms of operator involvement (i.e. the placing of contaminated objects into a vessel). However, the technique is limited to a relatively small sized items that can be contained within the vibratory decontamination vessel. Vibratory abrasion produces minimal waste as the abrasives can be re-used (in the case of metal abrasives) as the contamination can be washed from the abrasive media and vessel.

Abrasive methods are effective and easily deployed to remove tightly adhered contamination. The technique is relatively cheap and there are minimal equipment requirements. However, the technique does require intensive physical work, which is usually performed physically by a human. There are suitable alternative methods for deployment, for example remote operations. Another disadvantage to the technique is the significant amount of secondary waste and airborne contamination generated. This can be optimised to reduce secondary waste generation. Additionally, the contamination can be pushed into porous materials and access is needed to the surface as it is decontaminated directly.

A study at the Argentinian Constituyentes Atomic Centre concluded that the treatment was an effective pre-treatment process before additional chemical decontamination was used. The main parameters that impacted the effectiveness of the method were time, vibration intensity, solid media features and liquid media flux.[721]

8.2.2.2 Vacuuming

Simple vacuuming is used as a method of removing contaminated dust and loose material (wet or dry) from a range of different surfaces. It is often used as a pre-treatment before other decontamination techniques are utilised (e.g. removing powders from active gloveboxes).[722] Reported applications typically utilise 'nuclearised' industrial or domestic devices.[723] The advantages in vacuuming over more manual sweeping technologies include the reduction in time spent for the operators to decontaminate a surface.

Strippable coatings are typically polymer-based systems that are applied in liquid or gel form to a contaminated surface. The contamination is removed upon stripping of the cured coating from the surface ether manually or using a vacuum based device.[724] The coatings, before removal, may also act as contamination tie-down agent that may facilitate other decontamination techniques to occur with reduced airborne contamination being generated. Coatings (typically water-based polymers) maybe

---

720 J. Nesbitt, S. Slate and L. Fetrow, "Decontamination of high-level waste canisters," Pacific Northwest Laboratory, 1980

721 S. Fabbri and S. Ilarri, "Development of Decontamination Technology for Tubular Components. In: Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities: Final report of a coordinated research project," in Vienna, Austria, 2008.

722 IAEA, "Decontamination and Decommissioning of Nuclear Facilities," IAEA-TECDOC-511, Vienna, 1989

723 Tiger-Vac, "Tiger Vac Nuclear Safe Containment Vacuum," [Online]. Available: https://www.tiger-vac.com/nuclear-safe-containment-vacuum.

724 K. Archibald and R. Demmer, "Tests Conducted with Strippable Coatings," USDoE (INL) doi:10.2172/11008, 1999

applied manually or by remote technologies such as fogging and may perform variably depending on a number of factors including (but not limited to) surface roughness, temperature, pH, porosity etc.

The technique can be implemented for any surface with small, loose particulate contamination and the hand-held systems enable operators to remove contamination from areas which could be difficult to reach with other methods. This technique is labour intensive and time consuming.

Vacuuming can be valuable as a pre-treatment technique, for example, the concrete hot cells at Risø in Denmark were remotely vacuumed before further decontamination took place.[725]

8.2.2.3 Wet Surface Cleaning

See section 5.3.3

8.2.2.4 Mechanical Surface Removal

Grinding/shaving and scabbling are similar techniques that remove a thin layer of contaminated material from a surface.[726] Grinding typically uses carbide or diamond-based wheels or disks to remove surface layers in either hand-held devices or larger machines suitable for floor type applications. Scabblers (Figure 8.2-1) use a slightly different method for removing a surface from a material with the application of small 'piston heads' that pulverise the surface containing contamination. These devices range in size from small hand-held devices to larger machines suitable for floor type applications.

Figure 8.2-1 Decontamination of a floor using a scabbler, JPDR decommissioning project[727]

Deployment considerations are similar to those associated with the requirement for intensive physical work, which is usually performed physically by a human. There are suitable alternative methods for deployment, for example remote operations. Another disadvantage to the technique is the significant amount of secondary waste and airborne contamination generated. The techniques are widely used in other industries and offers relatively economic method for removing contamination from surfaces

[725] IAEA, "State of the Art Technology for Decontamination and Dismantling of Nuclear Facilities," Vienna, 1999.
[726] L. Noynaert, "13 - Decontamination Processes and Technologies in Nuclear Decommissioning Projects," in Nuclear Decommissioning; Planning, execution and International Experience, Woodhead Publishing Series in Energy, 2012, pp. 319-345
[727] IAEA, "State of the Art Technology for Decontamination and Dismantling of Nuclear Facilities," Vienna, 1999.

to facilitate material free release. Numerous examples of successful nuclear application have been reported.[728]

8.2.2.5 Laser Ablation (Scabbling)

Laser Ablation is a non-contact process where the top layer of a contaminated surface is removed by the application of a pulsed or continuous laser beam.[729] This results in the removal of the upper surface layers of a material, in a similar way to other ablation techniques (shot blasting), where the contamination containing material create a plasma or vapour.[730] The technique is particularly useful for removing hot-spots of contamination on various materials and offers a rapid, high precision decontamination technique that has low operational noise that produces a low volume of secondary waste. However, an extract system for the off gas is required to avoid the spread of contamination. Another flaw is the knowledge gap in species, properties and behaviour of the emissions from this technique in terms of radioactivity.[731]

A single mode pulsed laser was utilised to treat a contaminated 304 stainless steel pipe that was retrieved from a Boiling Water Reactor (BWR).[732]

8.2.2.6 Cryogenic Scabbling

Cryogenic scabbling covers two main techniques; nitrogen blasting (equivalent to high pressure water jetting with a cryogenic liquid) and CO_2 blasting (Figure 8.2-2). CO_2 blasting uses a solid form of CO_2 as a blast media usually in a stream of compressed air. The fluidised mixture of air and solid CO_2 is accelerated via a blasting unit to supersonic speeds, upon impact the CO_2 sublimates.[733] Three main processes facilitate the decontamination:

- Mechanical abrasion as per grit blasting or high-pressure water jetting;
- The cryogenic thermal shock as the cool media impacts the surface being treated;

[728] L. Noynaert, "13 - Decontamination Processes and Technologies in Nuclear Decommissioning Projects," in Nuclear Decommissioning; Planning, execution and International Experience, Woodhead Publishing Series in Energy, 2012, pp. 319-345

[729] L. Noynaert, "13 - Decontamination Processes and Technologies in Nuclear Decommissioning Projects," in Nuclear Decommissioning; Planning, execution and International Experience, Woodhead Publishing Series in Energy, 2012, pp. 319-345

[730] B. Peach, M. Petkovski, J. Blackburn and E. D.L., "An Experimental Investigation of Laser Scabbling Concrete," Construction and Building Materials, vol. 89, pp. 76-89, 2015.

[731] L. Carvalho, W. Pacquentin, M. Tabarant, A. Semerok and H. Maskrot, "Metal Decontamination by High Repetition Rate Nanosecond Fibre Laser: Application to Oxidized and Eu-Contaminated Stainless Steel," Applied Surface Science, vol. 526, no. 146654, 2020

[732] A. Kumar, T. Prakash, M. Prasad, S. Shail, R. B. Bhatt, P. G. Behere and B. J. Biswas, "Laser assisted removal of fixed radioactive contamination from metallic substrate," Nuclear Engineering and Design, vol. 320, pp. 183-186, 2017.

[733] L. Noynaert, "13 - Decontamination Processes and Technologies in Nuclear Decommissioning Projects," in Nuclear Decommissioning; Planning, execution and International Experience, Woodhead Publishing Series in Energy, 2012, pp. 319-345

- Gaseous interactions as the cryogenic media volume expands.

The main difference is that the CO_2 system is effectively a grit blasting system using CO_2 pellets as the grit, whereas nitrogen blasting is the cryogenic equivalent of high-pressure water jetting.

An academic study showed that CO_2 blasting was a faster decontamination process overall when compared to grid blasting and water jetting in a non-radiological environment. This was predominantly due to the reduced requirements for waste containment following the decontamination activities, however the actual physical decontamination process was slower in comparison.[734] Cryogenic scabbling is insensitive to surface conditions and is chemically inert. However, the use of nitrogen and CO_2 creates a potential asphyxiation (suffocation) risk for operators which requires mitigation. As with all blasting techniques the spread of contamination potential is high and adequate containment is required.[735]

Figure 8.2-2 CO2 decontamination of the turbine of the BR3, Belgium[736]

8.2.2.7 Microwave Scabbling

Microwave scabbling is a decontamination process whereby microwave radiation is used to remove the surface layer of a material. The proposed mechanism is that localised heating caused by the microwave generation generates steam and therefore pressure within the material which causes the surface of the material to fragment. Appropriate application of the microwave to the surface can be tricky, and additional hazards are generated to the workforce with the microwave radiation. However, it is applicable to non-metallic surfaces and offers good decontamination. A critical parameter being the water content of the surface having a significant impact on the success of the technique due to

---

[734] L. R. Milliman and J. W. Giancaspro, "Environmental Evaluation of Abrasive Blasting with Sand, Water and Dry Ice.," International Journal of Architecture, Engineering and Construction, vol. 1, no. 3, pp. 174-182, 2012
[735] M. Denton, "Innovative D&D Technology Utilizing High Pressure Liquid Nitrogen for Scabbling, Cutting and Decontamination," in WM2009, Phoenix, 2009
[736] L. Noynaert, "13 - Decontamination Processes and Technologies in Nuclear Decommissioning Projects," in Nuclear Decommissioning; Planning, execution and International Experience, Woodhead Publishing Series in Energy, 2012, pp. 319-345

the impact of microwave absorption. The technology has been demonstrated with engineering scale demonstrations[737] but has not been widely used commercially.

8.2.2.8 Plasma/Thermal Induced Melting

The technique generates a plasma at the surface of the material which is used to 'melt' the surface with the bulk of the contamination being carried away in an off-gas system. The technology is relatively low TRL but would offer the decontamination of surfaces with the majority if waste being generated as off-gas particulate. The abatement of the offgas being one of the areas of research interest for this technology. Research is active in this area with many groups working in this area although no reported commercial applications.[738]

8.2.2.9 Microbial Degradation

Microbial Degradation for decontamination has been considered for targeted applications where specific microorganisms, selected due to defined grow conditions, are chosen that will produce acid or other metabolites that will degrade the contaminated surface of choice. Control of microorganism growth therefore correlates the depth of surface decontamination required. Indeed, BNFL and Idaho National Laboratory (INL)[739] have considered the use of sulphur oxidising bacteria for this application on a cement substrate.

8.2.3 Experiences/Case Studies

At Sellafield, United Kingdom, the site is undergoing Post Operational Clean Out (POCO) operations which aims to reduce the radiological inventory with minimal breaks in containment, the next stage of decommissioning enables the use of more intrusive techniques to remove contamination. This conceptually uses various techniques to remove the surface layers of contaminated materials.

---

[737] T. White, T. Bigelsow, C. Schaich and D. Foster, "Mobile System for Microwave Removal of Concrete Surfaces". Patent 5635143, 1995

[738] K. Dickerson, M. Wilson-Nichols and M. Morris, "Contaminated Concrete: Occurance and Emerging Technologies for DOE Decontamination," DOE/ORO/2034, Knoxville, 1995.

[739] J. Benson and H. Eccles, "A Method of Decontaminating a Cementitious or a Metallic Surface". JP, KR, US, EU Patent PCT/GB95/01772, 28 07 1994.

8.3 Radioactive material treatment processes

One of the key aspects of the radioactive waste management cycle is the development of an approach which includes all possible waste streams. This section is designed to give an overview of the current technology that can be deployed during the treatment of a range of wastes in different decommissioning operations. The treatment processes detailed within this section allow the application of the waste hierarchy, by facilitating the reuse, recycle or conditioning of the waste material for the final storage.

8.3.1 Treatment of Metallic waste

Metallic wastes are produced from different parts of the nuclear fuel cycle. They include:

- Fuel Cladding
- Equipment, such as process items e.g. valves, piping, thermocouples etc
- Bulk items including internal cladding from cells, larger scale process vessels, heat exchangers,
- Items from decommissioning (for example glove boxes, vessels, structural metal work)
- Spent Metal Fuel
- Transport flasks

This section will detail the approaches taken for the conditioning and disposal of metallic wastes. Operations, such as the decontamination of metallic wastes, are covered in Sections 5.3 and 8.2.

8.3.1.1 Fuel Cladding

For recently used spent fuel, the approach in the United Kingdom is to remove the cladding from the spent fuel, either by physical removal (for metal fuels, in the case of Magnox fuel rods, the natural uranium fuel is clad in magnesium with a small amount of aluminium present) or chemical dissolution (for oxide fuels, where the cladding is made from either stainless steel or Zircalloy). The waste from the cladding is then encapsulated in blended cement to provide a solid package which can then be kept in an above ground Engineered Product Store (EPS) prior to long-term disposal in a Geological Disposal Facility (GDF), which it yet to become available. [740]

An example of fuel cladding encapsulated in cement can be seen in Figure 8.3-1.

---

[740] C. Bayliss and K. Langley , "Nuclear Decommissioning, Waste Management and Environmental Site Remediation", Oxford, Elsevier, 2003.

Figure 8.3-1 Magnox metal cladding (L) and stainless-steel cladding from oxide fuel (R) encapsulated in cement[741]

8.3.1.2 Legacy Wastes

For legacy materials (such as untreated fuel cladding that has been subject to extended interim storage either in air or underwater), a different approach is being taken whereby the partially corroded material is retrieved and transferred to stainless steel containers, equipped with concrete bunds.[742] These containers, along with the raw waste, will be kept in an Engineered Store for the foreseeable future. It is planned that the waste will undergo a finalisation step prior to consignment to a GDF. This is to ensure that it is compliant with the requirements for transport, disposal and the GDF safety case.

8.3.1.3 Plutonium Contaminated Materials (PCM)

Metals that originate from items used within the process plant which are contaminated with Transuranic (TRU) elements, are known in the UK as Plutonium Contaminated Material (PCM). This is a broad stream, which contains materials with a wide variety of properties such as PVC, polythene, plant and equipment, protective clothing and other items from plutonium related operations. The treatment approach currently taken with PCM is for the waste to be consigned into approximately 200 Litre drums (Figure 8.3-2), which are then super-compacted to form pucks.[743] These pucks are then placed within a cage inside a 560 Litre stainless steel drum, which is then filled with a cementitious annulus grout.

[741] C. Bayliss and K. Langley , "Nuclear Decommissioning, Waste Management and Environmental Site Remediation", Oxford, Elsevier, 2003.
[742] Sellafield Ltd , "The 2016/17 Technology Development and Delivery Summary," Sellafield Ltd, Seascale, Cumbria, UK , 2017
[743] C. Bayliss and K. Langley , "Nuclear Decommissioning, Waste Management and Environmental Site Remediation", Oxford, Elsevier, 2003.

Figure 8.3-2 An example of a PCM drum containing compacts within an annulus filled drum[744]

8.3.1.4 Bulk Metals

Bulk metals can be melted to achieve decontamination; an example of this is the Centraco facility operated by EDF-Cyclife.[745] The facility is capable of processing large items up to 19 metres long with a maximum weight of 200 tonnes, such as heat exchangers or spent fuel pond storage, without the need for pre-treatment by the waste consignor. The melted waste is then poured into ingots, enabling a volume reduction factor of 6-10 and allowing physical/chemical/radiological characterisation of the waste to be performed. Less contaminated material is melted and potentially released for use, while more contaminated materials will be returned for disposal by the waste consignor.

8.3.2 Concrete

The treatment and subsequent disposal of contaminated concrete can be a significant issue in plant decommissioning due to the predicted large volumes of material that will arise from demolition operations. The proportion of contaminated material arising from a building depends on the use on that building, and in many cases, contamination is limited.[746]

Several techniques have been developed for concrete decontamination, see Sections 5.3, 6.7 and 8.2.[747]

---

[744] C. Bayliss and K. Langley , "Nuclear Decommissioning, Waste Management and Environmental Site Remediation", Oxford, Elsevier, 2003.

[745] EDF-Cyclife, EDF-Cyclife, July 2020. [Online]. Available: https://www.cyclife-edf.com/en/cyclife/governance/cyclife-france/our-solutions/melting.

[746] Organisation for Economic Cooperation and Development Nuclear Energy Agency (OECD NEA), "Decontamination and Dismantling of Radioactive Concrete Structures," OECD NEA, Paris, 2011.

[747] Organisation for Economic Cooperation and Development Nuclear Energy Agency (OECD NEA), "Decontamination and Dismantling of Radioactive Concrete Structures," OECD NEA, Paris, 2011.

8.3.3 Aqueous liquids[748]

The general objective of the treatment of aqueous wastes is to split the effluent into two fractions (1) a small volume of concentrate that contains the majority of the radionuclides and (2) a large volume where the radionuclide concentration is sufficiently low to permit its discharge to the environment. The Decontamination Factor (DF) is defined as the ratio of the initial concentration of radionuclides in the waste A_o to the final concentration in the largest waste stream A_f.

$DF = A_o/A_f$

The variety of chemical and radiochemical compositions of aqueous wastes has led to several different treatment technologies being developed and deployed, including:

- Evaporation
- Chemical precipitation
- Sorption and ion exchange
- Microfiltration
- Reverse osmosis
- Electrodialysis
- Coagulation
- Ultrafiltration

In practice several methods may be used in combination, in order to achieve the required DF's. Examples of some of the technologies currently deployed are detailed in the section below.

8.3.3.1 Evaporation

Evaporation is a method for the treatment of liquid radioactive wastes that provides high DF's and a good concentration of salts and bottom residues. Clean(er) water is removed as vapour, resulting in the concentration of non-volatile salts and radionuclides, which can then be stored or sent for further treatment.

Evaporation of liquid radioactive waste with low salt content (1-5 g/l) is generally carried out as a two-stage process, with decontamination carried out in the first stage and concentration in the second stage. For high salt content wastes (up to 400 g/l), the evaporation is generally carried out as a single-stage process. An example of this process would be the Highly Active Liquor Evaporation and Storage (HALES) Process[749], which is used for volume reduction of liquors from reprocessing at the Sellafield site in the UK, prior to subsequent vitrification of the waste.

[748] M. Ojovan, W. Lee and S. Kalmykov, "Chapter 16 - Treatment of Radioactive Wastes", An Introduction to Nuclear Waste Immobilisation, 3rd Elsevier, 2019, 231-269

[749] C. Nixon , "The Application of Research and Technology on the Highly Active Liquor Storage and Treatment Facilities at Sellafield," in Atalante 2004, Nimes (France) , 2004

8.3.3.2 Chemical Precipitation

With Chemical Precipitation (also known as reagent coagulation) processes, impurities are precipitated from an aqueous waste stream through a change in pH, redox potential or co-precipitation using precipitants such as ferrous or aluminium sulphate. A special case of coagulation is reagent oxidation in which an oxidising agent (such as potassium permanganate or dichromate) is added to destroy organic components or to change the oxidation state of multivalent ions.

A typical chemical precipitation process involves four stages:

1. Addition of reagents to the feed stream, adjustment of the pH to form the precipitate
2. Flocculation (the process by which fine particulates are caused to clump together into a floc)
3. Sedimentation
4. Solid-liquid separation

Chemical precipitation typically forms sludge and therefore always needs a physical method to separate the sludge from the supernatant liquor. The radionuclides from the feed stream are concentrated in the smaller volume of sludge material, which is separated with the larger volume of supernatant, potentially suitable for direct discharge because of its very low residual activity.

This type of chemical precipitation operation has been used at the Sellafield site in the UK for the treatment of liquid effluents in the Enhanced Actinide Removal Plant (EARP), shown in Figure 8.3-3.[750] In a similar way to the historic processes (that used sedimentation), EARP removes actinides using flocculation. The acidic effluent is neutralising with an alkali, in this case sodium hydroxide. The overall effectiveness of the activity removal process is enhanced by the addition of reagents that remove soluble radioactive species.

Figure 8.3-3 Enhanced Actinide Removal Plant (EARP) facility[751]

Rather than using a settling process, EARP uses crossflow ultrafiltration to de-water the floc. The resulting permeate has very low residual activity levels and is suitable for sea discharge, following confirmatory sampling and analysis. The concentrated radioactive floc is transferred to a separate

---

[750] E. Winstanley, "Radionuclide Uptake During Iron (Oxyhydr)oxide Formation: Application to the Enhanced Actinide Removal Plant (EARP) Process", The University of Manchester, 2018

[751] E. Winstanley, "Radionuclide Uptake During Iron (Oxyhydr)oxide Formation: Application to the Enhanced Actinide Removal Plant (EARP) Process", The University of Manchester, 2018

downstream plant, the Waste Packaging and Encapsulation Plant (WPEP). Within WPEP, the concentrated floc is mixed with cement powders in an encapsulation process that produces a stable cemented product within a stainless-steel container suitable for long-term safe storage.[752]

8.3.3.3 Sorption and Ion Exchange

Sorption is a water purification process in which an aqueous waste stream is passed through a layer or bed of materials such as activated charcoals, synthetic ion exchange resins or natural zeolites. Radionuclides in the feed stream are removed through the process of sorption as a result of their physical and chemical interaction with the sorption/ion exchange media. Ion exchange is a special case in which the ions in solution exchange with those on the surface of the media and are removed from the solution. A mixed bed of ion exchange resins is employed where there is a requirement to remove both cations and anions from the solution.

An example of an industrial scale process using the natural zeolite is the Site Ion Exchange Plant (SIXEP) at the Sellafield site. SIXEP began operation in 1985 and is part of a suite of plants utilised clinoptilolite to decrease discharges into the environment. The plant was designed specifically to reduce levels of soluble caesium and strontium species within liquid effluents.

The plant treats pond water from various fuel storage ponds across the Sellafield site. The process involves feeding the pond water through sand filters to remove any suspended solids; it is then neutralised prior to ion exchange. After neutralisation, the filtered effluent is fed through an ion exchange medium to remove radioactive caesium and strontium. The treated effluent is normally discharged continuously to the sea via a break pressure tank. A proportional sampler continuously collects a liquor sample for retrospective accountancy. The treated effluent is continuously monitored by an in-line gamma monitor that will automatically stop the discharge pump and close its discharge valve on the detection of higher than normal activity.[753]

The spent beds of combined sand and clinoptilolite are currently stored in Bulk Storage Tanks (BSTs), with work underway to assess the most suitable treatment method to produce waste forms suitable to allow the waste to be consigned to a geological disposal facility.

8.3.3.4 Physical conditioning/separation

Physical conditioning/separation means that waste is separated into two or more components. For solid waste this can be done with mechanical techniques such as shredding and sorting, or by phase separation for liquids.[754]

---

[752] "The Engineer," [Online]. Available: http:/www.engineerlive.com/power-engineer/nuclear-power/14432/treating-a-50yearold-legacy-of-radioactive-sludge-waste.thtml.

[753] European Union, "Verification under the terms of Article 35 of the Euratom Treaty," European Union , Brussels , 2004

[754] International Atomic Energy Agency, "TRS- 427 Predisposal Management of Organic Radioactive Waste," IAEA, Vienna, 2004

For example, the Segregated Effluent Treatment Plant (SETP) is designed to handle low risk, low active acidic and alkaline effluents arising from Magnox reprocessing operations, in addition to other feeds from across the Sellafield site. The acidic effluents have their pH increased by the addition of sodium hydroxide prior to mixing with the alkaline stream. The combined effluent is filtered to remove debris prior to transfer to one of three SETP sea tanks, where it is proportionally sampled and sentenced prior to discharge to sea.

8.3.3.5 Mobile Units

In addition to fixed plants that are built as part of the infrastructure for an operational site, where typically high-volume waste streams are produced from chemical processes and storage ponds, another option that exists is the use of mobile units.

The mobile units are characterised by flexibility that allow them to treat a wider range of wastes, which may be more variable in composition.[755] Such systems are generally designed for application where the volumes of liquid waste are relatively low (typically <500m^3/year and salinity <3g/l).

Figure 8.3-4 Aqua Express modular mobile aqueous waste treatment facility[756]

An example of this is the Aqua-Express unit built in Russia (Figure 8.3-4), which consists of three autonomous treatment modules and includes an initial filter container with a selective ferrocyanide sorbent, filtration and ultrafiltration modules.[757]

The ferrocyanide sorbent consists of either nickel ferrocyanide of copper ferrocyanide deposited on silica gel and is designed to selectively extract caesium from the waste. However, the modular nature of the mobile plants means that alternative sorbents, with properties which are tailored to the specific radionuclides present in the liquid waste can also be used.

After the removal of the primary radionuclides using the sorbent, the remaining liquid waste is routed to an ultrafiltration module, where the concentration of the feed stream is performed. Part of the

[755] International Atomic Energy Agency, “Mobile Processing Systems for Radioactive Waste Management,” IAEA, Vienna, 2014
[756] M. Ojovan, W. Lee and S. Kalmykov, “Chapter 16 - Treatment of Radioactive Wastes”, An Introduction to Nuclear Waste Immobilisation, 3rd Elsevier, 2019, 231-269
[757] W. Lee, “Chapter 16 - Treatment of Radioactive Wastes,” in An Introduction to Nuclear Waste Immobilisation, 3rd ed., Elsevier, 2019, pp. 231-269.

permeant feed stream is then passed through an ultrafiltration membrane to produce a final liquid effluent from which suspended solids, polymeric materials and colloids have been removed.

The clean-up of the cooling water at Fukushima also involves many of these techniques. In this stream, cooling water which was in direct contact with the spent fuel rods, has been mixed with sea water, leading to increased corrosion and further non-active components in the feed stream. As a result, the mobile unit for the clean-up of the cooling water is a multistep system consisting of:

1. An oil separator;
2. A caesium absorption step developed by Kurion, with three subcomponents. A pre-treatment column packed with a modified zeolite for removing remaining oil and technetium, four parallel columns of the sorbent herschelite for Cs removal and a column packed with Ag-impregnated herschelite to remove iodine. In combination, these filters remove Cs and strontium from the liquid waste.
3. A system for the removal of residual Cs, using precipitation and coagulation, was developed by Areva.
4. A second line, known as SARRY (Simplified Active Water and Recovery System), which utilises crystalline silicotitanates for the removal of caesium.

8.3.4 Non-Aqueous Liquids[758]

The objectives of non-aqueous liquid radioactive waste treatment are;

- Conversion to a solid form
- Conversion to an inorganic form to enable conditioning
- Volume reduction
- Decontamination for reuse
- Conversion to an organic form which is suitable for cementation

The main treatment methods for non- aqueous liquid wastes are:[759]

- Incineration
- Emulsification to facilitate encapsulation in cementitious materials/ Direct immobilisation
- Absorption onto a matrix
- Distillation
- Wet oxidation
- Alkaline Hydrolysis

[758] M. Ojovan, W. Lee and S. Kalmykov, “Chapter 16 - Treatment of Radioactive Wastes”, An Introduction to Nuclear Waste Immobilisation, 3rd Elsevier, 2019, 231-269

[759] International Atomic Energy Agency, “TRS- 427 Predisposal Management of Organic Radioactive Waste,” IAEA, Vienna, 2004

8.3.4.1 Incineration

This is an attractive option for liquid organic wastes because many are readily combustible and high-Volume Reduction Factors (VRF) can be achieved. The primary technical difficulties relating to the requirement to achieve complete combustion of the waste and the monitoring and maintenance of the stack emissions with discharge limits, which have been agreed with the regulatory agencies for both radionuclides and organic waste components.

Figure 8.3-5 : Inside view of a radioactive waste incinerator. Fire grates seen downward from the waste loading unit (L) and post combustion chamber filled with silicon carbide (R)[760]

8.3.4.2 Emulsification to facilitate encapsulation in cementitious materials/ Direct immobilisation

The direct immobilisation technique describes the process where raw waste is directly mixed with a binding material. The original product does not change but is wholly embedded and thus isolated from the environment. The advantage of direct immobilisation is that a disposable product is obtained at the production site in a single step. In Germany, the Netherlands, Sweden, the USA and various other countries, ion exchangers and sludges are directly immobilised in cement.[761]

8.3.4.3 Absorption

A simple approach which applies the use of absorbents, to incorporate all the liquid organic material. Typical absorbents include:

- Natural fibres (sawdust, cotton);
- Synthetic fibre (polypropylene);
- Vermiculite (mica);
- Clays;
- Diatomaceous earth.

[760] M. Ojovan, W. Lee and S. Kalmykov, "Chapter 16 - Treatment of Radioactive Wastes", An Introduction to Nuclear Waste Immobilisation, 3rd Elsevier, 2019, 231-269
[761] International Atomic Energy Agency, "TRS- 427 Predisposal Management of Organic Radioactive Waste," IAEA, Vienna, 2004

The absorption efficiency of the different absorbents can vary by a factor of 2-3 and waste volume increase can be up to 300%.

8.3.4.4 Distillation

The distillation technique has two steps, evaporation and condensation. The technique is very useful if the distillate can be used again, so it is mainly used for specific liquids such as TPB. A disadvantage is that volatile radionuclides such as tritium are not removed from the distillate.[762]

8.3.4.5 Wet oxidation

The wet oxidation technique uses soluble salts of redox sensitive elements with hydrogen peroxide or air/oxygen to affect the chain reaction oxidation of organic materials, producing carbon dioxide, water and inorganic salts. The technique uses degradable oxidising agents (e.g. H_2O_2) and is suitable for low concentration water miscible organic feeds. The technique frequently relies on soluble heavy metal catalysts and can result in incomplete oxidation, leaving alcohols.[763]

8.3.4.6 Alkaline Hydrolysis

Solvent Treatment Plant (STP) at Sellafield destroys the solvents currently stored on-site, producing an aqueous residue containing the bulk of the radioactivity. This includes solvents such as tri-butyl phosphate and odourless kerosene, which are used in reprocessing operations and gradually lost by dissolution, or entrainment into monitoring tanks.[764] This is then sent to EARP for further treatment. STP commenced active commissioning in 2000.

8.3.5 Organic Materials

Within several countries, radioactive waste producers and waste management organisations are faced with low- and intermediate-level radioactive organic waste streams for which the current conditioning methods generate waste forms whose safe long-term storage and disposal is difficult to achieve and / or demonstrate, because they are considered not sufficiently stable and / or too highly reactive in the alkaline conditions expected to prevail in many final repositories. Among these waste streams are ion exchange resins (IER) and conditioned organic waste forms like polymerised waste and bituminised waste. There is a work package in PREDIS dedicated to the conditioning and behaviour (in disposal) of liquid and solid organic waste.[765]

---

[762] International Atomic Energy Agency, “TRS- 427 Predisposal Management of Organic Radioactive Waste,” IAEA, Vienna, 2004

[763] International Atomic Energy Agency, “TRS- 427 Predisposal Management of Organic Radioactive Waste,” IAEA, Vienna, 2004

[764] European Union, “Verification under the terms of Article 35 of the Euratom Treaty,” European Union , Brussels , 2004.

[765]PREDIS, PREDIS, [On Line]. Available: https://predis-h2020.eu/predis-project/. [

8.3.5.1 Liquid

A wide variety of organic liquids are used in the nuclear industry. These can be broadly categorised as lubricants, solvents, process fluids and diluents, and decontaminants. [766]

Liquid organic wastes these may be treated using the same techniques as the non-aqueous wastes noted in Section 8.3.

8.3.5.2 Solid

The solid organic materials most widely used in the nuclear industry are plastics, rubber and cellulose (covering paper, wood and natural fibres such as cotton). Less common solid organic materials include ion exchange resins. [767]

The main treatment methods for solid organic wastes are:

- Supercompaction and encapsulation;
- Incineration and pyrolysis;
- Direct encapsulation.

8.3.5.2.1 Supercompaction

This is a well-established technique in which solid waste in containers is subjected to compression in order to reduce the overall volume occupied. A range of different types of compactor are available, with the forces used between approximately 10 and 50t. The volume reduction factors achieved are typically in the range of 2-5. [768]

Figure 8.3-6 : Supercompacting unit for radioactive waste drums, Russia[769]

[766] PREDIS, PREDIS, [On Line]. Available: https://predis-h2020.eu/predis-project.
[767] PREDIS, PREDIS, [On Line]. Available: https://predis-h2020.eu/predis-project/..
[768] M. Ojovan and W. Lee, An Introduction to Nuclear Waste Immobilisation, Elsevier, 2014
[769] M. Ojovan and W. Lee, An Introduction to Nuclear Waste Immobilisation, Elsevier, 2014

Higher volume reduction factors (up to 100 depending on the types of waste being treated) can be achieved using supercompactors, which are capable of applying forces of >1200 -1500t.[770]

Example of the use of compaction include the treatment of PCM at the Sellafield Waste Treatment Complex (WTC)[771], by ANDRA[772] and the for the treatment of certain Low-Level Wastes prior to infill grouting.

8.3.6 Very Low-Level Waste (VLLW)

VLLW is the lowest sub-category of Low-Level Waste (LLW). In general, VLLW contains some residual radioactivity, but it does not reach high specific activity limits. Sites that produce VLLW can dispose of the waste with regular household waste or industrial waste at permitted landfill facilities. The majority of VLLW produced at a nuclear site will come from building rubble, soil and steel items. These arise from the dismantling and demolition of nuclear reactors and facilities.[773,774]

8.3.7 Low Level Waste (LLW)

LLW contains relatively low levels of radioactivity, not exceeding 4 gigabecquerel (GBq) per tonne of alpha activity, or 12 GBq per tonne of beta/gamma activity. Most LLW comes from the operation and decommissioning of nuclear facilities. The waste includes items such as scrap metal, paper and plastics. Hospitals and universities also produce a small amount of LLW. About 94% of all radioactive wastes (by volume) are in the LLW category.[775] Several techniques are available for the treatment of LLW including:[776]

- The treatment of metal wastes (Sections 5.3, 8.2 and 8.3) - These techniques are all designed to reduce the volume of contaminated waste by removing the higher levels of contamination, to produce a smaller volume of higher activity waste, with a larger volume of lower activity waste.
- Smelting
- Incineration (Combustibles)
- Supercompaction
- Volume Reduction, for example
    - Hot and cold cutting

---

[770] M. Ojovan and W. Lee, An Introduction to Nuclear Waste Immobilisation, Elsevier, 2014
[771] C. Bayliss and K. Langley , Nuclear Decommissioning, Waste Management and Environmental Site Remediation, Oxford: Elsevier, 2003.
[772] ANDRA, "Synthesis Report - National Inventory of Radioactive Materials and Waste," ANDRA, 2018.
[773] Nuclear Decommissioning Authority, "UK Radioactive Waste Inventory," Nuclear Decommissioning Authority, 20 July 2020. [Online]. Available: https://ukinventory.nda.gov.uk/about-radioactive-waste/what-is-radioactivity/what-are-the-main-waste-categories/.
[774] International Atomic Energy Agency, "Management of very low level radioactive waste in Europe - application of clearance (and the alternatives)," sat science GMBH, 2010. [Online]. Available: https://nucleus.iaea.org/sites/connect/IDNpublic/R2D2/Workshop%2008/management-of-very-low-level-waste.pdf.
[775] Nuclear Decommissioning Authority, "UK Radioactive Waste Inventory," Nuclear Decommissioning Authority, 20 July 2020. [Online]. Available: https://ukinventory.nda.gov.uk/about-radioactive-waste/what-is-radioactivity/what-are-the-main-waste-categories/.
[776] Low Level Waste Repository, "Waste Management Services," LLWR, 2020. [Online].

- Shredding

8.3.8 Intermediate level wastes (ILW)

ILW exceeds the upper activity boundaries for LLW but does not require provisions for heat dissipation during its storage or disposal. The current concept for disposal of high activity radioactive waste in the United Kingdom is a GDF, which will store all ILW and HLW arising from nuclear facilities.[777] A similar approach is adopted in other countries as well.

The current strategy for treatment of ILW is encapsulation by an ordinary Portland cement matrix, combined with super-compaction where appropriate. In contrast to the encapsulation processes, thermal treatment technologies offer the potential to reduce conditioned and packaged waste volumes. Thermal treatment allows the destruction of combustible materials, removal of entrained water, and minimisation of void space which can reduce the conditioned waste volume by a factor of between 2 to 100, compared to the volume of unconditioned waste.[778] Therefore, thermal treatment of ILW would dictate considerable financial, environmental, safety, and security benefits that arise from lowering the volume and number of waste packages requiring interim storage, transport, and placement in a repository.

For further information on encapsulation and thermal treatment see Section 8.4.

[777] [Online]. https://www.neimagazine.com/features/featurethermal-treatment-of-ilw/ .

[778] IAEA, "Application of thermal technologies for processing of radioactive waste", TECDOC-1527, IAEA, December 2006.

8.4 Radioactive waste conditioning

There is some commonality between waste treatment and conditioning. In this document the definitions [779] used are:

- Treatment - can comprise thermal/chemical/physical processes and results in the change of the waste characteristics to facilitate subsequent management steps, such as recycling or disposal.
- Conditioning/Immobilisation - changes the form of the waste so the resulting product can be safely handled, transported, stored and disposed.

From this some of the techniques identified in Section 8.3 are also applicable to radioactive waste conditioning. In this section, technologies which have been used and which are being developed for the treatment of Intermediate Level Waste (ILW) are described.

8.4.1 Cement Based Encapsulation

Cementation is one of the more commonly used technologies for the treatment of ILW. Examples of operating processes in the United Kingdom include: [780, 781]

- Magnox swarf and metallic hulls from reprocessing operations through infilling of waste packages with premixed grouts;
- The encapsulation of slurry wastes such as barium carbonate (from the abatement of carbon-14 containing aerial effluents and ferric flocs from aqueous wastes) through in-drum mixing with the addition of dry cement powders;
- Annulus of supercompacted pucks containing Plutonium Contaminated Materials (PCM).

These operations all use 500 litre capacity stainless steel drums which have been designed to be compatible with storage and transport equipment for onward transit to a GDF, when such a site becomes available in the United Kingdom.

Similar approaches are used in other countries for the treatment of wastes of this type, such as at the Aube facility in France by ANDRA,[782] with similar approaches used across the international nuclear industry.[783]

---

[779] Nuclear Decommissioning Authority, "Integrated Waste Management - Radioactive Waste Strategy", Nuclear Decommissioning Authority, Cleator Moor, Cumbria, 2019.

[780] C. Bayliss et K. Langley , "Nuclear Decommissioning, Waste Management and Environmental Site Remediation", Oxford: Elsevier, 2003.

[781] E. Butcher, R. Caldwell, I. Godfrey, M. Hayes et E. Miller, "Development and Implementation of Technology for the Treatment and Encapsulation of Operational Intermediate Level Waste", chez ICEM 03 , Oxford, 2003.

[782] ANDRA, "The Short-Lived, Low and Intermediate Level Waste (LILW-SL)", ANDRA, [On Line]. Available: https://international.andra.fr/operational-facilities/aube-waste-disposal-facility-csa/short-lived-low-and-intermediate-level-waste-lilw-sl.

[783] International Atomic Energy Authority, "The Behaviours of Cementitious Materials in Long Term Storage and Disposal of Radioactive Waste - Results of a Coordinated Research Project", IAEA, Vienna, 2013.

The EU funded PREDIS project, is currently testing and evaluating innovations in cemented waste handling and pre-disposal storage. The project intends to identify options for post treatment of packages and potential approaches to improve package design, construction and maintenance.[784]

Further details of the work performed to date is provided on the PREDIS website.[785]

8.4.2 Polymer Based Encapsulation

The use of polymer encapsulation for the conditioning of nuclear wastes has also been applied in a number of instances, including for organic ion exchange resins from the Trawsfynydd nuclear power plant.[786] This was a specific waste treatment process that was developed during the 1980's and which has operated on a campaign basis to treat wastes produced by the power plant. Processes using polymers for the treatment of ion exchange resins have also been developed and operated in other European countries.[787] In addition, work has been performed to assess the use of polymers for the treatment of wastes such as radium contaminated components,[788] which are subsequently annulus grouted in a cemented waste package.

8.4.1 Geopolymer Immobilisation

Geopolymers are made by adding aluminosilicates to concentrated alkali solutions for dissolution and subsequent polymerisation to form a solid to take place. They are amorphous to semicrystalline three-dimensional aluminosilicate networks. Their physical handling is similar to that of Portland cement, hence they have been considered for low and intermediate level (ILW) immobilisation.[789] For such applications, a correct comprehension of the binder structure, its macroscopic properties, its interactions with the waste and the physico-chemical phenomena occurring in the wasteform is needed to be able to judge of the soundness and viability of the material.[790]

There is a work package in PREDIS dedicated to researching the behaviour of geopolymers to stabilise effluents, organic wastes and residual wastes from thermal treatment.[791]

---

[784] https://predis-h2020.eu/predis-project/

[785] https://predis-h2020.eu/predis-project/

[786] Nuclear Decommissioning Authority and Radioactive Waste Management, "Geological Disposal: Guidance on the use of organic polymers for the packaging of low heat generating wastes", WPS 901/02, NDA RWM, Cleator Moor, 2015.

[787] International Atomic Energy Agency , "Technical Report Series 254 Treatment of Spent Ion Exchange Resins for Storage and Disposal", IAEA, Vienna, 1985.

[788] A. Green, Polymer Encapsulation of Nuclear Waste: Alternatives to Grout, Royal Society of Chemistry, 2 June 2009. [On Line]. Available: http://www.rsc.org/images/AndrewGreen_tcm18-156622.pdf.

[789] D. Perrera, M. Blackford, C. Dickson, E. Vance, A. Aly, R. Trautman, "Geopolymers for radioactive waste immobilisation made from New Zealand fly ash", The fifth conference on nuclear science and engineering in Australia, 2003. Conference handbook

[790] V. Cantarel, T. Motooka, I.Yamagishi, Geopolymers and Their Potential Applications Geopolymers and Their Potential Applications, JAEA Review, June 2017, DOI:10.11484/jaea-review-2017-014

[791] https://predis-h2020.eu/predis-project/.

8.4.2 Thermal Treatment

A number of thermal treatment technologies are being developed, the EU funded Horizon 2020 THERAMIN project is designed to give a EU community wide strategic review and assessment of the value thermal technologies applicable to a broad range of waste streams (ion exchange media, soft operational wastes, sludge, organics and liquids).[792]

Figure 8.4-1 : GeoMelt Rig, National Nuclear Laboratory, UK[793]

Technologies assessed include:

- SHIVA in Can melting technology for incineration-vitrification of organic and inorganic ion exchange resins;
- Thermal gasification;
- GeoMelt In Container Vitrification (Figure 8.4-1);
- Hot Isostatic Pressing (HIP);
- VICHR Chrompik II treatment technology.

Further details of the work performed to date is provided on the THERAMIN website[794] . The Theramin project hosted an international conference on Thermal Treatment of Radioactive Waste. Papers prepared for the conference have been published in a special issue of the "IoP Conference Proceedings - Materials Science"[795]

[792] http://www.theramin-h2020.eu/downloads.htm.
[793] http://www.theramin-h2020.eu/downloads.htm.
[794] http://www.theramin-h2020.eu/downloads.htm.
[795] journal (open access): https://iopscience.iop.org/issue/1757-899X/818/1 .

8.5 Radioactive waste packaging and logistics

Packaging and logistics are related to two major aspects of radioactive waste management: [796] storage and transport. Consequently, the focus in this section is centred around the availability of suitable containers for these applications.

Waste storage has different requirements[797] depending on the time frame: interim/long-term storage or final disposal.

The transport of radioactive material [798, 799, 800, 801] is necessary for moving the waste location where it is generated or treated to the storage or repository site, but it is also required if the material is needed to be transported to an off-site facility for treatment and/or conditioning processes.

Transport may involve more than one country, meaning international transfers are required. This means that the transport container must fulfil all regulatory requirements of the departure, transit and destination countries. Therefore, standardisation of regulation at either an international or European level would optimise this process.

To improve the use of containers and minimise the effort required to obtain the proper certification and licensing of containers. A growing area of research interest has in the design of multipurpose containers: dual-purpose (transport and interim storage) or triple-purpose (transport, storage and disposal).

8.5.1 Description of technologies

The requirements for a storage and/or transport container depends on the type of waste and its final use. For these purposes, we can classify the waste into the following main categories:

- Solid unconditioned LLW and VLLW
- Solid conditioned LLW and VLLW
- Solid ILW
- Irradiated fuel

[796] IAEA, "Handling and Processing of Radioactive Waste from Nuclear Applications", IAEA Technical Reports Series, No. 402, Vienna (2001).
[797] IAEA, "Storage of Radioactive Waste", IAEA Safety Standards Series, Safety Guide No. WS-G-6.1, Vienna (2006).
[798] IAEA, "Regulations for the Safe Transport of Radioactive Material - 2018 Edition", IAEA Safety Standards Series, Specific Safety Requirements No. SSR-6 Revision 1, Vienna (2018).
[799] IAEA, "Schedules of Provisions of the IAEA Regulations for the Safe Transport of Radioactive Material", IAEA Safety Standards Series, Specific Safety Guide No. SSG-33, Vienna (2012).
[800] IAEA, "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material", IAEA Safety Standards Series, Specific Safety Guide No. SSG-26, Vienna (2014).
[801] Transport Container Standardisation Committee, "Transport of Radioactive Material - Code of Practice: Design of Transport Packaging for Radioactive Material", Transport Container Standardisation Committee, TCSC 1042, (2002).

- HLW
- Liquid radioactive waste

As general features, any container should at minimum fulfil the following criteria:

- All packaging should provide at least two independent barriers designed to contain radioactive materials throughout handling, transport and storage/disposal operations (e.g. waste immobilisation and containerisation);
- The container should be compatible, with conventional handling equipment (i.e. geometry, volume and weight).
- The container should have sufficient structural strength to withstand stacking, dropping, penetration tests and accident scenarios (for example, fire testing). The addition of ancillary equipment such as overpacks and shock absorbers, to meet the transportation criteria, is an advantage as it removes the need for the material to be repackaged for storage;
- At the time of handling, transport and storage/disposal operations and emplacement in the repository the container should be watertight

The existing technologies for the different types of applications are described here and the gaps for improvement by R&D are identified in Table 8.5-1.

8.5.2 Unconditioned LLW and VLLW

The most common container for the storage of unconditioned LLW and VLLW is the carbon steel drum, with versions of different capacity. The 220-litre drum shown in Figure 8.5-1 is used most frequently. The drums can be licensed as type-A containers, so they can also be used for transport. An over-pack might be necessary depending on the specific activity and dose rate at the surface.

Figure 8.5-1 220-litre drums[802]

[802] Osmanlioglu, Ahmet Erdal (2012). "In-Situ Chemical Precipitation of Radioactive Liquid Waste", 12492. WM2012: Waste Management 2012 conference on improving the future in waste management, United States

8.5.3 Conditioned LLW and VLLW

Currently, grout is the most commonly used matrix for conditioning LLW and VLLW to make the waste suitable for final disposal. Waste is normally immobilised through the addition of grout generally within a rectangular a steel container (heterogeneous conditioning), see Figure 8.5-2. Liquid radioactive waste can be immobilised by mixing cementitious materials with the water in the waste to produce the final grouted waste form (homogeneous conditioning).

The grouting matrix must fulfil a criterion, including compressive strength, long-term stability and a capability to fill voids.

Figure 8.5-2 Container for conditioned LLW/VLLW

8.5.4 ILW

An example of a waste package that has been developed for the packaging of decommissioning wastes is the $6m^3$ concrete box used for reactor components from the Windscale Advanced Gas Cooled Reactor (WAGR). Developed in the 1980's and patented in 1985, the container is available in two variants, a standard density concrete version and another which uses high density concrete, designed for the packaging of higher activity wastes. [803]

In comparison to stainless steel packages, the $6m^3$ concrete box is self-shielding, allowing it to be transported using conventional infrastructure such as high capacity forklift trucks and stored in unshielded waste storage buildings.

[803] NDA, "Geological Disposal Upstream Optioneering Overview and uses of the 6 cubic metre concrete box", NDA Technical Note No. 18959097, March 2013

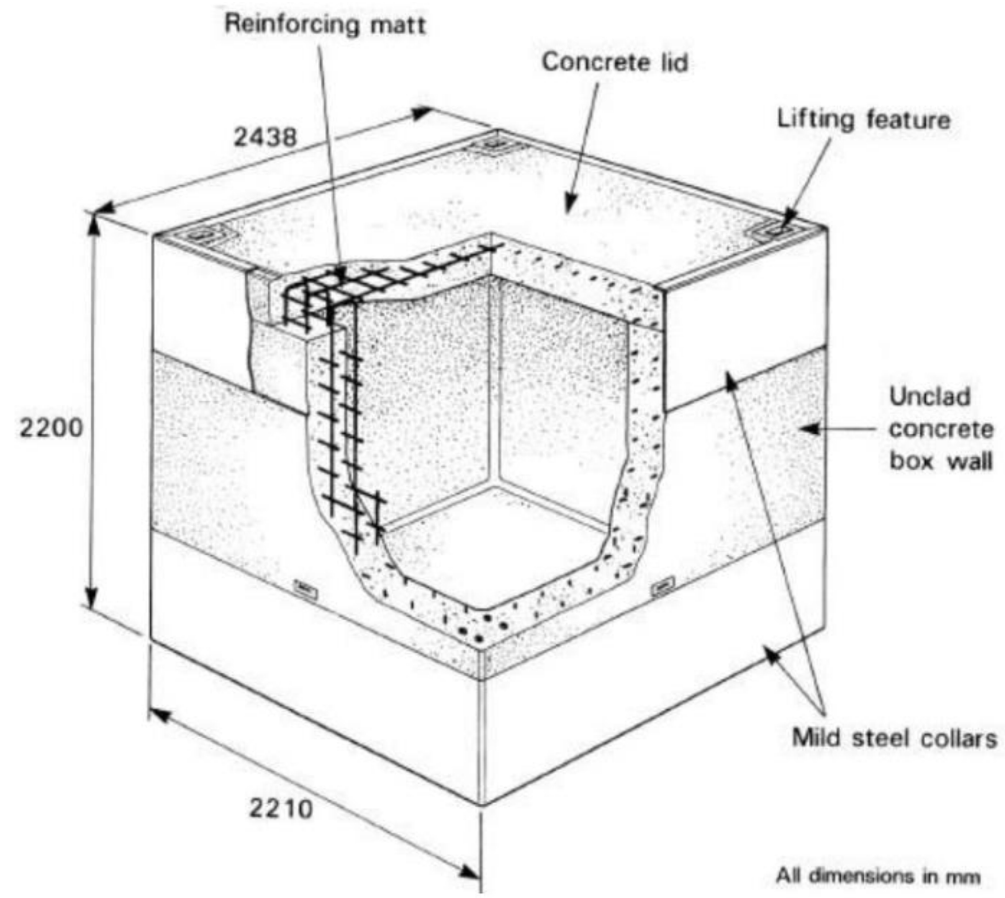


Figure 8.5-3 Cutaway of 6m3 concrete Box[804]

The 6m^3 box has been approved for use in the packaging of a range of wastes including

- Activated metals, such as stainless steels, mild steel, aluminium
- Activated other materials such as bioshield concrete
- Contaminated metals and other materials
- ILW Graphite
- Mixed wastes

For the packaging of legacy wastes such as fuel cladding and reactor components that are being retrieved from storage facilities, the approach that is being taken is to store the raw waste in 3m^3 stainless steel containers, equipped with internal concrete bunds to provide additional protection. These packages will be consigned to an interim above ground engineered store, until a GDF is

[804] NDA, "Geological Disposal Upstream Optioneering Overview and uses of the 6 cubic metre concrete box", NDA Technical Note No. 18959097, March 2013

available.[805] The intention is that the wastes in the package will be treated after interim storage to produce waste packages that meet the disposal criteria for the GDF when it is available.

Figure 8.5-4 3m3 stainless steel waste package[806]

8.5.5 Irradiated fuel and HLW

Transport and disposal[807] casks for spent fuel and HLW must provide enhanced radiation shielding and if necessary, also some heat dissipation.

The most commonly used containers for spent fuel are large and heavy steel containers that can host up to nearly 20 PWR or 50 BWR fuel assemblies (CASTOR type or similar), several equivalent models are shown in Figure 8.5-5. A variant in pre-stressed concrete is also available. Smaller casks are also available for single assemblies, pins or shorter fuel bundles.

Figure 8.5-5 Typical shipping casks for LWR spent fuel assemblies

Generally, these containers are suitable both for storage and transport (dual purpose), but at the time of writing, they are unsuitable for final disposal for the repositories currently under construction in

---

805 Sellafield Ltd, "Annual Research and Development Review 2018/19", Sellafield Ltd, September 2019

806 Sellafield Ltd, "Annual Research and Development Review 2018/19", Sellafield Ltd, September 2019

807 E.J. Harvey, M.J. White, J. Mackenzie, I. McKinley and S.P. Watson, "Geological Disposal Concept Options for Vitrified HLW", NDA, 2012

Finland and Sweden. In both sites, a dedicated treatment plant is planned to repack the spent fuel into a different canister for final disposal. Nevertheless, concepts for triple purpose casks (DCTC = Disposal Canister Transport Container) are under development.[808]

It is common practice for the shipping cask to be fitted with shock absorbers which may be required during transportation (see Figure 8.5-6).

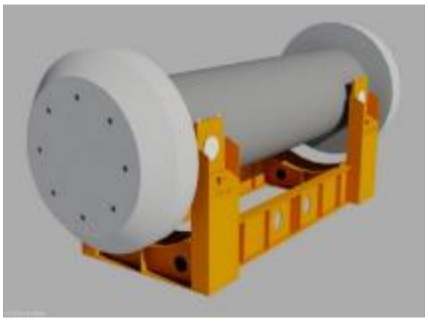


Figure 8.5-6 DCTC with shock absorbers and transport frame [809]

8.5.6 Liquid radioactive waste

Liquid LLW is generally stored in dedicated tanks, usually, these are double-wall stainless steel. For small quantities of liquid, 220-litre drums may be used for temporary storage.

For liquid ILW and HLW, special containers must be designed with shielding capabilities. There are very few standardised Commercial Off The Shelf (COTS) models available on the market. Waste owners are required to develop a specific technical solution, based on the characteristics and location of the waste. The containment of liquid ILW and HLW during transportation is an additional challenge. Depending on the country, there might be very few or no currently licensed transport containers for liquid waste. This requires waste owners to build on-site facilities for liquid waste treatment and conditioning, a solution generally not economically viable, especially as it is often small volumes requiring treatment. The availability of standardised containers for liquid waste transportation is crucial, in a waste disposal strategy based on centralised treatment centres.

8.5.7 Experiences/Case studies

AEA Technology coordinated a benchmarking exercise for transport packaging of nuclear and radioactive material. This exercise was organised by the OECD/NEA Reactor Physics Committee. [810] The UK's Transport Container Standardisation Committee has since created a code of practice for the

[808] Nuclear Decommissioning Authority, "Geological Disposal – Operational Aspects of Waste Transport", NDA Report No. NDA/RWMD/047, NDA, 2013.
[809] Nuclear Decommissioning Authority, "Geological Disposal – Operational Aspects of Waste Transport", NDA Report No. NDA/RWMD/047, 2013.
[810] A. Avery & H. Locke "NEA-CRP Comparison of Codes for Radiation Protection Assessment of Transportation Packages", AEA Technology, 1994.

design of transport packaging for radioactive material.[811] The IAEA provide a training manual on the safe transport of radioactive material.[812]

A comprehensive review of relevant case studies for shipment of spent nuclear fuel and HLW can be found in reference. [813]

[811] Transport Container Standardisation Committee, "Transport of Radioactive Material Code of Practice, Design of Transport Packaging for Radioactive Material", TCSC, December 2002
[812] IAEA, "Safe Transport of Radioactive Material", IAEA, Vienna, 2006
[813] K.J. Connolly and R.B. Pope, "A Historical Review of the Safe Transport of Spent Nuclear Fuel", ORNL/SR-2016/261, US-DoE 2016.

Table 8.5-1 Packaging improvement options by R&D

Field of application	Description	Advantages	Disadvantages/what is missing	Area of improvement
Storage and transport of SNF and HLW	Concrete or metal packages (casks)	Safe dry storage, standardisation, light surface facilities to host the packages	Heavy and expensive Not for disposal	Triple purpose, aging of structural material and high-burnup SNF assemblies for very long-term storage (more than 50 years)
Transport of Liquid waste	Typically, carbon steel drums[814]	Enable treatment at centralised facilities	Lack of currently licensed containers	Standardisation
Storage and transport of ILW	Containers	Common practice	Heavy and very expensive considering the size of the inner cavity	Standardisation
Conditioning LLW for storage/disposal	Grouting in concrete matrix	Common practice	Long-term behaviour is unknown	Innovative matrices

[814] IAEA, Handling and Processing of Radioactive Waste from Nuclear Applications, TECHNICAL REPORTS SERIES No. 42, Vienna, 2001

8.6 Characterisation and survey of containerised radioactive waste

Characterisation of waste packages tends to be built around a combination of results obtained from various destructive and non-destructive assay techniques, production data and calculation results.

This chapter is complementary to chapters 4, 6 and 7.

8.6.1 Methodology

The EU-CHANCE survey[815] (see also the international initiative at the beginning of Chapter 8) highlights that the methods used for radioactive waste characterisation depend on the country's approach. The majority of countries apply spectroscopic techniques (alpha, beta and gamma) as well as neutron measurements and dose-rate calculations. Few institutions use more "exotic" techniques, like muon tomography and calorimetry in the process of radioactive waste characterisation. Nuclide Vector (NV) and Scaling Factor (SF) methods are used by several European countries (e.g. Belgium, France, Germany, Spain, Sweden, United Kingdom) for radioactive waste characterisation. For this application, multiple measurement techniques could be used depending on the presence and need to measure specific radionuclides. Mostly, a combination of different techniques are used in combination (e.g. alpha spectrometry, mass spectrometry and gamma spectrometry). The sample and location for sampling are very important and have to be representative of the waste stream considered.

High-energy photon imaging (radiography, tomography) provides essential information on waste packages, such as density, position and the shape of the waste inside the container and in the possible binder, quality of coating and blocking matrices, presence of internal shields or structures, presence of cracks, voids, or other defects in the container or the matrix, liquids or other forbidden materials and facilitates the identification of the different types of soft operational waste (metals, glass, plastics or cellulose). It can also be used to reduce measurement uncertainties by using the images obtained to refine the numerical models for interpreting gamma spectrometry and neutron measurements.

The simplest devices are derived from the luggage screening equipment used for security at airports. These devices are equipped with a standard X-ray generator tube and a set of detectors. However, the geometry of these devices causes an adverse parallax effect when using X-rays to characterise the waste packages.

Imaging systems that use horizontal detectors which operate in the same plane as the X-ray or Gamma-ray source (using a linear or fan array of detectors) are used to obtain images without a parallax effect.

It is also possible to use detection shields in 2D-imaging and reconstruct the images obtained using cone beam geometry.

---

[815] https://3okrv814vuhc2ncex71gwd1e-wpengine.netdna-ssl.com/wp-content/uploads/2019/08/D2.2-CHANCE-Synthesis-EUG-questionnaire-answers.pdf

Radiological assessment is performed using a series of non-destructive techniques such as gamma-ray spectroscopy, which allows characterisation of a wide range of radioactive and nuclear materials, passive neutron coincidence counting and active neutron interrogation with the differential die-away technique, or active photon interrogation with high-energy photons (photofission), to measure nuclear materials. Industrial radiological waste characterisation systems are essentially equipped with gamma spectrometers and passive neutron[816] counters, where the inventory of non-measurable radionuclides relies on the use of a fingerprint which is normally established or validated through radiochemical analysis.

Prompt Gamma Neutron Activation Analysis (PGNAA) can be employed to detect toxic chemicals or elements which can influence the above measurements, such as neutron moderators or absorbers. Digital auto-radiography can also be used to detect alpha and beta contaminated waste.

These non-destructive assessments can be complemented by gas measurements, to quantify the radioactive and radiolysis gas releases, and by destructive examinations such as coring homogeneous waste packages or cutting the heterogeneous ones, to perform a visual examination and a series of physical, chemical, and radiochemical analyses on samples. These also allow for checking of the mechanical and containment properties of the packaging envelope, or of the waste binder, to measure toxic chemicals, to assess the activity of long-lived radionuclides or pure beta emitters, and to determine the isotopic composition of nuclear materials.

In the context of decommissioning, waste characterisation is limited by the knowledge of the waste matrices, the radiological distribution and the wide range of activity levels which vary from just a few Bq/g to several TBq/g.

Combined measurements are often used to identify radionuclides and determine the level of activity (Bq) contained in packages[817,818]. Each technique is used to meet the analytical target expressed in the form of a radionuclide to be characterised, a dynamic measurement range and associated uncertainty.

Gamma spectrometry on high-activity objects has been enhanced with the development of digital signal processing electronics such as ADONIS[819] and small-pixel spectrometry detectors based on Cd-Zn-Te crystals.

---

[816] F. JALLU, A. RENELEAU, P. SOYER and J. LORIDON, "Dismantling and decommissioning: The interest of passive neutron measurement to control and characterize radioactive wastes containing uranium", Nuclear Instruments and Methods in Physics Research, B 271 (2012), pp. 48-54.

[817] Deliverable (D2.3) of EU- H2020 CHANCE project (GA 755371): "R&D needs for conditioned waste characterization", 21/11/2019

[818] B. PÉROT, J.-L.ARTAUD, C. PASSARD and A.-C. RAOUX, "Experimental Qualification With a Scale One Mock-Up of the 'Measurement and Sorting Unit' for Bituminized Waste Drums", http://dx.doi.org/10.1115/ICEM2003-4597

[819] BARRAT, "Performance of ADONIS-LYNX System for Burn-up Measurement Applications at AREVA NC La Hague Reprocessing Plant", http://dx.doi.org/10.1109/ANIMMA.2013.6727894

The isotopic composition of plutonium is required for neutron counting processes. Substantial progress has been made in spectra processing techniques, thanks in particular to solutions such as the CEA's IGA code.[820] However, a large number of packages containing major actinides, in the presence of fission and/or activation products, which can mask the gamma emissions from the actinides, making neutron measurements essential. Interfering neutron emissions caused by (E, n) reactions is highly dependent on the chemical form (metal, oxide, etc.) and isotopic composition of the contaminant. The weight of the contaminant can be a major determinant of the measurable quantity which can then become difficult to interpret using coincidence techniques based on slow detectors like He-3 counters. Faster detectors like plastic scintillators (which are also cheaper given the shortage of He-3) are currently being investigated.[821]

Lastly, spontaneous fission can produce interfering emitters (such as Cm-244), which can mask plutonium emissions. Active neutron measurement may be the only method available for measuring fissile mass.[822]

The data produced complements calculation results. The isotope spectrum is determined for radionuclides based on depletion calculations and the processes involved (such as enrichment, chemical separation and conditioning) to give the fingerprint. Neutron and *gamma* particle transport codes are used to estimate the proportion of radiation emitted by waste packages. This identifies radioactive tracers, for example, Co-60 for activated waste produced by pressurised water reactors (PWR).

Radiochemical analysis results complement the radionuclides inventory. The relationship between the activity of radionuclides, like Cl-36 or Ca-41, which are difficult to measure and their associated radioactive tracer is a determining factor when defining and subsequently using a radiological waste characterisation system. The uncertainty associated with the reconstructed activity may also reach particularly high values for poorly characterised heterogeneous legacy waste. In this context, complementary techniques like photon imaging can be used to reduce the uncertainty associated with the lack of knowledge about the matrix.[823]

---

[820] A.-C. SIMON, "Determination of Actinide Isotopic Composition: Performances of the IGA Code on Plutonium Spectra According to the Experimental Setup", IEEE Transactions on Nuclear Science, 05/2011, 58 (2-58), pp. 378-385.

[821] C. DEYGLUN, B. SIMONY, B. PÉROT, C. CARASCO, N. SAUREL, S. COLAS and J. COLLOT, "Passive neutron coincidence counting with plastic scintillators for the characterization of radioactive waste drums", ANIMMA 2015, Advancements in Nuclear Instrumentation Measurement Methods and their Applications, 20-24 April 2015, Lisbon, Portugal

[822] F. JALLU, P.-G. ALLINEI, PH. BERNARD, J. LORIDON, D. POUYAT and L. TORREBLANCA, "Cleaning up of a nuclear facility: Destocking of Pu radioactive waste and nuclear Non-Destructive Assays", Nuclear Instruments and Methods in Physics Research, B 283 (2012), pp. 15-23.

[823] R. ANTONI, C. PASSARD, J. LORIDON, B. PEROT, M. BATIFOL, S. LETARNEC, F. GUILLAUMIN, G. GRASSI and P. STROCK, "Matrix effect correction with internal flux monitor in radiation waste characterization with the Differential Die-away Technique», IEEE Transactions on Nuclear Science, vol. 61, Number. 4 (2014), pp. 2155-2160.

The production of decommissioning waste is pushing all the boundaries of assay techniques, not only in terms of the intrinsic quality of detectors and their associated electronics but also when developing systems based on numerical simulation. Specially adapted analysis equipment will be needed for these new technologies to handle the high throughput of waste packages and new levels of activity. The model of centralised measuring stations has been replaced by remote characterisation stations located as close to the waste production site as possible.

8.6.2 Experiences/Case Studies

8.6.2.1 Experience in Europe

Information about methods applied in the characterisation of conditioned radioactive waste in Europe can be found in the synthesis of the end-users survey given in H2020- CHANCE. [824]

8.6.2.2 Implementation at CEA

A range of destructive and non-destructive measurements has been either already implemented in decommissioning projects or is under development at CEA.[825,826] The aim of this is to allow users to be able to perform the most complete radioactive waste characterisation. High energy X-ray imaging system, with an electron beam produced by a linear accelerator, are also widely used at CEA[827] (e.g. facility CINPHONIE[828] in Cadarache) for large, dense packages, equipped with a wide-field shield for qualitative radiography and tomography. A CdTe linear detector array with collimators is also used for quantitative tomodensitometry. A collaborative project, TOMIS,[829] is currently developing a shielded modular system to allow for the "in-field" use of tomography. At the other end of the scale, conventional large-volume detection technologies meet the need for characterising VLLW packages, for example, like the NaI (Tl) scintillators used on low-level counting stations.

[824] https://3okrv814vuhc2ncex71gwd1e-wpengine.netdna-ssl.com/wp-content/uploads/2019/08/D2.2-CHANCE-Synthesis-EUG-questionnaire-answers.pdf

[825] Monograph on "Decommissioning of nuclear facilities", E-DEN, © CEA Paris-Saclay, Éditions du Moniteur, Paris, 2017 *ISSN1950-2672*

[826] The characterization of radioactive waste: a critical review of techniques implemented or under development at CEA, France, Bertrand Pérot1, Fanny Jallu, Christian Passard, Olivier Gueton, Pierre-Guy Allinei, Laurent Loubet, Nicolas Estre, Eric Simon, Cédric Carasco, Christophe Roure, Lionel Boucher, Hervé Lamotte, Jérôme Comte, Maïté Bertaux, Abdallah Lyoussi, Pascal Fichet and Frédérick Carrel, https://www.epj-n.org/articles/epjn/full_html/2018/01/epjn170038/epjn170038.html

[827] N. ESTRE, D. ECK, J.-L. PETTIER, E. PAYAN, C. ROURE and E. SIMON,"High-Energy X-Ray Imaging Applied to Non Destructive Characterization of Large Nuclear Waste Drums",ANIMMA2013, Third International Conference on Advancements in Nuclear Instrumentation, Measurement Methods and their Applications, pp. 23- 27 June 2013, Marseille, France.

[828] Cellule CINPHONIE : une plateforme de tomographie haute énergie pour des objets de grandes tailles, David Tisseur, Laboratoire de Mesures Nucléaires (LMN), CEA, France, COFREND, Strasbourg 2017, https://www.ndt.net/search/docs.php3?showForm=off&id=21328

[829] Project TOMIS https://www.andra.fr/sites/default/files/2018-06/Fiche%20projet%20TOMIS%20VF.pdf

8.7 Material clearance

Clearance is defined as 'the removal of radioactive materials or radioactive objects within authorised practices from any further regulatory control by the regulatory body'.

8.7.1 Methodology and procedures

Significant quantities of material from nuclear facilities decommissioning projects have been generated in the past and will be generated in the near future. The clearance process allows the optimisation of the management of materials throughout decommissioning, reducing the generation of radioactive wastes.

During recent years, many countries have updated their legislation to either implement release standards on a national or case-by-case level. Guidance documents (see Chapter 7) are provided by the IAEA and the EU. Guidance regarding clearance levels is identical if the facility is in operation or undergoing decommissioning. A significant number of countries have developed national regulation based on this international guidance. In just a few cases, international guidelines have been directly adopted by countries as their national regulation (e.g. Japan and Spain) or are used by countries to regulate on a case-by-case basis (e.g. Italy). The mass-specific limits for the same radionuclide may vary from country to country, but always within the RP 89 limits[1]. One example is Cs-137 where the mass-specific limit varies from 0.1 Bq/g to 1 Bq/g. For mixtures of artificial radionuclides, the weighted sum of the nuclide specific activities or concentrations (for various radionuclides contained in the same matrix) divided by the corresponding release limit must be applied. This is typically referred to as a "sum of fractions" or "summation formula". In some countries, additional surface specific limits are applied.

During recent years, there have been significant efforts in several countries, to utilise radionuclide-specific clearance limits for unconditional and conditional clearance. In these situations, the summation formula is typically applied for situations where a mixture of radionuclides exists. To approve clearance, the result of summation must be less than 1.

Despite this effort, it appears that clearance levels remain insufficiently developed, and the harmonisation of release criteria between countries is not sufficiently developed to be used for the general adoption of recycling and reuse of materials from decommissioning projects. Furthermore, regulations introduced since 1996 have tended to reduce the clearance levels for significant radionuclides, like Cs-137 and Co-60, thus requiring more decontamination and characterisation before materials can be eligible for unconditional release.

An opportunity for greater harmonisation has arisen within the countries of the European Union (EU). In December 2013, the EU issued *2013/59/Euratom, Basic safety standards for protection against the dangers arising from exposure to ionising radiation* (EU, 2013). Implementation of the directive by member countries should help to harmonise the EU community and align them to the IAEA international guidance. The directive does not deal with the release limits and criteria for conditional clearance, and member countries can implement conditional release limits at a national level.

Harmonisation of national regulations is necessary not only to share techniques or procedures to minimise/reduce the volume of waste but also to increase public acceptance.

8.7.2 Instrumentation and logistics

Verification of the clearance levels uses a statistical approach based on the spectrometric measurements of a representative number of “handling units” of materials packed in suitable containers. The container measurements are performed by commercial total gamma counting chain knowing the final nuclide vector. For every representative container, the parameter X_n is calculated. When it is less than one this confirms that it meets the requirements in respect of the clearance level for the whole homogeneous group.

Figure 8.7-1 Material clearance measurement (Jose Cabrera NPP)

Logistics is a key element for the correct implementation of the material clearance process. Allocation of materials and tasks associated with the process (in situ initial characterisation, temporary storage of releasable materials, clearance measurement, dispatch of cleared materials) should be designed and considered in the waste management plan.

Implementation of clearance methodology implies that material should be organised in homogeneous “handling units” considering different factors (origin and nature of materials, isotopic composition, geometric shape, etc.). Accurate segregation of materials is an essential requirement to assure the success of the clearance process.

Figure 8.7-2 Material clearance process (Enresa)

Dismantling activities can produce a huge amount of releasable materials of different nature (rubble, metallic scrap, soils, etc.). Development of automated systems for segregation and sorting of materials, arising from decommissioning works, represents an opportunity to improve the performance of the clearance process. Another way to increase the efficiency of the process is to develop technologies and procedures to characterise bigger "handling units" to simplify logistics linked to the process.

Figure 8.7-3 Gravels segregation system associated to soil washing plant (Jose Cabrera NPP)

8.8 Management of hazardous and toxic materials (asbestos, lead in paint, etc.)

The type of waste arising from decommissioning is often different from the waste generated during operation or routine maintenance of the nuclear facility. These differences may include its chemical and radiological properties, the physical form and the general quantity. Considering these specific characteristics, some of the waste could be considered as problematic, for example, waste for which application of common methods of handling, treatment and conditioning are not appropriate and therefore requires specific management options. Waste may also be considered problematic because it is hazardous due to either its physicochemical properties or its inherent toxicity. These types of material represent a potential hazard to human health or the environment when improperly treated, stored or disposed of, or otherwise mismanaged.

For such hazardous materials, proper selection of appropriate waste management and material management options during decommissioning planning is particularly important for logistics and safety considerations.

Several preparatory activities should be performed during the transition period to facilitate subsequent dismantling and decontamination operations including:[830]

- Radiological and physical inventory of the plant
- Decontamination of systems
- Discharging systems and components
- Draining circuits and systems
- Removal of non-radiological components
- Modification/Construction of new auxiliary systems / facilities
- Removal of hazardous materials

From the beginning of the project there is a requirement to carry out an exhaustive inspection campaign for the identification of different types of toxic non radiological materials (asbestos, chemicals, lead, etc.) existing in the facility to reduce risks during decommissioning. The risks associated with non-radiological hazards do not decrease with time. The removal of toxic materials may result in an overall decrease in risk improving safety conditions of operators and workers conducting the dismantling activities.

Asbestos is a material that was widely used during the construction of older nuclear plants and nuclear installations as in other industries. This material can be found in several locations of the nuclear facilities: piping and systems heat insulation, claddings, cement, paintings, etc. Reactor pressure

[830] V. Michal and V. Ljubenoc, "IAEA Perspectives on Preparation for Decommissioning", PREDEC 2016: IAEA Perspectives on Preparation for Decommissioning, February 16-18, Lyon, France 2016

vessels and reactor containment and turbine buildings were commonly insulated with material containing asbestos.

Figure 8.8-1 : Insulation in Turbine building and reactor vessel (Jose Cabrera NPP)

The removal of asbestos in conventional areas of the nuclear facility can be treated similarly to other industries and following the national regulations. The removal of asbestos can require the use of scaffolding depending on the location of materials. During these activities, it is necessary to ensure that asbestos is contained to maintain a safe working environment for the workforce. Asbestos is a concern when the fibres become airborne because it is only when the fibres are present in the air that people can inhale them. Safety precautions to be observed during asbestos removal are more stringent than for many other decommissioning works or activities at a nuclear facility. The removal of the contaminated insulation greatly improves the work environment for further decommissioning work. After removal, the asbestos and any contaminated material are placed into sealed heavy-duty, labelled plastic bags.

In some countries, piping and system heat insulation is removed early after final shutdown to improve the accessibility of equipment. Although this practice may generate high volumes of LLW, asbestos or insulating material requiring temporary storage, it is considered a good practice as it helps to enhance characterisation of both the insulation materials and the equipment. This methodology can save time on the critical path which results in cost savings.

The cost of surveying, removing and disposal of asbestos can be significant depending on the characteristics and the extension of these materials in the plant. The decommissioning programme can be affected because of the appearance of unexpected hazardous materials.

Figure 8.8-2 Removal of asbestos (Jose Cabrera NPP)

Presently the use of asbestos is forbidden in many countries. Due to its hazardous nature and specific physicochemical properties there are limited possibilities for recovery or reuse of asbestos. Nevertheless, some treatments have been developed. Asbestos materials can be thermally and chemically treated to alter the fibres such that they are non-hazardous and can be recycled.[831] Thermally treated asbestos can be recycled for use in cement, ceramics and other products. Asbestos waste (pure chrysotile asbestos and asbestos cement) can be treated under hydrothermal conditions using different acids in various temperatures in order to produce a material that is non-toxic and can be used as an adsorbent for petroleum pollutants.

Oils or other chemicals, that are not required, and the plant is shut down and awaiting decommissioning, should also be removed as soon as possible from buildings containing radioactive material. A fire protection programme should be implemented for managing oils and chemicals. After waste removal, the fire protection system needs to be assessed according to the remaining risk and some fire protection features may be retired or require modification.

Lead is widely used in nuclear facilities as a shielding material in the form of bricks, sheets or wool. The physical form of the lead shielding material depends on the nature of its use. In addition, lead based paints and primers were routinely used during the construction of many facilities. Initially the toxic properties of lead were not fully understood, and as a result lead became widely used and the various sources of lead are generally not enclosed, encapsulated or labelled. In nuclear facilities lead is predominantly contaminated rather than activated.

There are several possibilities for the recovery and reuse of lead inside and outside of the nuclear industry. For reuse outside of the nuclear industry decontamination of the lead is generally required. There are different methods (shaving, blasting, melting, etc.) that can be employed for this purpose. The method used is dependent on the form of the lead (i.e. mechanical methods are preferred for lead blocks) and the form of the contamination. The methods include mechanical and non-mechanical

[831] Paolini, V., Tomassetti, L., Segreto, M. et al. "Asbestos treatment technologies", J Mater Cycles Waste Manag 21, 205–226, 2019

means. The reuse of lead in nuclear facilities is encouraged, especially if the waste is activated and the clearance of this material is difficult to demonstrate. It is difficult to confirm whether contamination has been incorporated into the matrix, because of the density of the lead. The history of the previous use of the lead must be known in order to ensure that the contamination is only a surface effect, otherwise samples of the lead need to be taken to ensure that it meets the clearance levels.

A common problem during decommissioning is the occurrence of polychlorinated biphenyls (PCB). Due to their technical properties (e.g. water insolubility, fire resistance, long life, chemical inertness, high thermal conductivity, high electrical resistance) and their low cost, PCBs were widely used in technical installations, including nuclear facilities, as components in many applications (electrical transformer and machine oils, epoxy paints, lubricants) When such types of materials are used in controlled areas they can become radioactively contaminated.

The inherent hazards associated with this class of compounds do not allow the future reuse of PCB-containing material. In many countries the production and use of PCBs is now forbidden.

During decommissioning, the initial removal of PCB-containing material and its processing are the biggest problems concerning the personal protection of workers from exposure to these types of material. Once PCB containing waste is removed the recommended approach is incineration it in an appropriate treatment facility at temperatures exceeding 1200°C.

The different types of wastes that can be generated during the decommissioning of a nuclear facility can be problematic due to a combination of their conventional hazardous or toxic nature and their radiological hazard.

Other problematic material may arise, especially in research facilities in which complex tasks and a variety of experiments have been undertaken over decades. These, often small, quantities of hazardous waste for have no clear disposal path. Management of these materials is challenging and should be considered on a case by case basis.

The existence of hazardous and toxic materials should be considered in the decommissioning planning. Poor or incomplete record-keeping on the use of potentially hazardous material at the facility can result in unexpected combinations of problematic waste during decommissioning. The decommissioning plan should include the careful characterisation of all problematic material present in the facility to avoid difficulties at a later stage of the decommissioning process during the management of the generated waste.

Some of the toxic and hazardous but non-radiological waste existing in nuclear plants are common to other industrial sectors. Therefore, a standardised approach considering the best practises in other industries is recommended. Management of problematic waste combining toxicity and radiological contamination can represent an issue in establishing treatment procedures and final disposal.

8.9 Conventional and cleared materials recycling (circular economy)

The EU parliament has started to promote a move away from the traditional, linear, economic model (essentially 'take-make-consume-throw away') to a more holistic approach aimed at keeping waste production to a minimum by recycling material and increasing its value by re-using the same materials multiple times (Figure 8.9-1). This has been classed as a circular economy and defined as:

'A model of production and consumption, which involves sharing, leasing, reusing, repairing, refurbishing and recycling existing materials and products as long as possible.'[832]

Figure 8.9-1 Circular economy

The concept of the circular economy relates to the reuse and recycle elements of the waste hierarchy. The waste hierarchy (as already introduced in 8.1 and represented in Figure 8.1-1) provides a tool to which methods of waste management has the most beneficial impact on the environment. Prevention is at the top of the diagram (most preferable) followed by reuse, recycling, recovery and then disposal at the bottom (least preferable). The current waste hierarchy was introduced in the 2008 EU Waste Framework Directive (Directive 2008/98/EC).

The utilisation of incorporating life cycle thinking, for example by completing a life-cycle assessment (LCA), into decision-making processes and technological development embraces circular economy principles. The inclusion of LCA is highlighted in the EC's Better Regulation Toolbox as a method for supporting the impact assessment of policies. This approach is being utilises by international initiatives, such as PREDIS.[833]

Significant volumes of waste arise from decommissioning nuclear facilities. A large portion of the waste is concrete and steel with very low/ no radioactivity and can therefore be recycled or reused (subject

[832] European. Parliment, "Circular economy: definition, importance and benefits," European Parliment, 10 04 2018 https://www.europarl.europa.eu/news/en/headlines/economy/20151201STO05603/circular-economy-definition-importance-and-benefits.

[833] https://predis-h2020.eu/predis-project/.

to compliance with national/regional regulations). [834, 835] Recycling these materials can see benefits in the form of reduced long-term storage and, in some cases, off-setting the cost of some decommissioning activities. There are several options for recycling and reusing decommissioned material:[836]

- Material which is uncontaminated and can be completely released.
- Material that can be melted in a regulated environment followed by recycling for consumer products (conditional clearance).
- Material contaminated with short half-life radioisotopes that can be melted and fabricated in a regulated environment and released for specific industrial applications (e.g. steel bridge).
- Material that may be recycled/reused for within the nuclear industry.

To support the promotion of recycling and reusing nuclear material and in the anticipation of a large increase in decommissioning in the immediate future, the NEA Co-operative Programme on Decommissioning (CPD) commissioned a Task Group on Recycling and Reuse of Materials (TGRRM) in 2014. The TGRRM was aimed at reviewing practices both for metals and for other materials (notably concrete) arising in significant volumes from decommissioning activities. [837]

The task group concluded after treatment, significant quantities of materials and waste generated from decommissioning could be recycled and reused.[838]

8.9.1 Factors influencing recycling and re-use activities

The IAEA published a report in 2000[839] which outlined the factors that influence whether radioactive and non-radioactive materials arising from nuclear fuel cycle facilities should be reused/ recycled or whether alternative routes are more applicable (such as storing in a repository). The report identified the main factors which determine recycle/ reusability of waste to be:[840]

- Availability of appropriate clearance/release criteria;

---

[834] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[835] World Nuclear Association, World Nuclear Association, 05 2020. [Online]. Available: https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-wastes/decommissioning-nuclear-facilities.aspx#ECSArticleLink3 .
[836] World Nuclear Association, World Nuclear Association, 05 2020. [Online]. Available: https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-wastes/decommissioning-nuclear-facilities.aspx#ECSArticleLink3 .
[837] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[838] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[839] LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.
[840] LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.

- Consideration of cost;
- Technical feasibility;
- National waste management policy and strategy;
- Public acceptance of recycle and reuse options.

Each of the criteria cannot be assessed in isolation, as the inter-relationship between them is complex, and even when all the criteria are satisfied, recycle and reuse may not be selected as a way of dealing with the waste or contaminated materials. [841]

8.9.1.1 Quantity of material

The substantial quantity of material generated during the decommissioning of nuclear facilities creates a large opportunity to practise recycling and reuse. A significant portion of waste arising from decommissioning nuclear facilities is only slightly contaminated with radioactivity and can therefore be recycled/reused. This provides a more economical approach and can have a higher impact on reducing the size of repositories.

8.9.1.2 Appropriate clearance and release criteria

Several national and international standards and release criteria have been published to support the recycle/reuse of decommissioned material. High level guidance has been developed by the IAEA and EC to support authorities when developing clearance levels[842]:

- IAEA RS-G 1.7, Application of the Concepts of Exclusion, Exemption and Clearance.
- EC Radiation Protection 89 (RP 89) – Recommended Radiological Protection Criteria for the Recycling of Metals from the Dismantling of Nuclear Installations.
- EC Radiation Protection 113 (RP 113) – Recommended Radiological Protection Criteria for the Clearance of Buildings and Rubble from the Dismantling of Nuclear Installations.
- EC Radiation Protection 122 (RP 122) – Practical Use of the Concepts of Clearance and Exemption – Part 1, Guidance on General Clearance Levels for Practices.

Individual countries apply specific regulations for clearance of nuclear waste for unrestricted use. Most of the regulations are based on the guidance documents set out above, however due to countries applying regional laws, precedents and restrictions, there is a lack of international consensus around threshold clearances and possible uses of recycled metal from nuclear facilities.

[841] LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.

[842] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.

Within the EU, efforts are ongoing with several member countries to create standard criteria across the member states. In 2013, the EU published a Council Directive laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation (2013/59/EURATOM). [843]

8.9.1.3 Cost

Cost is a major factor which determines whether recycling is feasible. Several aspects need to be considered when addressing the cost, some of which are described below[844]:

- Cost of retrieval and processing of nuclear related materials, involving its removal, characterisation, decontamination, transport and licensing.
- Contingency funds to mitigate financial risks incurred by unforeseen events.
- Marketability of the resources must be researched to determine the expected selling price by forecasting.
- The cost of available waste management options such as storage or disposal that may support or hinder the use of recycling/reuse.
- Understanding of national policies that will allow help recycling (tax breaks) or hinder it (policies that limit the resale of materials).

8.9.1.4 Technical Feasibility

The availability of technologies to treat, recycle and reuse materials in the fuel cycle is essential. For a technology to be feasible secondary waste should be minimised as large amounts of secondary waste would add to the processing and disposal demands and would in turn involve an additional cost and environmental burden.

Several technologies exist at various levels of the Technology Readiness Level (TRL) spectrum which can support a circular economy. The main technologies which identify whether materials can be recycled are related to decontamination and characterisation. [845] Examples of such technologies are covered in other sections of this report.

8.9.1.5 Sustainability

Several methods exist to sustainably recycle decommissioned material in the nuclear industry. Generally, this involves minimising the contamination as much as possible which can be achieved via

---

[843] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[844] LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.
[845] LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.

proper ventilation in nuclear facilities; tie down coatings and even dry decontamination processes (such as abrasive blasting). [846]

Materials such as copper from motor windings, exotic alloys and shielding materials like lead can be desirable to recycle. Demolition, which may expose some of these materials, comes with its own issues around wastes such as asbestos, wood or plastic that should be utilised in recycling if possible.[847]

Materials that are lightly contaminated or contaminated with a small range of known, characterised contaminants are more readily recycled. The large scale of nuclear decommissioning makes it an attractive source for recycling and creating value from recycling.

With a desire for other waste routes for very low-level waste (VLLW), recycling can potentially meet the needs of minimising waste in the nuclear sector. Delay and decay approaches have been known as effective means to lower the cost for decontamination. The delay and decay approaches describe the process of leaving metallic waste contaminated with radionuclides with short half-lives to decay to below the clearance threshold, meaning that the waste can be reclassified.[848]

8.9.1.6 Access to expertise and competence

The expertise and competence to implement circular economy, exists in different countries. For example:

- Studsvik Nuclear has developed methods to recycle metallic components from both operational plant and during decommissioning. By 2015, Studsvik had processed, by melting, 32,000t of carbon steel, 5200t stainless steel, 2033t aluminium, 1153t lead, and 3896t copper cables.
- Within France, work has been completed on the technical considerations for recycling steel and concrete in order to optimise decommissioning; this is set out as part of the 'Plan national de gestion des matières et des déchets radioactifs' referred to commonly as the PNGMDR.
- LLWR Ltd announced a £65M contract for the treatment of radioactive waste from Active Collection Bureau, Augean Treatment, Cyclife UK, Tradebe Inutec, Urenco Nuclear Stewardship and Westinghouse. Part of this investment is to explore the possibilities of recycling: since 2010, over 12 000 tonnes of metal have been recycled by LLWR through this approach resulting in savings of ~£30M[849]. Furthermore, in the UK recycling of components has been explored at Berkeley where Magnox Ltd has worked with LLWR Ltd to generate a solution for boilers involving metal recycling[850].

---

[846] LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.
[847] LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.
[848] Vlaanderen, "Acceptatie criteria voor het niet-geconditioneerd radioactief afval van," Vlaanderen, 2014.
[849] SKB, "Avfallshandbok – låg- och medelaktivt avfall, version 4.0", SKBdoc 1195328," Svensk Kärnbränslehantering AB, Stockholm, 2014.
[850] Vlaanderen, "Acceptatie criteria voor het niet-geconditioneerd radioactief afval van," Vlaanderen, 2014.

- In 2012 five steam generators from UK plants were shipped to Studsvik in Sweden for recycling. Cyclife UK have a plant in Cumbria, to recycle materials from nuclear facilities, and this became fully operational in 2013, processing 2000 t of metal from numerous sites and recycling 96% of it.[851]

8.9.2 Experiences/Case Studies

The substantial quantities of materials generated during the decommissioning of nuclear facilities create opportunities for recycling and reusing. Several studies have been conducted globally which take advantage of the recovery and reuse of materials[852].

Table 8.9-1 Recycle/reuse applications in decommissioning[853]

Project	Site	Country	Material	Amount (tonnes)	Physical Form	Endpoint
BWR turbine rotors Ringhals NPP	Ringhals NPP	Sweden	Steel	360	Large/whole component	Conventional recycler
Berkeley boilers	Berkeley	United Kingdom	Steel	3200	Segmented component/ melting	Conventional recycler
Lead from removeable shielding	BR3 NPP	Belgium	Lead	34	Encapsulated lead/melting	New hot cells
Concrete from PWR Containment	Ringhals NPP	Sweden	Concrete	200	1 tonne concrete blocks	On-site construction
Release of cable	Wiederaufarbeitungsanlage	Germany	Copper	4.15	Off-site cable shredder	Conventional recycler

851 World Nuclear Association, World Nuclear Association, 05 2020. [Online]. Available: https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-wastes/decommissioning-nuclear-facilities.aspx#ECSArticleLink3 .

852 LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.

853 Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.

Project	Site	Country	Materia l	Amount (tonnes)	Physical Form	Endpoint
	Karlsruhe (WAK)					
Concrete debris recycling	JRR-3 research rector	Japan	Concret e	1 800	Concrete rubble	Site remediation
Plant decommissioni ng	NPP Vandellos-1	Spain	Concret e	78 962	Concrete structures	Reuse on-site

8.9.2.1 Metals

Metals (particularly steel) are the most common materials recycled and reused. Metals are expensive when compared to other construction materials and are relatively straightforward to decontaminate[854].

Metallic components with activity under the 'clearance threshold' (the regulatory radioactivity allowed in materials) can be reused/ recycled for either restricted nuclear use or defined non-nuclear applications. Work has previously been conducted by IAEA, OECD and EC to define the requirements for unconditional release (the release of metal for wide use in non-nuclear industry)[855]. Metallic components are typically decontaminated or melted to reduce the activity level prior to recycling[856]. For example, at NPP Vandellos-1 (Spain) 8000t of materials (mainly metals) from the active area and an additional 8000t from conventional areas/components have been decommissioned and sent for recycling.

The majority of waste which is recycled is cleared prior to reuse, however examples also exist where waste has been reused without clearance. In these cases, the materials have been reused for the purposes of shielding or construction of radioactive waste containers[857]. In France, Socodei has utilised scrap metal from nuclear facilities to produce radiation protection in waste packages. Lead has also

[854] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[855] U.K. Government, "GOV.UK," 18 May 2020. [Online]. Available: https://www.gov.uk/government/news/llwr-awards-multi-million-pound-metal-framework-contract.
[856] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[857] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.

been utilised in this manner. 72t of contaminated ferrous material was sent by NPP Vandellos-1 (Spain) to EnergySolutions to recycle in the manufacture of shielding for the Fermi Laboratory (USA)[858]. EnergySolutions is a nuclear services company, which recycles and repurposes metal for use in the nuclear industry, particularly in repurposing metal to create shield blocks[859].

Copper is often recycled due to its high value. Most of the copper is found in cables and are either recycled as intact cables or can be processed via separating the copper from the plastic insulation. Examples exist in WAK, Germany, where 4.15t of copper from decommissioned cables were cleared and sent for recycling[860].

Recycling of aluminium is attractive both from an environmental and economical perspective. A large amount of energy is required to produce aluminium and it can be problematic to store. The treatment of aluminium with highly alkaline grout causes the rapid production of hydrogen gas, a product from aluminium's oxidising reaction. For example, in Belgium the acceptance criteria limits the amount of aluminium to 10kg when stored with unconditioned waste in a drum.[861] Other organisations such as LLWR[862], Vlaanderen[863] and SKB[864] employ similar acceptance criteria.

At the Ningyo-Toge uranium enrichment plants (Japan), 11t of aluminium from gas centrifuges was cleared to meet the requirements for recycling. The aluminium was reused for the construction of flower beds at the uranium gallery site. [865]

Lead is similar to aluminium such that recycling is preferred to storage due to its chemotoxic properties. Examples exist of lead being reused in the nuclear sector. For example, approximately 32t of lead was recycled and processed into shield walls and U-shaped lead blocks.[866]

---

858 Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
859 Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
860 Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
861 Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
862 EPRI, "Graphite Decommissioning: Options for Graphite Treatment, Recycling, or Disposal, including a discussion of Safety-Related Issues," Palo Alto, CA: 2006. 1013091.
863 IAEA, "Reducing Risks in the Scrap Metal Industry", Vienna, 2005.
864 NTI, "Truck Carrying 25 tons of Radioactive Scrap Metal Detained at Chernobyl," 11 September 2009. [Online]. Available: https://www.nti.org/analysis/articles/truck-carrying-25-tons-radioactive-scrap-metal-detained-chernobyl/.
865 Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
866 LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.

8.9.2.2 Concrete

Decommissioning nuclear facilities results in large amounts of concrete waste, the majority of which arises from cleared buildings. The buildings are typically cleared after surface and/or mass activity measurements prior to demolition. Concrete can be recycled by either treat as conventional rubble; cut into smaller segments and recycled individually; or by being crushed on site.[867]

Examples do exist where concrete from decommissioned facilities has been recycled and reused both for conditional and unconditional release. For example, at the NPP Vandellos-1 (Spain), 77000t of concrete from building structures together with 1900t of cleared concrete from active areas have been reused on site for land restoration purposes.[868]

Figure 8.9-2 One-tonne concrete blocks for clearance[869]

The recycling of conditionally cleared concrete has mainly been applied for site remediation purposes. Backfilling with concrete crushed on site can reduce exposure to the general public and therefore allows higher clearance levels compared to unconditional clearance. This procedure has been applied for decommissioning of the sorting plant at the uranium processing facility in Ranstad (Sweden).[870] There are other potential applications for reusing concrete that has undergone conditional clearance in the nuclear industry. These include in the preparation of the filler, backfill or encapsulation material for waste drums and containers in near-surface storage sites; fabrication of concrete for certain radiological protection shields; and the fabrication of waste storage containers.[871]

8.9.2.3 Graphite

Across the world graphite has been used extensively as a reflector and moderator material in over one hundred nuclear power plants, research reactors and plutonium production reactors. Over 230,000

---

[867] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[868] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[869] NDA, "Strategy: Effective from April 2011," 2011.
[870] Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.
[871] NDA, "Strategy: Effective from April 2011," NDA, 2011.

tonnes of irradiated graphite which will ultimately need to be managed and treated, potentially as radioactive waste. The majority of graphite exists either in-situ within reactors or in vault/silo storage [872].

Historically, the concept of recycling and reusing graphite has not been considered seriously due to the abundance of carbon. The production process for graphite is complex and it would be difficult to implement recycled graphite into this process. As a result, utilising recycled graphite for new graphite is not currently cost effective.[873]

However, with the concept of circular economy and waste minimisation becoming more important there is a new impetus in exploring techniques to treat and reuse graphite. The case for recycling is further strengthened when comparing the significantly higher cost of disposing graphite as ILW against the cost of manufacturing new graphite [874].

To address this EURATOM have commissioned research programmes focused on the management and treatment or irradiated graphite:

- CARBOWASTE
- GCR-MINWASTE

8.9.3 Challenges

Several challenges exist when considering recycling and reusing material from decommissioning nuclear facilities. The NEA task group interviewed SME’s to identify the most common challenges and summarised this in Figure 8.9-3.

[872] NDA, "Strategy: Effective from April 2011," NDA, 2011.
[873] European Commission, "Recommended Radiological Protection Criteria for the Clearance of Buildings and Buidling Rubble Arising from the Dismantling of Nuclear Installations," 1999.
[874] European Commission, "Recommended Radiological Protection Criteria for the Clearance of Buildings and Buidling Rubble Arising from the Dismantling of Nuclear Installations," 1999.

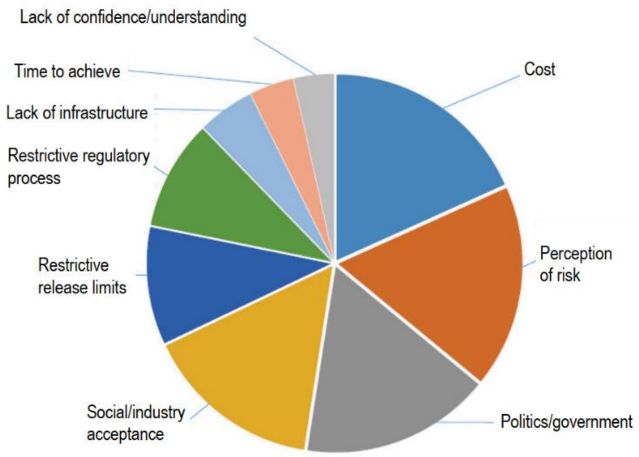


Figure 8.9-3 Stakeholder-identified challenges to the recycle and reuse of materials[875]

The main technical challenge facing recycling of materials in the nuclear context is the ability to characterise the material in question. This slows down the processes required to gain acceptance for free release practices required to recycle and reuse materials[876].

Part of the difficulty with recycling materials from nuclear facilities is public perception. Incidents have occurred regarding the improper storage, recycling, and selling of material[877] which have caused a detrimental impact on public confidence in the industry's ability to recycle:

- A case in France in 2000 that showed a nuclear worker's watch showed up as contaminated with Cobalt-60. The contaminated metal in the watch was traced back to China where metal from a nuclear site was improperly recycled.
- Turkey in 1993 where disused radiotherapy sources were improperly transported, stored and sold, resulting in 18 hospital admissions.
- Spain in 1998 where metal contaminated with Cs-137 was melted, contaminating eventually 500 tonnes of dust that was released into a marsh with remediation costs reaching in excess of $25M.
- Thailand in 2000 where a radiotherapy source was improperly stored and sold resulting in 1870 people being exposed and 3 people dying within 17 days of exposure.

With such events negatively linking the resale of nuclear decommissioning materials into non-nuclear industries, it is possible that more education and assurance is needed to address the worries of the general public. This also highlights the need for effective characterisation, sentencing, management and regulation/enforcement. It is also worthy of note that on-going, illegitimate sales of metal from

---

875 Nuclear Energy Agency Organisation for Economic Co-operation and Development, "Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities (NEA No. 7310)," NEA OECD, 2017.

876 LLW Repository Ltd, "Waste Acceptance Criteria – Low Level Waste Disposal, WSC-WAC-LOW, V5.0 Issue 1," 2016.

877 NEA, "Decontamination and Demolition of Concrete Structures," OECD, NEA/RWM/R(2011)1, 2011.

around the Chernobyl site have furthered negative sentiment to recycling on nuclear sites, emphasising the need for transparency and development of public education in this area.[878]

From an international perspective, work is still needed to consolidate and standardise the regulations for unconditional release across different countries. With countries applying regional laws, precedents and restrictions, there is a lack of international consensus around threshold clearances and possible uses of recycled metal from nuclear facilities.

There is also increasing concern about double standards developing in Europe which permit ~30 times the dose rate from non-nuclear recycled materials than from those out of the nuclear industry (only Norway and Holland provide consistent standards). For example, scrap steel from gas plants may be recycled if it has less than 500,000 Bq/kg radioactivity. This level is one thousand times higher than the clearance level for recycled material (both steel and concrete) from the nuclear industry, where materials above ~500 Bq/kg may not be cleared from regulatory control for recycling.[879] As such, materials with the same radionuclides, at the same concentration, can either be sent to deep disposal or released for use in building materials, depending on its origin.

In 2011, 16 decommissioned steam generators from Bruce Power in Canada were to be shipped to Sweden for recycling. Although the Canadian Nuclear Safety Commission (CNSC) approved Bruce Power's plans in 2011 and confirmed steam generator processing is an excellent example of responsible and safe nuclear waste management practices, this caused public controversy at the time, and following the Fukushima nuclear accident plans for this shipment.

[878] Alan Wareing, Liam Abrahamsen, Anthony Banford, Martin Metcalfe and Werner von Lensa, "CARBOWASTE: Treatment and Disposal of Irradiated Graphite and Other Carbonaceous Waste," European Commission, Brussels, 2013.

[879] World Nuclear Association, World Nuclear Association, 05 2020. [Online]. Available: https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-wastes/decommissioning-nuclear-facilities.aspx#ECSArticleLink3 .