

Management of Radioactive Waste after a Nuclear Power Plant Accident



Radioactive Waste Management

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after a Nuclear Power Plant Accident**

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Foreword

In 2014, the NEA Radioactive Waste Management Committee (RWCM) established the Expert Group on Fukushima Waste Management and Decommissioning R&D (EGFWMD). The primary aim of the EGFWMD was 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.

Members of the group include experts who have experience in waste management, in managing radiological contamination situations or in decommissioning and waste management R&D after the Three Mile Island and Chernobyl accidents, as well as in existing contamination situations like Windscale or even potential situations like the Kola Peninsula. The experts provide technical opinions and ideas for waste management and R&D at the Fukushima site.

The EGFWMD has held five meetings and two site visits to the Fukushima Daiichi nuclear power plant and to the Chernobyl nuclear power plant, and has drafted this report based on its work since 2014. The report provides information on post-accident waste management and decommissioning challenges. It also provides lessons learnt from past nuclear accidents or site remediation and summarises important points in post-accident waste management.

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Executive summary

Introduction

The NEA Radioactive Waste Management Committee (RWMC) established the Expert Group on Fukushima Waste Management and Decommissioning R&D (EGFWMD) in 2014. Members of this group include experts who have gained experience in waste management, in radiological contamination or in decommissioning and waste management R&D after the Three Mile Island accident and the Chernobyl accident. They provide technical opinions and ideas for waste management and R&D at the Fukushima Daiichi site.

The EGFWMD was established with the primary aim of offering advice on the management of large quantities of Fukushima Daiichi on-site waste that has complex properties, and of sharing experiences with the international community and NEA member countries. The aims of the expert group are:

- to share knowledge on post-accident waste management and decommissioning challenges arising at the Fukushima Daiichi nuclear power plant;
- to provide advice to the Japanese government on the R&D programme being carried out, specifically on waste management and decommissioning of the Fukushima Daiichi nuclear power plant;
- to draft a report to be submitted to the RWMC on specific and general lessons learnt.

The EGFWMD held five meetings and two site visits to the Fukushima Daiichi nuclear power plant site and the Chernobyl nuclear power plant. Members of the group visited waste storage facilities and treatment facilities in both nuclear power plants in order to gain a deeper understanding of the facilities and activities on waste management through discussions with workers.

Figure E1. Site visits to Fukushima Daiichi (left) and Chernobyl (right) nuclear power plants



Table E1 provides an outline of waste volume storage at the Fukushima Daiichi site.

Table E1. Waste volume stored at Fukushima Daiichi nuclear power plant as of the end of 2015

Waste		Storage volume
Debris (metal, concrete)	Less than 0.1 mSv/h	115 600 m ³
	0.1-1.0 mSv/h	31 400 m ³
	1.0-30 mSv/h	19 700 m ³
	Over 30 mSv/h	6 200 m ³
	Total	172 900 m ³
Trimmed trees	Total	85 100 m ³
Used protection clothes	Total	66 000 m ³

Chapter 1: Case studies

Members of the EGFWMD presented background and contextual information about decommissioning and waste management at four sites: the Chernobyl Nuclear Power Plant (Ukraine); the Three Mile Island nuclear power plant (United States); the Site for Temporary Storage of Spent Fuel and Radioactive Waste in Andreeva Bay (Russia), and the site of the Windscale Pile fire (United Kingdom), as well as the Fukushima Daiichi nuclear power plant (Japan).

Commentary and advice on the experiences from case studies have been consolidated in three areas: i) dialogue, regulators and other stakeholders; ii) waste description: physical and chemical nature, radiological characterisation and waste classification/categorisation; and iii) conditioning, storage and disposal. This material is supplemented by overall conclusions from the expert group.

Chapters 2 and 3: Dialogue, regulators and other stakeholders

Applying existing regulations to accident situations, in so far as these existing regulations are protectively and practically effective for application in the circumstances of the specific accident, has been shown to be the preferable solution. Contingency regulations for dealing with clean-up and waste management should be developed with the understanding that an accident may occur in the future and existing regulations may not be sufficient to address the prevailing circumstances. Protection objectives must be clear but the method(s) to achieve the level of protection should not be prescribed, as lessons from the past have demonstrated that each accident is different and large initial uncertainties exist in terms of the waste characterisation.

Regulatory guidance on how to meet existing and contingency regulations may need further development to be made applicable to an accident situation.

While protection objectives should be clear, the methods for showing compliance may best be left to guidance documents or later specification. Compliance demonstration may include a combination of measurements and assessments. For example, because of the potentially large uncertainties in waste characteristics, current safety requirements and waste acceptance criteria (WAC), where they are relevant to interim storage and disposal, should be taken into account as far as possible. If accident waste arising does

not fit comfortably within the existing framework, it should be determined how it is different and which, if any, changes or additional regulatory guidance may be needed. An early question to be addressed is whether new package types will be needed. A similar question may be asked in relation to transport regulations and packaging requirements.

At some stage, it will be necessary to develop derived standards that can be used by engineers in the design of containment systems, including specification of packages and waste forms. For example, a required decontamination level or derived environmental standard should be designed so that protection objectives are met. These definitions will help to define monitoring programmes used to support compliance demonstration.

Regulatory supervision is likely to be an iterative process, with iteration at each stage in decommissioning and waste management. Each stage can be supported by engagement with technologists and other stakeholders, as well as safety and environmental impact assessments and risk analyses.

To facilitate this process, licensing and other regulatory processes can be more effective if the activities of different regulatory bodies and policy organisations are co-ordinated through a joint regulatory co-ordination group. The participation of operators and other stakeholders in such a group has also been found to contribute to activities being more relevant and efficient, provided that a clear and transparent process is used within the operation of the group and that each organisation maintains its own separate responsibilities. Experience suggests that such exchanges can be relatively straightforward in planned situations, but that further consideration may be necessary following a major accident. Thus, interaction with stakeholders prior to an accident is important.

Any timeline for implementation of a decommissioning strategy needs to include time for regulatory approval at appropriate intervals or each stage of decommissioning. Early interaction between the regulator and operator is important for timely, safe and effective decommissioning, especially in the case of decommissioning after an accident.

The decommissioning policy, strategy and planning should be closely linked to, and developed jointly with, the waste management policy, strategy and planning. Initial decisions and planning for clean-up operations soon after an accident, without consideration of final disposal, can make final disposal more difficult and may create more waste than necessary. At the same time, for transparency reasons, it should be acknowledged that this may be a necessary outcome if urgent action is needed (for example to save lives) during the emergency phase.

Optimisation is a major feature of radiological protection and its regulation. It includes consideration of economic and social factors, which can raise difficult questions, such as:

- How should regulators include economic and social factors in their decision-making processes, without being or appearing to be involved in political issues?
- How can you practically separate the scientific and the social value judgements?
- How do regulators compare or balance the different short and long-term risks, for different groups of people, that are expected for different options? How are the results of that information used in regulatory decision making? And how are these results, based on regulatory considerations (i.e. there might be wider issues to consider), communicated to stakeholders?
- If we do not convert radiation doses into risks, how can a regulator, or anyone else, compare the radiological consequences with other human health consequences associated with legacy remediation options?

These questions raise their own challenges regarding the regulatory decision-making process, including in relation to:

- the development of consistent protection objectives and proportionate regulatory approaches for radioactive and other contaminants, for people and the environment;
- the corresponding development of consistent derived standards relevant to these protection objectives, and approaches for their assessment;
- the development and application of transparent methods to support decisions on choices between options, and maintain a balanced response to “all” risks;
- the improved communication of risks and uncertainties so that people affected can make informed decisions.

Experience suggests that these questions and challenges are useful to consider and can be supported by a comprehensive stakeholder engagement process. Experience also shows that a stepwise approach is useful. One early step is to identify the relevant stakeholders, some of which are outlined in the full report. Three key aspects of the stepwise process are:

- an inclusive radiation monitoring system;
- a health surveillance system;
- an education system on radiation/radiological exposure.

The policy should be aimed at reducing any kind of “victim” feeling and the transition from a risk compensation policy to compensation for actual damage. For example, it may be useful to consider how to compensate for the risk of a stochastic effect. Are subjective risks real risks? It seems reasonable to say that they are real risks since there can be “real” harm, but they are not the same as objective risks.

Experience demonstrates that the contribution of local community representatives through self-help protection is the “engine” of long-term recovery from nuclear accidents. The role of experts is to serve local community representatives and to facilitate the development of their ability to assess and manage their own situation. The following points should thus be taken into account:

- The radiological protection focus for stakeholder involvement during the accident recovery phase should be on objectives and the delivery of long-term technical support.
- Technical support can be very resource intensive. Therefore, an increase in resources is required in post-accident radioactive waste management.
- Trust is a necessary and central component of successful stakeholder involvement.
- A positive vision of their future will help individuals to choose to stay or to go.
- Individual decisions, whether to stay or to go, are all valid.

Chapters 4, 5 and 6: Waste description – Physical and chemical nature, radiological characterisation and waste classification/categorisation

The key issues for characterisation of accident facilities are to:

- Determine the extent to which the accident has affected the radiological, physical and chemical characteristics of the solid and liquid waste that has been or will be produced.

- Provide the ability to optimise clean-up operations and waste management strategies.
- Appropriately estimate the need for waste conditioning, storage and disposal facilities.

Before beginning extensive clean-up work, the theoretical parameters of a post-accident situation should be considered, to the extent possible. A key question that might be asked is: “What characteristics of the accident waste are such that they cannot fit into normal requirements and practices?”

Proceeding on arbitrarily conservative or optimistic assumptions may be counter-productive because the real situation will likely be different. Best estimates should be used, from a realistic range of possibilities, including the value of a characteristic that may lead to the need to adopt a different strategy, because for example it means some logistic or safety limit is exceeded. The same applies to predictive analyses and assessments. If a conservative approach is taken at every step, then results do not inform a balanced or optimised managed process but instead introduce a bias towards the most pessimistic assumptions without a clear understanding of the implications.

Emphasis must be placed on having the good characterisation information necessary for facilities, infrastructure and land. Failure to do this presents real risks of a need to redo work at higher costs and, ultimately, adding delays. The problem is that any delay resulting from the need to understand the waste in every detail may itself introduce delays and inefficiencies. There are no rules that say with complete certainty that characterisation has been fully sufficient. Hence the need for an iterative process that in its final stage includes confirmation of compliance with protection and other objectives.

Physical characterisation

In the early phases of clean-up, a centralised, high-priority effort is needed to provide data on actual physical conditions. Visual observations are essential to understanding and efficiency. For example, specific methods are needed to locate and quantify the fuel when significant quantities are displaced from the original core region. It is then necessary to estimate the volumes of each type of solid and liquid waste that will arise, and the times at which they are expected to arise.

Radiological characterisation

A key lesson in the area of radiological characterisation is that once an accident occurs, there is typically insufficient capacity to address the waste characterisation needs. The ability to radiologically characterise materials should not be the limiting factor that controls the rate of decommissioning activities. Acquiring sufficient radiological characterisation data is likely the most important short-term challenge following an accident. Typically, there is a need to increase the capacity of analytical laboratories and train staff to operate them, and a related need to develop and optimise analytical methodologies relevant to the waste arising.

Radiological characterisation plans should be developed for implementation in the event of a future accident. The details cannot be planned in advance, but the plan can set out who is responsible to act so as to resolve these issues:

- Sampling plans, which specify the numbers, types and locations of samples to be analysed, should be developed and justified. Such plans will be iterative, and will develop as understanding of radionuclide concentrations and their spatial distribution improves. At the beginning of the project, such plans are likely to involve combinations of expert judgement and statistical considerations (for example, based on the US Environmental Protection Agency’s [EPA] data quality objectives [DQO] approach).

- Use of radionuclide vector “fingerprint” data can reduce the number of samples and radionuclide measurements on those samples. However, it is likely that fingerprints for materials developed prior to the accident will be inaccurate or even irrelevant. In that case new fingerprints would be required.
- As characterisation data are obtained, it may be appropriate to also include geostatistical approaches to optimise future data collection.
- New technologies giving direct cartographies and thus avoiding sampling should be used more widely.
- Sampling and characterisation plans could involve calling upon resources from other countries according to pre-accident arrangements.
- Sufficient characterisation equipment should be made available, both in off-site analytical testing laboratories and on the accident site to meet the needs of the sampling and characterisation plan.
- In the short term after an accident, it will be necessary to sort waste (for example, to consign it to appropriate storage facilities) on the basis of quick, simple, easily measurable parameters such as surface dose rate. As soon as possible after the accident, routine on-site analyses for easy-to-measure radionuclides such as Cs-137 should be started.
- In addition, suitable sample preparation and analysis methods should be identified for all of the significant material types that will require characterisation (concrete, metals, soil, vegetation and possibly agricultural produce).

More detailed advice on radiological characterisation is provided in the full report.

Physico-chemical characterisation

In addition to radiological characteristics, the physico-chemical nature of the waste is also needed for treatment and storage purposes to determine:

- the “materials inventory” for the waste that will be produced from the clean-up programme;
- whether the waste contains any hazardous non-radioactive substances such as toxic metals or asbestos, and assess the implications of these hazardous substances on waste handling, conditioning, interim storage and disposal;
- whether the waste contains chemical complexing or chelating agents, and assess the impacts of such complexing or chelating agents on waste conditioning and disposal.

Ideally, a holistic approach to the assessment of radionuclides and any hazardous non-radioactive materials should be applied, such that consistent assumptions are employed in assessments and consistent criteria used in the evaluation of risk. International guidance on how to do this could be helpful.

Waste classification/categorisation

Practice shows that national schemes for waste classification and/or categorisation reflect nationally relevant factors, particularly the waste that is expected to be produced from normal operations. In the case of a major accident, it should be determined whether the existing classification/categorisation scheme can be applied to larger volumes of accident waste. There may be a need to recognise or define different radioactive waste (RW) categories from those adopted in planned situations. These may be based on modified protection objectives, taking into account the wider needs and interests of those affected. Any new proposed scheme can benefit by taking into account relevant

international practice and lessons learnt at the sites of previous major accidents. The scheme should incorporate all waste arising and include classification of waste which does not need to be managed or regulated as radioactive waste. In other words, it should account for other hazardous features, so as to avoid planning, regulatory and safety management contradictions. If it is decided to revise the waste classification schemes, it will be important to record the reasoning for the changes and keep the memory of the specification for accountability.

Ideally, the classification scheme adopted should support all aspects of management in a holistic manner, leading to and not foreclosing on options for final disposal. Although a new approach to management may be needed, insights from experienced senior technical advisors are invaluable. While it may be difficult to integrate with the workforce, their third-party review is essential in fields where new approaches or techniques are applied. Details on the regulatory basis are added only when they can be justified.

Most waste categorisation/classification schemes developed in the past were devised to address technical issues and were necessarily expressed in technical terms. They do not readily indicate the scale of hazard associated with the waste and therefore do not help explain to stakeholders the significance of the hazards. It has been suggested therefore that any new classification scheme for accident waste include a component that relates the hazard of each class of waste to another readily understandable or commonly encountered hazard. Ideally this would include consideration of chemical as well as radiological hazards within a single coherent and proportionate approach to risk management.

The proposed radiological characterisation should enable waste to be assigned to existing waste categories, but it will also be important to ensure that sufficient characterisation data are collected to enable waste to be sentenced against alternative categories if this becomes necessary.

Iterative process

An important lesson is that it should be clarified at an early stage who will be responsible for the development of acceptance criteria (waste acceptance criteria or WAC) for accident waste for storage and disposal, and what will be the procedure for their development. Without this information, it is difficult to specify which information is needed about the waste and hence, it is difficult to specify the characterisation programme. A degree of iteration in the process should be expected.

The details of WAC and arrangements for managing fragments of fissile material are quite specific to the conditions and circumstances of the accident. It highlights the need to put in place the regulatory basis for waste disposal in parallel with waste characterisation work (see above), and have a corresponding categorisation system that facilitates meeting protection objectives and other needs. Without this regulatory basis, it is difficult to know what characteristics have to be investigated. To resolve this problem, it is recommended to start with setting the objectives of the decommissioning programme (e.g. to do things safely) and proceed iteratively from there.

Chapters 7 and 8: Conditioning, storage and disposal

Waste conditioning, decontamination and reduction

As a contingency in case of a major accident, national regulatory guidance should be available that provides advice on how to deal with the increased volume and radiological and chemical characteristics of post-accident waste and on the flexibility needed to meet regulatory standards, rather than specific criteria.

In the immediate aftermath of an accident, actions will be required to stabilise the facility and prevent the spread of a mobile source term to the environment. These actions could lead to the generation of waste that contains high concentrations of fission products and/or quantities of fuel materials which will likely not comply with existing waste acceptance criteria. Lessons learnt from prior accident situations or similar waste conditioning activities such as decommissioning and legacy waste recovery at reprocessing facilities should be available to the affected facility to guide their emergency response actions.

Following the reaction to the immediate emergency, as described above, the choice of waste conditioning processes is important in the overall road map of post-accident waste management. These decisions will need to consider existing regulations on waste disposal and drive actions on characterisation, treatment, conditioning, storage, optimisation of waste management scenarios and final disposal of the waste.

Destination (storage/disposal)

The largest volume of waste arising from an accident will probably be compatible with existing disposal criteria. However, the volume of waste may overwhelm existing disposal site capacity, and therefore early consideration of volume reduction techniques is essential.

A smaller volume of waste may fall outside existing waste acceptance criteria, and thus the management of the damaged facility working in co-operation with governmental and research organisations needs to develop a plan for the disposal of this waste.

Damaged fuel and fuel debris are a special case as the prevention of a criticality event at any point during the accident clean-up process is required. Therefore, application of existing requirements for maintaining spent fuel subcritical should also be applied to damaged fuel. Thus, the design of containers to house damaged fuel must meet the same reactivity criteria as undamaged spent fuel.

Overall conclusions

The case studies in the report present substantial information on the history of accident site management and lessons learnt, leading to many potentially helpful recommendations. Material provided also includes information on:

- state-of-the-art techniques and experiences with waste characterisation and classification, including application after major accidents;
- regulatory supervision: regulations, regulatory guidance and regulatory procedures, e.g. review of safety cases;
- application of international recommendations, standards and guidance.

Every accident is different. The details of any post-accident (after emergency) scenario are unpredictable and specific to the prevailing circumstances. Responding to them requires elements that are not within the usual experience of conventional utility and service management organisations. Managing decommissioning and radioactive waste after a major accident may also require a different approach from that used following normal planned operations.

Centralised authority and stakeholder involvement

There is a need for a centralised authority to manage the situation, for example, a high-level governmental commission, so as to co-ordinate and oversee the planning and implementation of effective measures. Government authorities, the industry and

research institutions must work co-operatively to plan and implement these measures. This authority will need to develop a comprehensive strategy with clear objectives to manage the situation, taking into account the interests of a wide range of stakeholders. Effective stakeholder engagement processes are needed to identify these interests.

Implementation strategy

A plan is needed to implement the strategy through a series of tasks designed to meet the stated objectives, identifying who is responsible for implementing each task and providing the powers and resources necessary to those with responsibility for implementation.

A major component of this strategy should be connected to the establishment of a regulatory framework for decommissioning and radioactive waste management. This should be based as far as possible on the existing framework for these activities, but specifically modified to account for the special factors linked to the prevailing circumstances arising from the accident, as identified through waste characterisation and other processes.

Special factors include the need to set appropriate reference levels as well as derived standards and monitoring procedures, the application of which would result in meeting the reference levels and demonstrating an ability to comply with them.

A heavily project-focused approach is more effective than one focused on a large functional organisation of engineers and designers responsible for small areas involving several projects.

Although redundancy in organisational functions is expensive and difficult to manage, some degree of redundancy is prudent to ensure that all options and potential problems can be considered.

There is likely to be a need for iteration of the strategy with more detail added at each stage, taking account information obtained from the previous stage including radioactive waste characterisation data. Responsibilities for the implementation and resourcing of tasks at each stage may need to be updated. In early stages, it may be useful to pursue flexible/parallel approaches. In any case, a careful step-by-step approach is strongly advised, so as to reduce the chance of creating legacies requiring future management.

At the same time, it has been noted that excessive caution may delay appropriate timing of decisions. Examples include delay of return to normal land use, even though it would be safe to do so, or delay in the introduction of appropriate restrictions, resulting in extended continuation of risky conditions, as well as potential added costs. This problem should be acknowledged, alongside the need for balance, which should be achieved with stakeholder engagement.

In developing the iterative strategy, it is important to leave time to obtain regulatory approval. Public access to land and normalisation of land use is urgent, providing many hard to measure benefits to those who normally occupy the land. The contamination levels can be expected to be relatively low off-site so that remediation work is very extensive but not complex from a technical and safety point of view. However, once the emergency is declared over, decommissioning of the damaged building and remediation of the nuclear site itself is not so urgent. It may also be much more hazardous and present further risks of repeat accidents. It is therefore necessary to anticipate the need to take the time to do this work, as has been the case at TMI-2, ChNPP and Windscale Pile.

Optimisation

Optimisation is an important aspect of radiological protection and is best done when taking into account social and economic factors, and not just radiological factors, for

example meeting reference levels. Again, the process should be supported by stakeholder engagement. Solid waste minimisation, as has been recommended, can be achieved by discharging more waste to air and water, for instance through incineration or dissolution, or creating higher-level waste that is not suitable for shallow land burial. It is not entirely evident that such discharge is the optimum management method, and so the choice would need to be supported by a relevantly structured assessment. More generally, it has been noted that the minimisation of one detrimental impact is always likely to result in something else detrimental not being minimised. Hence the need for a holistic view of optimisation, both as developed in radiological protection and as would be more widely understood by stakeholders.

Storage and disposal

In addition to large quantities of fuel debris, the remediation and decommissioning response to an accident of the Fukushima Daiichi NPP type is likely to generate radioactive waste that exceeds limits for near-surface disposal or intermediate-depth disposal. This waste needs to be appropriately stored and stabilised until a final disposal solution is developed.

The large quantity of waste created by an accident may exceed existing radioactive waste disposal capacity or be in a waste class for which a disposal solution is not currently available. It may be necessary to create interim stores, but they should be designed taking into account that final disposal will be needed in due course, and may need to remain effective for extended periods while sites for final disposal are identified and licensed. The accident site however, may not be the location to site this interim storage facility.

Safety analyses

Safety analyses and radiological and environmental impact assessments are necessary to support identification of priorities, identify feasible management options and select preferred options from feasible alternatives. This process needs to be technically underpinned, but must be informed by stakeholder engagement, particularly as regards local conditions, but also in terms of ensuring that the assessments address issues of interest to stakeholders.

These analyses must, to the extent possible, be based on existing regulations and regulatory guidance. Only in exceptional circumstances and based on a safety case that demonstrates compliance with the safety basis of the applicable regulation(s) should exemptions from these criteria be permitted.

Thus, the design and content of these analyses and assessments should be specific to the purposes of the assessments, including the interest of the intended audience for each analysis or assessment.

International co-operation

Further development of plans for international co-operation in the event of a major accident would be useful as would further guidance on the application of international recommendations, standards and guidance in the post-emergency phase of a major nuclear accident.

Pre-planning guidance on decommissioning and radioactive waste management should consider:

- what can be planned in advance;
- what cannot be planned until the parameters of the accident are understood;

- the scope for sharing characterisation resources, staff and equipment, both nationally and internationally.

Additional guidance in other areas could also be useful, including on:

- the transition from emergency response to normal radiation exposure regulation;
- stakeholder engagement, with emphasis on later stages of recovery;
- communication processes;
- how to address chemicals alongside the radiological risks.

Introduction

Following the accident at the Fukushima Daiichi nuclear power plant (NPP) in March 2011, different types of post-accident radioactive waste were generated, for example from on-site decontamination activities, the management of contaminated water, decommissioning work on the four reactors and from the hydrogen explosions that occurred. The radioactive waste resulting from the accident has different properties compared with the waste generated by nuclear power plants operating under normal conditions. Specific management methods or strategies will therefore be needed to manage the post-accident waste.

After the Fukushima Daiichi accident, the NEA Radioactive Waste Management Committee (RWMC) underlined the importance of including post-accident waste management and co-operation on decommissioning techniques for the Fukushima Daiichi NPP in the strategic areas of the NEA Programme of Work as it relates to radioactive waste management.

At the plenary meeting of the RWMC held in March 2014, the Ministry of Economy, Trade and Industry (METI) of Japan had highlighted the difficulties in managing the post-accident radioactive waste. Major difficulties encountered by METI are waste categorisation and classification, due to the difficulties in accessing samples for measurements and the vast amount of waste and contaminated material that needs to be managed. The RWMC decided to establish the Expert Group on Fukushima Waste Management and Decommissioning R&D (EGFWMD) to evaluate the management of post-accident waste. The expert group focuses on technical issues on waste management, such as radiological characterisation and categorisation of post-accident radioactive waste and contaminated materials, but also on social issues such as stakeholder engagement and interactions between the regulator and implementer. This report provides advice on post-accident waste management, particularly to research and development (R&D) institutions in Japan on their overall strategy for managing the waste generated on-site by the accident. It also provides information on strategies to be implemented in the case of an unplanned, unexpected accident in the future.

Characterisation and categorisation of post-accident radioactive waste are among the most difficult challenges for waste management at the Fukushima Daiichi NPP in the near term. Therefore, a strategic approach should be developed to manage the complex characterisation process of the waste, which includes a sampling and characterisation plan based on statistical approaches, calculation methods, and a review and evaluation process for the data obtained. International knowledge and experience in managing legacy waste, accident waste and other pertinent examples will be valuable to help Japan address challenges in managing the Fukushima Daiichi waste.

This report provides information on waste management and remediation of contaminated areas, as well as on R&D activities, by identifying the gaps between experiences in past accidents or contaminations and current activities being undertaken at the Fukushima Daiichi site:

- Chapter 1 provides general descriptions and a short introduction to nuclear accidents or radiological contaminations; for instance the Chernobyl NPP accident, the Three Mile Island Unit 2 accident and the Windscale fire accident.

- Chapter 2 provides experiences on regulator-implementer interaction in both normal and abnormal situations, including after a nuclear accident. Chapter 3 provides experiences on stakeholder involvement after accidents. These two chapters focus on human aspects after an accident and provide recommendations on how to improve communication between stakeholders so as to resolve issues arising after unexpected nuclear accidents.
- Chapters 4, 5 and 6 provide information on technical issues related to waste management after accidents. Chapter 4 focuses on the physical and chemical nature of the waste, Chapter 5 on radiological characterisation, and Chapter 6 on waste classification and categorisation. The persons involved in waste management after an accident should address these issues as soon as possible after the accident.
- Chapters 7 and 8 also focus on technical issues but with a long-term perspective of the waste direction in the future. Chapter 7 relates experience on waste conditioning, reduction and decontamination, and Chapter 8 provides information on the destination of radioactive waste storage and disposal.

1. General description of case studies

1.1. Chernobyl nuclear power plant accident

Chernobyl accident and “Chernobyl waste”¹ in the 30-km exclusion zone²

The Chernobyl accident occurred on 26 April 1986. Because of an accident at unit 4 of the Chernobyl nuclear power plant (NPP) the reactor core was completely destroyed and the systems related to safety were also entirely destroyed. Large amounts of radioactive activity were released and the surrounding area was contaminated, resulting in high levels of exposure doses and fragments of nuclear fuel and graphite around the destroyed unit.

Regarding the mitigation of the accident consequences, a number of organisational and technical decisions were made and implemented:

- the creation of a special governmental commission for the brain-storming decision-making process; with full power, clear responsibility, availability of resources, scientific support and immediate involvement of different specialists, if needed;
- the overall decontamination of the industrial site and surrounding residential areas and roads, with the main goal of decreasing the exposure doses, to allow workers to continue their activities at units 1, 2 and 3, and to allow re-evacuation of the population (the establishment of a “30-km exclusion zone” made this last point unnecessary);
- arrangement of facilities for decontamination, special treatment of trucks and personnel involved in the accident mitigation;
- collection and removal of radioactive waste (RW); organisation of “temporary RW storage,”³ designing and construction of RW facilities; and reduction of environmental risks in the surrounding area.

From the mitigation of the Chernobyl accident consequences, we can derive the following experiences and lessons learnt:

- There was no preparedness for this type of accident, as nobody believed that such an accident was possible.
- There was a lack of a prompt monitoring system for emergency situations; such a system could have been very helpful in the decision-making process.

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1. “Chernobyl waste” means the radioactive waste originating from the Chernobyl accident.
 2. The 30-km exclusion zone designates the territory where the population was evacuated in 1986. This exclusion zone includes land that has been removed from normal economic usage.
 3. “Temporary RW storage facilities” refers to sites for temporary RW localisation.

- Decontamination to effectively zero levels was not always effective because of secondary radioactive contamination with wind flows and adverse weather conditions.
- There was a clear need for a centralised management of the situation, like a high-level governmental commission, to take effective measures.
- The need also arose for the creation of “RW disposal facilities”⁴ and “temporary RW storage facilities”. As there was no experience in managing large amounts of emergency radioactive waste, storage and disposal for “Chernobyl waste” were conducted in extreme conditions, without adequate waste isolation technology and classification and registering of waste (its amount and activity); neither was the environmental impact of storage and disposal sites considered; “Chernobyl waste” was stored and disposed of under conditions that do not fully comply with safety requirements; and therefore it needs to be re-disposed.

Because of the lack of infrastructure for the treatment of the large amount of “emergency RW”, the decision was taken to place the waste in unorganised trenches, so-called “temporary RW localisation points”. Such localisation points were set up nearby the Chernobyl NPP and they lacked either the design documentation or the records on characteristics of the waste. It is the lack of records on the “RW temporary localisation points” which has created many problems, and consequently, have delayed decisions on further waste treatment.

In summary, 90% of the radioactive waste in Ukraine is “Chernobyl waste”. This waste contains varying compositions of radionuclides, as well as long-lived nuclides. It is located or has been placed mostly in the exclusion zone and, because of the above-mentioned issues, it needs to be re-disposed of. For this purpose, the following should be taken into account:

- a there is a need for a great deal of pre-operational work to be done with the “Chernobyl waste” before disposal;
- there is a need for the development and implementation of projects for RW retrieval, removal, treatment – so-called re-disposal;
- there is a need to prioritise the retrieval of RW from the points that most influence the environment because of flooding and barrier faults.

Safety analyses and environmental impact assessments are necessary for a decision to be taken on further brining of RW storage facilities in compliance with radiation safety requirements and optimisation of RW conservation or re-disposal costs. Particular administrative measures, risks and countermeasure assessments shall be developed to prevent incidents during the construction and operation of new RW management facilities. Consequently, assessment methodologies for risks, emergency response plans and countermeasure designs should be updated and approved. The introduction of an

4. “RW disposal facilities” mean the facilities for RW disposal created between two to three years after the Chernobyl accident: “Pidlisny”, “The 3rd turn of the Chernobyl NPP” and “Buryakivka” Only the RW near-surface disposal facility “Buryakivka” is under operation until now. The RW disposal site “Pidlisnyi” has been in operation since December 1986 through 1988. RW of an exposure dose rate (EDR) up to 50 R/h was accepted by the RW disposal site “Pidlisnyi”, then based on a Governmental Committee Decree, RW of EDR up to 250 R/h was received. The overall amount of RW is 11 000 m³, and the total activity is assessed as 2.59E+15 Bq. The RW disposal site “Chernobyl NPP Stage III”, which was in operation until the end of 1986, was constructed for RW with EDR up to 1 R/h, but waste with higher EDR was accepted there as well. The RW disposal site contains 26 000 m³ of low- and medium-level waste, including long-lived waste, of a total activity of 3.43E+14 Bq.

integrated monitoring system for RW storage facilities, an environment within RW storage facility areas, and particularly a general hydrological and hydrogeological situation within the exclusion zone, is vital for taking informed decisions.

Re-disposal is a possible way of fundamentally changing the situation to preclude the release of radionuclides from temporary RW localisation sites. However, not all the trenches are filled with RW. Some of them may be released from regulatory control because of corresponding levels having been reached.

In Ukraine, there are two options for disposal: near-surface for low- and intermediate-level short-lived RW and geological disposal for long-lived RW, high-level RW and fuel-containing materials from the shelter object. Taking into account the practical experience of RW management in the 30-km exclusion zone and lessons learnt from the management of large amounts of RW from accidental origins, it should be concluded that an improvement of Ukrainian legislation (“safety requirements”) is also needed, taking into account the peculiarity of “Chernobyl waste” and the peculiarity of disposal of such waste within the 30-km exclusion zone (safety analysis of long-term safety, dose limits). In addition, improvement of RW classification is needed so as to consider the category of “very low-level waste”.

RW Management at the Chernobyl NPP site after the accident

Before the accident, the RW management system at the Chernobyl NPP included: treatment operations related to liquid RW volumes decreasing (evaporation), sorting of solid RW (based on activity level – low, intermediate and high level of activity) and storing of liquid and solid RW in temporary storage facilities. The “retrievability” of RW from existing storages is a great challenge today. RW in present conditions cannot be disposed of and should be retrieved, treated and conditioned before its disposal in near-surface repositories, or safely stored in a new storage facility.

The Chernobyl accident in April 1986 strongly influenced the two radioactive waste management areas mentioned above. It made the soil radioactive and engendered significant portions of alpha emitter content in the operational waste. The Chernobyl site presents a great variety of the radioactive waste types:

- soil contaminated during the Chernobyl accident;
- the Chernobyl NPP operational liquid and solid radioactive waste, including RW from the mitigation period of accident consequences;
- shelter object solid and liquid radioactive waste and fuel-containing masses.

Some years after the accident, it was decided to develop a document called the “Integrated Radioactive Waste Management Programme for the Units 1, 2, 3 Chernobyl NPP and for the Transformation of the Shelter Object into an Ecologically Safe System”. The main purpose of this document was not only to describe the activity and characteristics of collected RW and RW facilities, but also to analyse the needs to improve the situation with treatment and conditioning of existing and future RW streams. All waste streams on the Chernobyl NPP site shall be managed within the frame of an integrated RW management system, taking into account acceptance of conditioned RW for final disposal in near-surface repositories or storing of RW that cannot be disposed of in near-surface repositories.

Based on the results and conclusions of much research and many investigations relating findings or best solutions in terms of RW management at the Chernobyl NPP site, the decision regarding the creation of a set of new RW facilities was approved. These include:

- plants for solid and liquid RW retrieval, characterisation, treatment and conditioning at the Chernobyl NPP site;

- a storage facility for high-level waste and intermediate-level long-lived waste at the Chernobyl NPP site;
- near-surface disposal repository in the 30-km exclusion zone (this repository was constructed especially for conditioned waste from the Chernobyl NPP site).

One of the most critical tasks to be undertaken today by the Chernobyl NPP is the successful commissioning of facilities for liquid and solid waste treatment and conditioning; those facilities would handle the waste accumulated through Chernobyl NPP operation and waste that would be generated from the decommissioning of the Chernobyl NPP.

In RW management facilities, liquid and solid RW would be removed from the existing storages, then be processed by specific technologies and the final product of processing, which is cemented RW in special containers, would be transported for disposal at the near-surface repository.

The “Integrated Radioactive Waste Management Programme for Chernobyl NPP” is a living document that should be updated regularly by the Chernobyl NPP taking into account lessons learnt during the past, the necessity of solving new problems, , changes in regulations and others aspects (for example needs in additional RW facilities).

Shelter object (destroyed Chernobyl NPP unit 4) – Existing situation and plans for the future

Since the severe beyond-design-basis accident at Chernobyl NPP unit 4, this particular NPP has been permanently under focus from the Ukrainian government, the public and the international community. The accident is primarily associated with the shelter object, which is the ruined unit 4 covered by new metallic and concrete constructions.

Immediately after the accident, different materials were dropped inside the destroyed unit to provide fuel cooling (lead) to prevent a self-sustained chain reaction (boron carbide), to cease graphite burning (dolomite) and to filtrate the fission products release (sand and clay). Then, by April and May 1986, about 15 000 t of material were dropped. An underground layer of the local zone around shelter had 15 000 m³ of RW (contaminated soil, concrete pieces and slabs, metal structures, debris). From 400 000 to 1 740 000 m³ of RW are located in the shelter object and at its site. At the beginning of 2005, their total activity was about 4.1E+17 Bq. Over 10% of the total amount of RW is high-level waste (HLW), a great amount of which are concrete, metal structures and equipment or materials of backfill of the reactor. Over 2 800 t of HLW are fuel-containing materials, fragments of the reactor active zone, reactor graphite and fuel dust.

After the shelter object was constructed in November 1986, there was a big challenge in Ukraine to find appropriate solutions for such problematic questions as “How to regulate such a unique facility (as a nuclear facility or temporary storage for non-organised RW)?”; “How to classify the activity related to the shelter object?”; “What type of licence can be issued for the shelter object?” (“decommissioning of destroyed nuclear unit” or “operation of the shelter object?”) and “Do we need to develop separate laws and regulations for such a unique facility?”.

After many investigations on the technical level and discussions on the political level it was decided that:

- The shelter object for the remains of unit 4 is “not organised storage for not organised waste”.
- The activity to be performed on the shelter object shall be “transformation into ecologically safe system” only.
- The licence issued by the regulatory body was for “operation of shelter object”; a number of conditions were included in this licence – for existing and future

activity related “transformation into ecologically safe system”. The number of separate permissions to be obtained by Chernobyl NPP for different projects was also included in the licence.

The most difficult questions for the regulatory body were mainly about i) the necessity of developing specific regulations for the shelter object and ii) applying the safety requirements of existing regulations for nuclear and RW facilities to the shelter object. The detailed explanation of the “Ukrainian approach” is set out below.

The “national strategy related transformation of shelter object into the ecologically safe system” was approved by the Ukrainian government. This strategy includes three main stages:

- stabilisation of unstable construction and components of shelter object;
- new safe confinement (NSC) designing, construction and commissioning;
- extraction of radioactive waste and fuel-containing masses and safely disposing and storing them.

Since 1998, the so-called Shelter Implementation Plan (SIP) has been carried out. The plan was developed by an international expert group and approved by the Great Seven countries and the Ukrainian government. The SIP provided for a set of both short-term measures to maintain the shelter safety level and long-term ones, which are aimed at shelter transformation to an environmentally safe system. The SIP consists of two phases: preliminary (phase 1) and main (phase 2). The SIP phase 2 is today achieving a good pace of work.

This SIP includes 22 tasks, consisting of both long-term and short-term measures. Long-term measures primarily concern the shelter object isolation by the NSC and safe extraction at a later time of highly radioactive materials under confinement. Short-term measures focus on stabilising and improving the safety of existing objects by strengthening existing buildings, additional dust suppression, preventing the possibility of reactivity emergency and temporary water management.

SIP covers the first and second stages of the national strategy related transformation of shelter object into the ecologically safe system (stabilisation and NSC construction), but for the third stage, SIP envisages only some investigations-related strategy of RW and fuel-containing mass monitoring, as well as some approaches regarding the technology for their extraction.

Nevertheless, the main task is to transform the shelter object into a safe system, which means to extract radioactive waste and fuel-containing masses and safely dispose of them. During close to 20 years of examination, a great deal of work related-radiation and heat measurements, sampling and later destructive and non-destructive measurements, as well as calculations were done. Nevertheless, there is no final decision on the extraction technology for these most dangerous materials due to a lack of information. The Chernobyl NPP developed and the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) approved a stepwise strategy, which will result in a final decision on the extraction technology. The idea is to permanently support the fuel-containing mass (FCM) safety before final disposal in a deep geological repository.

The Shelter object integrated monitoring system and the result of multiple SIP tasks will allow for the development of some analytical models for FCM behaviour. Existing and future emergency systems can effectively support the under critical state of these materials, and future NSC will fix the optimum temperature and humidity for FCM during the construction period of the NSC and deep geological repository. The following systems are included in its structure:

- nuclear and radiation safety monitoring system;

- seismic monitoring system;
- monitoring system of the building construction;
- operation support systems: ventilation system; water supply system; sewerage system (including, liquid radioactive waste management); electro supply system;
- technological systems of radioactive waste and fuel-containing materials management.

In addition, fire safety and physical protection systems were created, and communications and TV networks will be mounted. The crane equipment will be mounted for unstable construction dismantling.

New safe confinement is a major SIP project. Since 1992, intensive work, including under international programmes, on shelter object transformation into an ecologically safe system has been conducted. The basic conclusion of this long-term research is: significant shelter object hazard decrease can be achieved as a result of new protective encasement construction over the object NSC.

The basic requirements for confinement are formulated in the law of Ukraine “on general principles of the further Chernobyl operation and decommissioning and destroyed fourth power unit of this NPP transformation into ecologically safe system”: Confinement is a protective construction that includes a complex of the technological equipment for removal from the destroyed fourth power unit of Chernobyl NPP materials containing nuclear fuel, radioactive waste management and other systems intended for the realisation of activity on the transformation of this power unit in an ecologically safe system and the safety of workers, the public and the environment.

1.2. Three Mile Island nuclear power plant accident

Introduction

Shortly after 4:00 a.m. Wednesday, 28 March 1979, with the nuclear plant operating at 97% power, a series of feedwater system pumps supplying water to the steam generators shutdown at the Three Mile Island NPP. This led to a cascading series of events that culminated in the partial meltdown of the Three Mile Island 2 (TMI-2) reactor core.

Shortly after the accident, the Richland, Washington and Barnwell commercial low-level burial sites were closed to any TMI-2 accident-related waste because of concerns about the volumes of waste that might be produced. Later in 1979, the site in Richland was reopened to TMI-2 waste. The Barnwell site did not reopen to TMI-2 waste until 1987. The agreement, signed by the City of Lancaster, the US Nuclear Regulatory Commission (NRC), and the Licensees of TMI-2, placed significant restrictions on the discharge of “Accident Generated Water”, which was defined as:

- water that existed in the TMI-2 auxiliary, fuel handling and containment buildings, including the primary system, as of 16 October 1979 with the exception of water which, as a result of decontamination operations, became commingled with non-accident-generated water such that the commingled water had a tritium content of 0.025 Ci/ml or less before processing;
- water that had a total activity of greater than 1 Ci/ml prior to processing, except where such water was originally non-accident water and became contaminated by use in clean-up;
- water that contained greater than 0.025 Ci/ml of tritium before processing.

Further, a large amount of the waste generated at TMI-2 exceeded commercial burial criteria for form and content and were thus even more difficult to dispose of. There was no disposal facility for this higher level or “abnormal waste”. In addition, much of the waste was not comparable to that produced at an operating power plant. This waste contained a high concentration of fission products or small quantities of fuel materials.

As part of the solution, the US Department of Energy (DOE) and the NRC signed a Memorandum of Understanding (MOU) in July 1981 to ensure the TMI site did not become a long-term waste disposal facility.

Because of issues with radioactive waste disposal, the cost of disposal and limited on-site storage, related to radioactive waste minimisation practices were adopted early in the clean-up process and continually refined. These practices included:

- reducing waste at the source;
- only taking into contaminated areas the tools needed to do the job;
- storage of contaminated tools;
- performing maintenance on tooling and equipment inside contaminated areas;
- recycling water to the extent practical;
- on-site waste reduction.

To support on-site waste reduction, a waste handling and packaging facility and a respirator cleaning and laundry facility were constructed.

- The Waste Handling and Packaging Facility (WHPF) began operation in February 1987. This 2 500 ft² (230m²) facility was designed and built to provide an environment for the decontamination of materials for unconditional release, volume reduction, sorting of materials and the compaction of materials in drums. The WHPF was justified by cost savings resulting from the commercial release of decontaminated material, improved packaging efficiency for non-compacted material in boxes and the improved packaging efficiency for compacted material in drums. In terms of volume reduction, the WHPF improved packaging efficiency by 25-30% and significant quantities of metal and other items were released for commercial scrap or reuse on-site.
- After the accident, a temporary contaminated laundry and respirator cleaning complex was set up to launder contaminated protective clothing and to decontaminate, clean and sanitise respirators. The complex operated from shortly after the accident until early 1985, when the permanent laundry and respirator cleaning facility was completed and became operational.

On-site waste storage

Two types of solid radioactive waste needed immediate attention: mildly contaminated trash, (solid low-level waste) and spent resins, and thus temporary on-site storage facilities were needed.

TMI-2 was not dismantled, but the quantity of lower-activity radioactive waste was difficult to control because the clean-up had to proceed as quickly as possible. Consequently, the project team focused on controlling the final volume to be shipped. This was done by decontaminating and reusing equipment or material whenever possible, solidifying waste when necessary and boxing or compacting the rest. Initially this activity was carried out inside the auxiliary and reactor buildings.

However, insufficient space was available for staging and destaging this waste, and containers and little equipment was available to dismantle decontaminate or temporarily store the tools, fixtures and large equipment needed for large-scale clean-up operations.

Solid low-level waste

The first project was the conversion of a paint shed into an interim facility for storage of drums and boxes containing low-level waste. This building had a usable floor space of approximately 140 m². All of the anti-contamination clothing and decontamination and contamination control material were stored in this facility until the waste could be shipped. The fenced in yard around the paint shed was also used to stage contaminated equipment. Use of the paint shed was limited after completion of other staging areas.

The interim solid waste staging facility or “carport” was built as a long-term solution to low-level waste storage prior to shipment for disposal. The facility consisted of a concrete pad that was approximately 870 m² and was protected by roof and aluminium sidewalls. A partial-height concrete block wall enclosed an area where higher-activity waste was stored. Six sumps collected any water in-leakage. Radwaste stored in this area consisted of solid low-level waste contained in shipping boxes and drums. This facility acted as the primary low-level waste storage area.

Spent resins

As a first response to the need for an interim spent resin storage area, liner storage modules were built. The area consisted of 14 large-diameter drainage culverts welded to a steel endplate. The culverts were placed vertically in the ground and the area was back-filled. A large three-foot thick concrete plug covered the top of each storage cell. This area was used to store EPICOR-II resin vessels but was taken out of service in 1980, shortly after the solid waste staging facility began operation. Only two of the cells were ever used to store radioactive resins; the others had held new resin liners in storage prior to their initial installation in the EPICOR system.

The solid waste staging facility was an engineered storage facility constructed as a long-term solution to spent resin liner storage. The facility consisted of two (2) modules containing 60 cells each. Each rectangular concrete module was approximately 50 feet (15 m) wide by 90 feet (27 m) long by 19 feet (6 m) high. The module base and walls were three feet (1 m) thick to ensure the surface radiation levels remained below 50 Sv/h. The 6 feet (2 m) diameter by 12 feet (4 m) high cells consisted of concrete-shielded, galvanised, corrugated-steel cylinders with welded steel base plates. A drain line from each cell led to a common sump. A three-foot (1 m) thick concrete lid covered each cell. EPICOR-II spent resin vessels were stored in this facility.

Core accountability

In addition to waste disposal issues, which are discussed further in later sections of this report, another significant issue that affected TMI-2 was accounting for the special nuclear material (SNM) inventory at TMI-2. Because of the accident, standard SNM accountability techniques could not be used.

The core debris removed from the TMI-2 facility was loaded into special canisters for shipment to the DOE Idaho National Laboratory. Each shipment was accompanied by a “Nuclear Material Transaction Report” (DOE/NRC Form 741), which recorded the net weight of the contents of each canister. Fuel accountability by the normal method (i.e. accounting for individual fuel assemblies) was not possible. Since the canisters were filled with a mixture of SNM, other materials and water, there was no feasible method to determine the exact SNM content in each canister. A statement to that effect was included on each DOE/NRC Form 741.

In October 1985, GPU Nuclear, DOE and the NRC entered into an agreement that final SNM accountability for TMI-2 would be performed after defueling was completed. The accountability would be based on a thorough post-defueling survey of TMI-2, which would quantify the amount of residual SNM in plant systems and components.

The entire TMI-2 plant was reviewed to determine where SNM could have been deposited as a result of the 1979 accident and subsequent recovery activities. Each area was classified into one of three categories:

- category 1 – locations where SNM was highly probable;
- category 2 – locations where it was possible that SNM could have been deposited;
- category 3 – locations where it was unlikely that SNM was deposited.

Category 1 locations required that measurements or, in selected cases, analyses, be performed for SNM. Category 2 areas were considered to have a lower probability for fuel deposits but were assessed in the same manner as category 1 areas. Category 3 areas were determined not to require SNM assessment based on analyses of the TMI-2 accident and review of recovery activities.

The quantity of residual SNM in each location was documented in a GPU Nuclear engineering calculation. The engineering calculations, in turn, provided the quantity of SNM for a specific area, system or component.

Final accountability was performed by summing the residual fuel quantities and reporting the results as the remaining plant inventory of SNM. The amount of fuel shipped to the Idaho National Laboratory was determined by subtracting the sum of the final plant inventory and the amount of SNM shipped as radioactive waste from the pre-accident plant inventory of SNM, as corrected for decay in the most recent SNM Material Balance Report.

Pre-accident reported inventory (corrected for decay):

- final in-plant inventory;
- SNM shipped as samples/radwaste;
- SNM shipped to Idaho National Laboratory (in canisters).

The resulting SNM inventory was reported in the Post-defueling Monitored Storage (PDMS) SNM Material Balance Report (DOE/NRC Form 742). This was the method used to demonstrate to the NRC that approximately 99% of the original TMI-2 core had been removed from the site.

Saxton concrete and soil disposition

Another waste disposal issue that affected GPU Nuclear and that is similar to issues at Fukushima Daiichi occurred during decommissioning of the Saxton Nuclear Experimental Corporation Facility (Saxton) in Saxton PA. Saxton was a small (23.5 MW) test reactor that operated from 1962 to 1972. As a result of Saxton operation, several hectares of surrounding property were affected and needed to be cleared of radioactive material.

During excavation of soil at Saxton, a standard backhoe was utilised. Each bucket of excavated soil was scanned by a health physics technician using a NaI detector screened for the Cs-137 peak. If the dose rate in the bucket exceeded a threshold value, the soil was considered contaminated and was packaged for disposal. If the dose rate of the soil in the bucket did not exceed the threshold value, it was staged in the soil lay down area for later processing.

There was only one instance of soil that was contaminated by hazardous material. In this case, it was discovered that a capacitor containing polychlorinated biphenyl (PCBs) had leaked. This contaminated soil was segregated during the excavation process and disposed of as mixed hazardous/radioactive waste. A total of 16 195 tonnes of material consisting of backfill debris (11 183 tonnes) and soil (5 012 tonnes) was surveyed through a radiation detection system developed by Shonka Research Associates. The scanned

debris and soil were separated into approximately 250-tonne piles called batches. Although there were a number of different types of materials present among the piles, each individual pile appeared to be a homogeneous mixture of the same type of material. A total of 56 batches of material, i.e. backfill debris (38 batches) and soil (18 batches), were surveyed.

The effective volumetric release limit for the soil and debris material was calculated. A conservative mix of seven radionuclides was used from Saxton debris samples to determine the limit. These radionuclides and mix percentages were as follows: Ni-63 (69.4%), Cs-137 (28.6%), Sr-90 (0.3%), Co-60 (1.0%), Am-241 (0.5%), Pu-238 (0.1%) and Pu-239 (0.1%). To ensure material would not exceed the release limit, an alarm set point was established at 70% of the limit.

The radiation detection system was a conveyor version of the subsurface multi-spectral contamination monitor (SMCM) that used four, five-inch (13 cm) diameter by two-inch (5 cm) thick thallium-doped sodium iodide (NaI (Tl)) detectors. The detectors were arranged in a line along the path of the conveyor and were located one-half metre apart. The nominal conveyor speed was established at 4 inches per second (0.1 m per second), with spectra collected every 19.7 inches (0.5 metres) of conveyor travel. The conveyor had material limited to 32 inches (0.8 m) wide and 4 inches (0.1 m) deep, with the face of the detectors located 13 inches (0.3 m) from the surface of the conveyed material. This height was chosen to provide a reasonable compromise between uniformity of response and sensitivity to localised sources.

The detectors were centred in 19.7-inch (0.5 m) diameter barrels. The detectors have thermal shielding, heaters, thermocouples and controls for temperature stabilisation, and are shielded with approximately 4 inches (10.2 cm) of sand to reduce the radiation background as well as any variability from changes in background (due to radon in air, moving vehicles, or changes in nearby soil and building debris piles). The detector array was located in an enclosure above the conveyor that is also heated to provide a uniform thermal environment without diurnal variation. The sand shielding restricted the field of view of the detectors to a downward looking, nominal 90-degree angle cone. A 12-foot (3.6 m) by 5-foot (1.5 m) trailer served as a mobile command centre (MCC). The SMCM process computer and post-processing computer were operated from within the MCC.

Twenty-eight alarms occurred during the survey that included 5 258 (includes 5% re-surveyed) tonnes of soil. If an alarm occurred, the conveyor was stopped and the data was investigated. The SMCM operator would review the strip chart on the SMCM process software screen. The strip chart shows the four detectors and the diagonal mean of the four detectors. From the strip chart, the operator is able to determine if the alarm is a point source or a distributed source and where along the belt the suspect material is located. The best estimate of the source distribution was then described for investigation. Generally, large source distributions would motivate removing dirt from the entire survey conveyor. If the source was localised to a single acquisition, the affected acquisition and at a minimum the two adjoining acquisitions were removed.

Following an alarm from the SMCM, a scan survey was performed on the suspect material using a Ludlum 2 350 (or equivalent) metre with a 2 inches (5 cm) by 2 inches (5 cm) sodium iodide detector. Any material indicating activity greater than or equal to a specified limit was removed and contained.

Summary

GPU Nuclear faced many of the same challenges as TEPCO does at Fukushima Daiichi: HLW, damaged fuel, water that could not be discharged and significant volumes of potentially releasable material. Working with the US nuclear industry, the NRC and the DOE, each of these challenges were met and allowed GPU Nuclear to complete the TMI-2 Cleanup Program and place the plant in monitored storage.

In decommissioning the Saxton nuclear experimental facility, GPU Nuclear was able to demonstrate that large volumes of potentially contaminated material were below regulatory limits and thus did not need to be sent to a low-level waste disposal facility. Instead this material was used for backfill on the site thus not only saving the cost of disposal but also saving the cost of purchasing backfill. These projects provide valuable lessons learnt on waste management and should be further studied.

Additionally TMI-2 provides some valuable insights for any utility struggling with recovery from a major nuclear accident. The Electrical Power Research Institute published an important report on the history of the TMI-2 Cleanup Project, “The Cleanup of Three Mile Island Unit 2 – A Technical History: 1979 to 1990”, EPRI NP-6931, in September 1990. In addition to technical details on the clean-up, it provides management insights.

The management of the TMI-2 clean-up was uniquely demanding. It not only comprised a complex technological mixture of necessary innovation and unfamiliar safety issues, but saw many of the technical decisions influenced by nontechnical factors.

The technical decisions and the course of the clean-up were inextricably bound with issues of management organisation, planning, funding, a sceptical public, media spotlighting, and regulatory investigations and restraints. Most technical decisions involved internal debates reflecting these issues.

Basic management decisions

Questions about how to do almost everything had to be evaluated in light of the unprecedented post-accident situation. How to organise? How to fund? What were the overall objectives? The answers resulted in a new company, a unique clean-up organisation, shared funding of the work, and novel forms of federal and industry involvement.

In programmatic terms, the following major management decisions were made:

- **Survive** – In a fundamental decision, GPU elected to fight the threat of bankruptcy and potential federal takeover of the clean-up. The utility created a subsidiary devoted strictly to nuclear matters (GPU Nuclear) and a division within it devoted solely to the clean-up of TMI-2. It also created a support division chartered to concentrate on radiological controls. By physically and operationally separating units 1 and 2, GPU removed one potential argument against the restart of unit 1, which was essential for corporate health. The credibility and progress of the unit 2 clean-up minimised the potential of it being used as an issue in the unit 1 restart proceedings.
- **Ensure utmost safety** – The decision to perform the work with safety as a paramount issue guided the clean-up. Although at times this was carried to conservative extremes – adding technical difficulties, expense and time – no alternative was acceptable. In fact, the clean-up was carried out at a personnel radiation exposure level within the NRC’s estimate (less than 6 500 person-rem) and with an Occupational Safety and Health Administration (OSHA) lost-time accident rate better than at many operating plants. The overall goal of the clean-up was to establish a condition of stability and safety such that there was no risk to public health or safety.
- **Use experts** – GPU Nuclear immediately realised that many aspects of the clean-up were beyond its expertise. The use of resources from the government, other utilities, national laboratories and universities brought a sophisticated technical presence to the clean-up. In particular, the DOE laboratories had skills and special facilities that did not exist elsewhere. Combining these outside resources with the on-site workforce was difficult, but the combination brought much needed

technical and financial support, new ideas and a channel to the worldwide technical community.

- Hire a contractor – In making this decision, GPU Nuclear hired Bechtel, the largest nuclear power A/E-constructor in the world; Bechtel had resources and expertise, or access to them. An alternative would have been for GPU Nuclear to have increased its staff and hired subcontractors – a drain on resources that GPU Nuclear was not in a position to undertake. By hiring a contractor, GPU Nuclear could get back on its corporate feet while performing normal plant operations. In 1982, GPU Nuclear decided to integrate with Bechtel staff and other subcontractors to form one clean-up organisation. In itself, this was a major and essential step that caused some painful adjustments and delays.
- No restart – For some time after the accident, GPU Nuclear envisioned returning unit 2 to service or at least they did not preclude a restart. Public opposition to restart would have been intense. As the extent of damage to the reactor core and expense of plant refurbishment became evident, the decision was made to focus on the defueling effort and work without regard to the final disposition of the plant. At first, this was difficult to accept for engineers and operators trained in maintaining or improving an operating power plant. The overwhelming advantage was that the decision focused available resources on near-term issues. Eventually, GPU Nuclear decided to place the plant in a long-term monitored storage condition after fuel removal.
- Pursue flexible/parallel approaches – No one knew how hard the clean-up would be or how long it would take. No one knew what the conditions were inside the reactor vessel or what defueling tools would work. In this situation, project management found, time and again, that schedules and plans were quickly outdated. The only practical approach became to establish an overall strategy and then take steps one at a time. Financial restraints played a role; but more importantly, the unique nature of the damage and the need to evaluate conditions before expending resources too quickly dictated that the strategy would be to “eat the elephant one bite at a time” (Dieckarnp, 1983). Flexibility required parallel and sometimes redundant approaches until an effective method was found. (Since the failure of one plan or piece of equipment could stall the work for months while another was developed, the policy made sense. It also meant that if one action was stymied by public or regulatory debate, progress could still be made.)
- Challenge the system – The project team struggled in a difficult regulatory and public environment. Since NRC rules had not been written for post-accident conditions, attempting to fit existing rules was often burdensome. By continually showing that plant conditions were safe, management slowly reduced the burden of specific operating plant requirements to reflect the stability of TMI-2 and the progress of the clean-up.

Management insights

In terms of managing the clean-up, several general insights stand out:

- The details of any post-accident scenario are unpredictable and specific to the situation. Responding to them requires elements that are alien to conventional utility and service management organisations. Managing the TMI-2 clean-up required an entirely different philosophy and approach from that used to design, construct or operate a plant.
- Before beginning much of the clean-up work, the theoretical parameters of a post-accident situation should be understood. At TMI-2, this was necessary before developmental work could be performed to prepare the way for production defueling work.

- A heavily project-focused approach is more effective than a large functional organisation of engineers and designers responsible for small bits of several projects. The personal involvement and the direct knowledge that this approach created were invaluable assets at TMI-2.
- While redundancy in organisational functions is expensive and difficult to manage, some degree of redundancy is prudent to ensure that all options and potential problems are considered.
- In the early phases of clean-up, a centralised, high-priority effort is needed to provide data on actual physical conditions. Visual observations are essential to understanding and efficiency. Visual observation was often necessary at TMI-2 before unexpected or hypothesised conditions were accepted as real.
- Proceeding on arbitrarily conservative or optimistic assumptions may be counter-productive because the real situation will likely be different. Emphasis must be placed on having hard characterisation information before building systems and facilities.
- Insights from experienced senior technical advisors are invaluable. Although difficult to integrate with the workforce, their third-party review is essential in fields where new ground is broken.

The on-site location of production staff and experts leads to increased efficiency and a pragmatic understanding of conditions.

1.3. Kola case study: With a focus on the remediation of the Andreeva Bay site of temporary storage of spent fuel and radioactive waste in northwest Russia

Background and history

A review of existing and future requirements for decommissioning of nuclear facilities in the Commonwealth of Independent States (CIS) (EC, 1999) included in its key conclusions the need for:

- clearly defined regulatory requirements for decommissioning of nuclear facilities;
- clearly defined waste management and disposal routes.

These conclusions strongly reflected weaknesses in the situation in the CIS at that time but remain key issues for any strategy addressing nuclear decommissioning.

Particular focus on the Kola Peninsula arose during the development of a Strategic Master Plan (SMP) with support of the Northern Dimension Environmental Partnership (NDEP). The SMP was designed to address the decommissioning of the retired Russian nuclear fleet and environmental rehabilitation of its supporting infrastructure in northwest Russia. Phase 1 of the plan, as reported in IBRAE (2007), drew special attention to the need for decommissioning and restoration of the previous shore technical bases at Andreeva Bay and Gremikha. These bases were developed initially in the 1960s for maintenance of nuclear submarines, performing receipt and storage of radioactive waste and spent nuclear fuel SNF. No further waste was received after 1985 and the shore technical bases have since been recategorised as sites of temporary storage (STS). Here below, the situation is described with respect to STS Andreeva Bay.

Site for the temporary storage of spent fuel and radioactive waste, Andreeva Bay

General description and source terms

The materials in store at STS Andreeva Bay, as reported in (AMAP, 2010) comprised SNF and solid and liquid RW: approximately 21 000 spent nuclear fuel assemblies and about 12 000 m³ of solid and liquid radioactive waste.

The STS consisted of the following main constructions (Shandala et al., 2008):

- fixed-site technological berth;
- blocks of dry storage – three partly underground 1 000 m³ stores, re-equipped to serve as facilities for the SNF storage;
- service site for the SNF store, including some buildings;
- basin-type SNF storage facility – building 5 being decommissioned after removal of the SNF from the building;
- liquid radioactive waste (LRW) storage facilities;
- building intended for water purification;
- storage facility for high-level concentrates of LRW after treatment;
- numerous constructions and sites for solid radioactive waste (SRW) store.

The following circumstances identified by a threat assessment carried out from a regulatory perspective in 2005 and reported in Ilyin et al. (2005) critically characterised this site as follows:

- unsatisfactory condition of facilities, hampering safe SNF and RW management;
- radioactive contamination dispersion from the STS territory to the adjacent marine environment;
- lack of regulatory requirements and guidance to deal with the existing abnormal radiation conditions;
- lack of relevant standards for the complete management of radioactive waste.

The following factors exacerbated the problem of management (Ilyin et al, 2005):

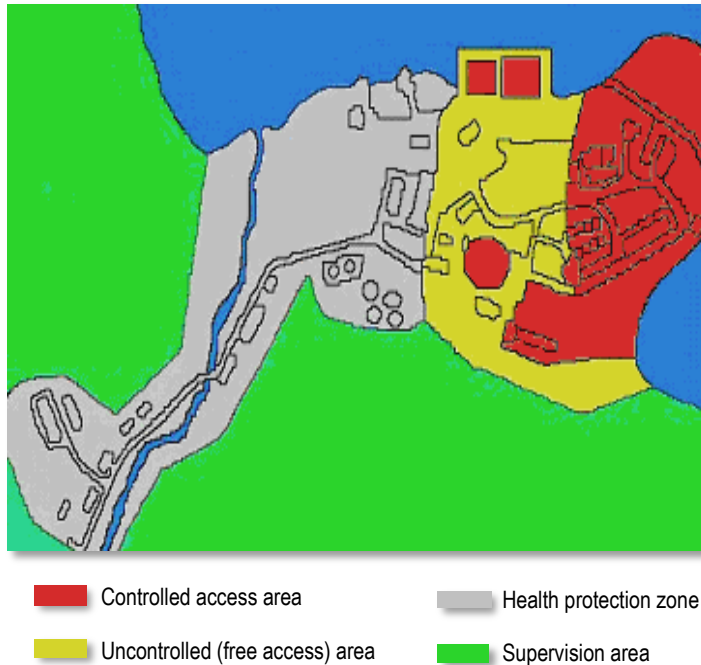
- damage to the SNF and the engineered barriers of the storage facilities, leading to radioactive contamination of the environment, and a continuing threat of further releases;
- gaps in regulations on procedures connected with specific aspects of SNF and RW management, including insufficient definition of requirements for remediation;
- justified public concern that environmental safety may be jeopardised not only in Kola Peninsula and the European part of Russia, but also in other countries of northern Europe.

The Russian strategy for addressing this situation drew upon a wide range of industrial projects which in turn receive support from donor organisations and technical institutions, co-ordinated through the International Atomic Energy Agency's (IAEA) Contact Expert Group (CEG). The Norwegian Radiation Protection Authority's (NRPA) bilateral regulatory co-operation programme with the Federal Medical-Biological Agency of Russia (FMBA) was designed to provide parallel support to the Russian regulatory authorities, with a view to ensuring that investments made to manage the nuclear legacy in northwest Russia would be spent safely within the context of an effective regulatory regime.

Site characterisation

For the purpose of the radiological protection of workers, Andreeva Bay has been divided into four separate radiological protection areas (see Figure 1.1). The zoned areas are subject to change boundaries as work progresses.

Figure 1.1. Area categorisation at Andreeva Bay



Source: Shandala et al., 2008.

Controlled access area (CAA): Facilities are located within this area where SNF and RW are stored and where radiation-hazardous operations are carried out. The facilities in this area have been the main subjects of remediation. This area was appointed with decontamination posts and a special regime of work was defined for the area. Personal protection equipment was applied in the CAA to provide radiological protection for personnel from Group A (dose limit – 20 mSv/y).

Uncontrolled (free access) area (UA): Facilities located in this zone support work implementation in the controlled access area. No radiation-hazardous operations are performed within this area. The main workplaces of the personnel from B group were located here. Personnel from B group were subject to the dose limit 5 mSv/y.

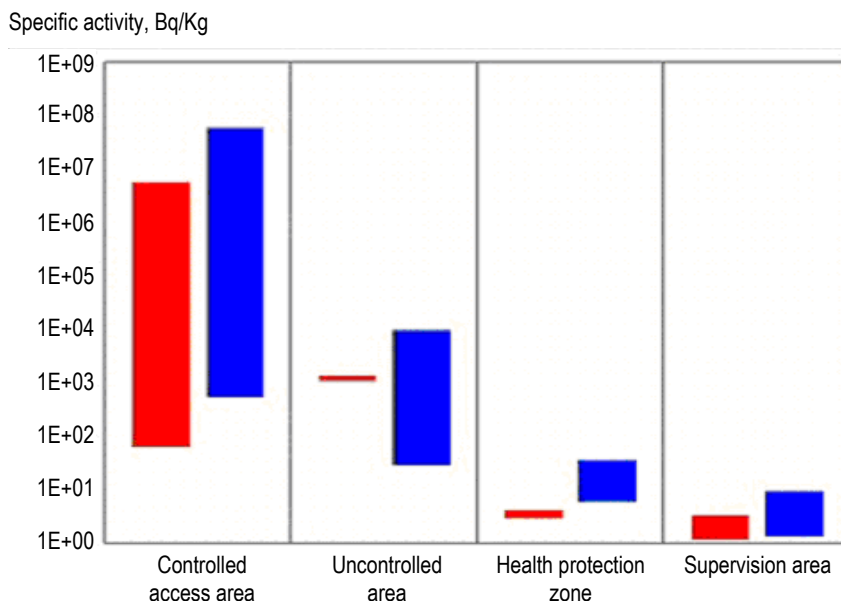
Health protection zone (HPZ): This is the area of STS administrative and technical provision. The external border of this area was limited by the system of physical protection of the engineered area.

Supervision area (SA): This area, with a radius of about 10 km, was the subject of supervision of the facility impact on the environment and the public (dose limit – 1 mSv/y).

Apart from the stored SNF and RW, there is significant contaminated soil and sub-soil at the site (Shandala et al, 2008). A wide range of environmental sampling has been undertaken, to support the planning of remediation operations and to help plan for the long-term management of the site. Figure 1.2 illustrates the ranges of Sr-90 and Cs-137

contamination in the different designated areas, as reported in NRPA. More details are provided therein, alongside the description of site-specific regulatory controls and requirements which were developed to cover: optimisation of radiological protection of workers, criteria for site monitoring and control, management of very low-level waste, and emergency preparedness and response.

Figure 1.2. Sr-90 (red) and Cs-137 (blue) content in soil on-site at the STS in Andreeva Bay



Source: Shandala et al., 2008.

Dose control and radiological assessment

Prognostic radiological assessment is considered a vital part of the planning for safe and effective remediation. It relies on a sufficient understanding of the current situation, in terms of the existing facilities, nature of the environment and radioactive sources. The assessment of future conditions, for example, dose rates at work places once some sources have been removed, or the radiological impacts of residual contamination, supports the selection of appropriate management operations, as well as effective regulatory supervision.

To this effect, two assessment tools have been developed, one related to the radiation situation and worker exposure monitoring (DOSEMAP) and one related to radio-ecological assessment (DATAMAP). Both are supported by GIS systems and are described in detail in NRPA (2011) and, with further illustrations, in Chizhov et al. (2014) and Sneve et al. (under review).

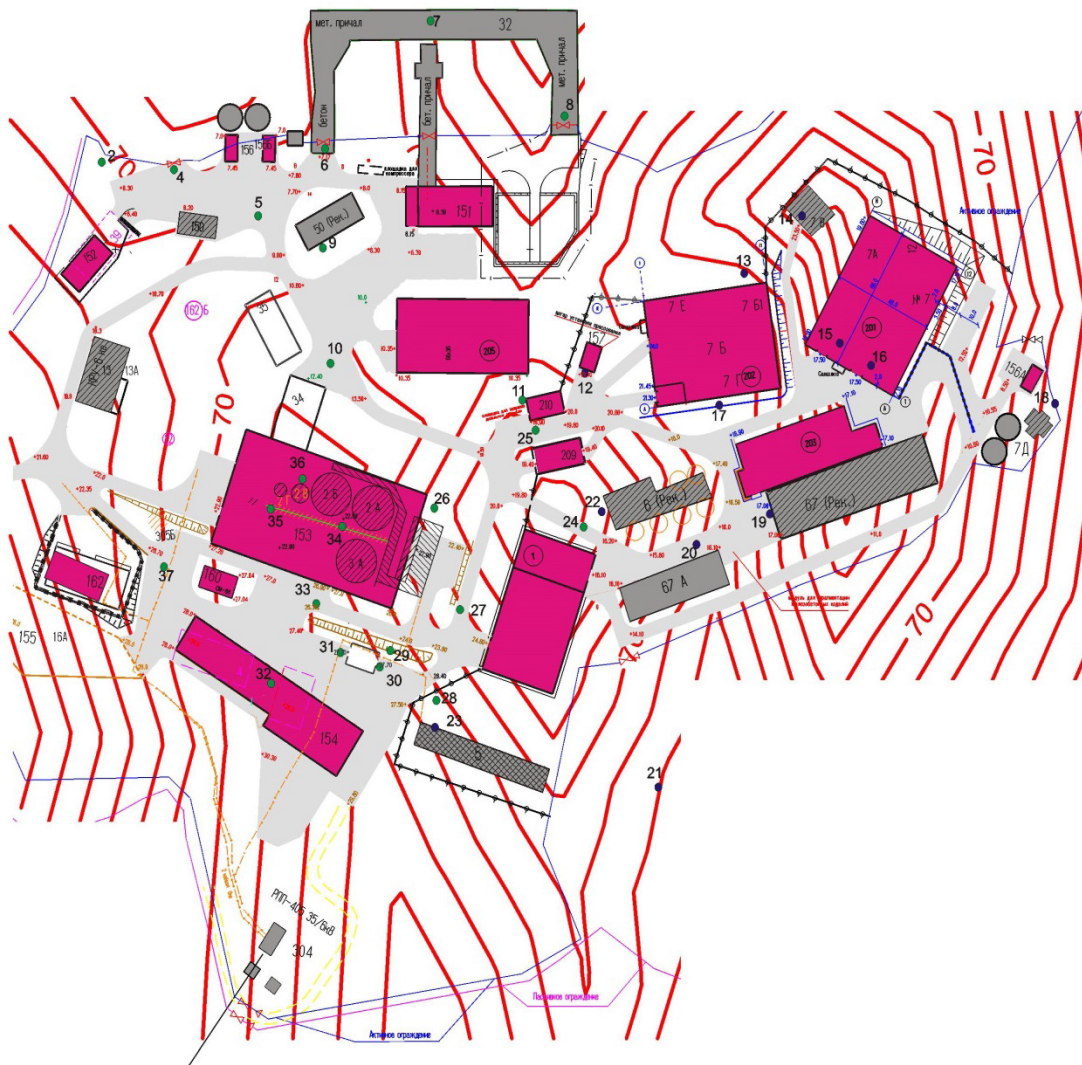
Figure 1.3 provides an example of output from DOSEMAP, showing the industrial site and isolines of ambient dose rate. The example serves only to illustrate potential graphic representation of real data, which can be developed from results of relevant measurements and put into the assessment tool database. Such techniques can be used to assess the actual current situation and the effect of removing particular source terms, and hence optimisation of the protection of workers.

The DOSEMAP assessment tool focusses on the management of environmental contamination. It relies on measurements on environmental radioactivity in soils, sub-soils and groundwater and on the use of suitable interpolation techniques. It provides the following functions:

- presentation of contamination levels in two or three dimensions;
- identification of areas where available data is not sufficient for necessary analysis of the radio-ecological situation;
- assessment of future contamination levels, allowing for radioactive decay and models for migration and accumulation of radionuclides.

Results like those presented in Figure 1.3 combined with proposals for remediation work activity locations can be readily used to identify the most relevant locations for future sampling activities, which in turn support the next iteration of assessment and work planning.

Figure 1.3. Andreeva Bay facilities and an illustration of how isodose curves can be presented in DOSEMAP

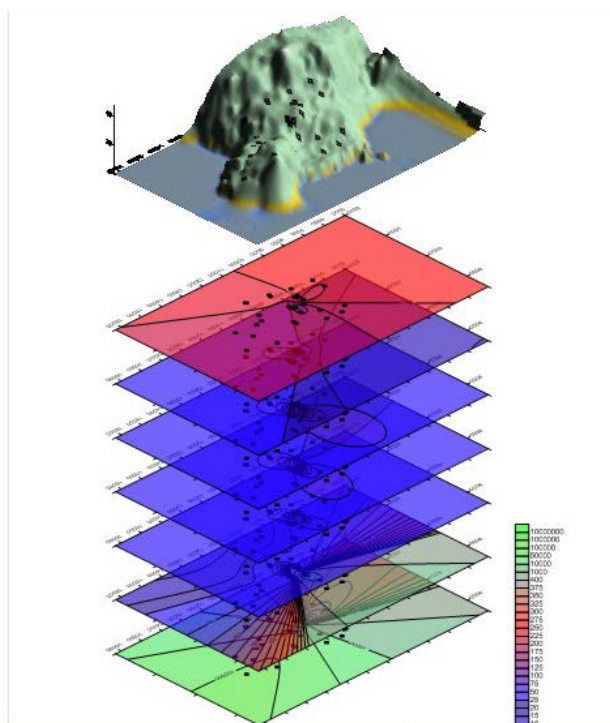


Source: Roudak et al., 2011.

The focus of environmental monitoring has been on Cs-137 and Sr-90 because historic measurement and assessments (e.g. Ilyin et al., 2005) indicate that these are the dominant radionuclides present and also likely to be the dominant contributors to external and internal doses, both during current operations outside buildings and in the longer-term stages of site management. These radionuclides are also considered to be the most relevant radionuclides to take into account when planning long-term site restoration and decisions on management of radioactive waste arising from remediation of contaminated areas, and as such, as many samples as practically manageable are collected.

Figure 1.4 provides an example of output from DOSEMAP of a 3D representation of activity distribution over depth in the ground. Such measurement and representation can support the understanding of the potential for migration of radionuclides in groundwater. The control of the spread of contamination by groundwater is a special subject to consider.

Figure 1.4. Three-dimensional plot of activity distribution over borehole depth



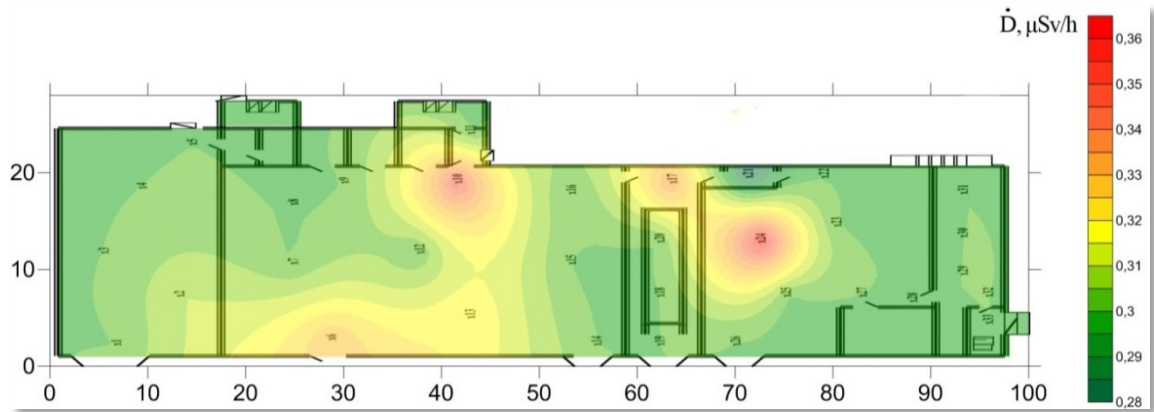
Source: Shandala, 2013.

Figures 1.5 and 1.6 illustrate the use of the DOSEMAP visualisation tools to support planning of work tasks and control of worker exposure (Chizhov et al., 2014).

Interpolation between sampling points is done by the standard method of kriging. Identification of positions where radiation control requires special attention is determined based on maximums of the dose rate gradient (method 1), and maximums of the interpolation error, i.e. the method of cross-validation (method 2), described further in Chizhov et al. (2014).

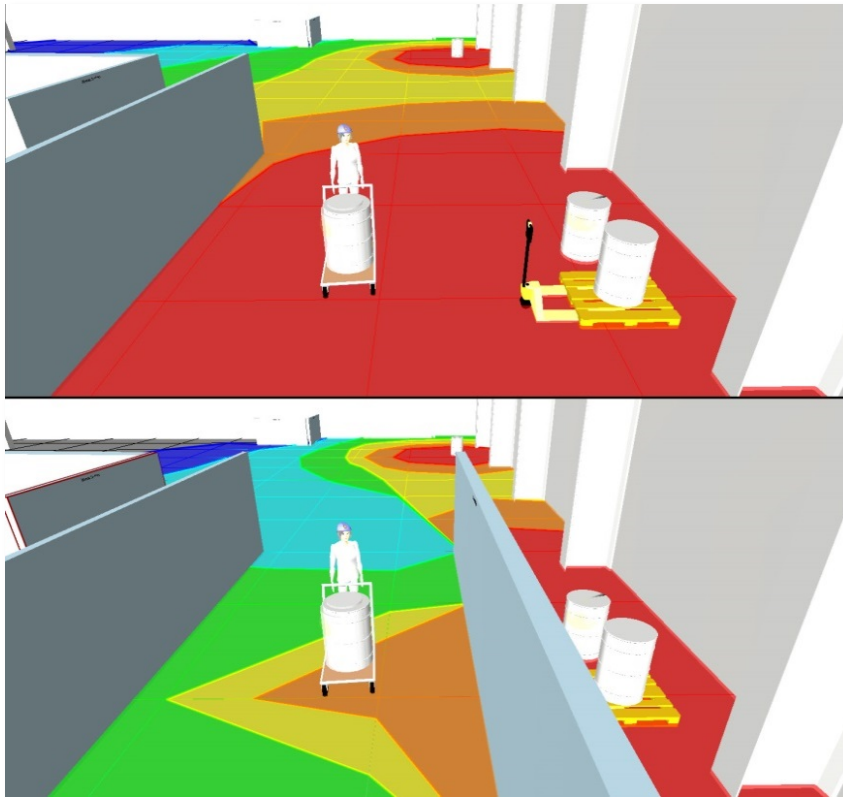
Apart from planning and optimisation of work in hazardous areas, the visualisation tools can be used in training for particular operations, as illustrated in Figure 1.7.

Figure 1.5. Visualisation of radiological conditions for supporting efficient zoning within a building containing hazardous radiation sources



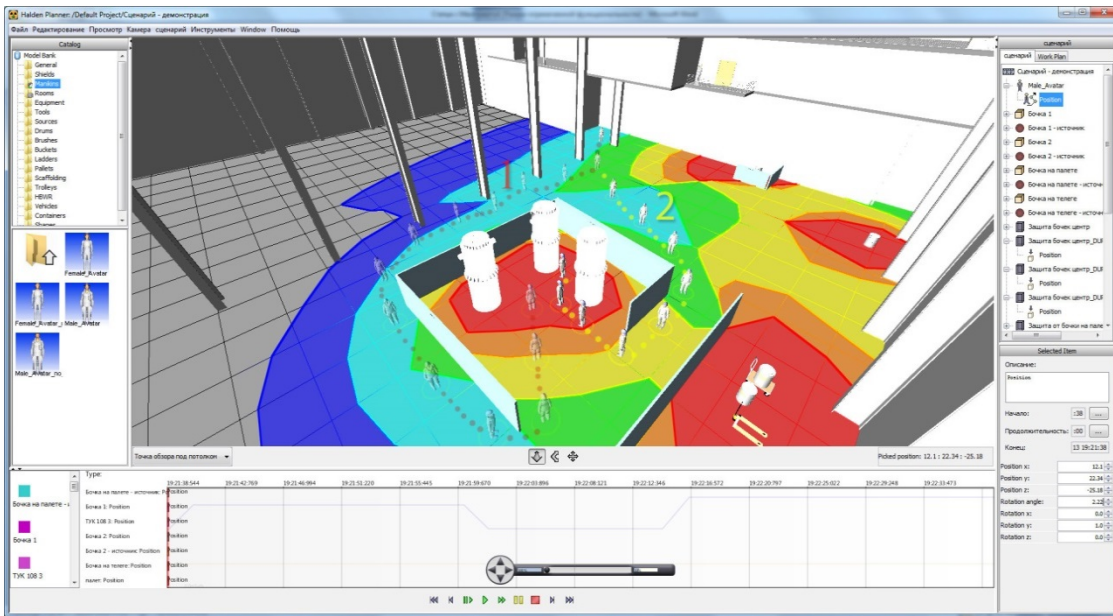
Source: Chizhov et al., 2014.

Figure 1.6. Screenshot of the Andreeva Planner demonstrating dynamic (real-time) the radiological risk assessment (dose rate with and without shielding wall)



Source: Chizhov et al., 2014.

Figure 1.7. Screenshot demonstrating application of DOSEMAP visualisation tools for supporting classroom training



Source: Chizhov et al., 2014.

Progress with remediation

Since 2002, substantial construction of new and reconstruction of available infrastructural components has been under way. Table 1.1 shows examples of the changed radiation situation in what were some of the more highly contaminated areas.

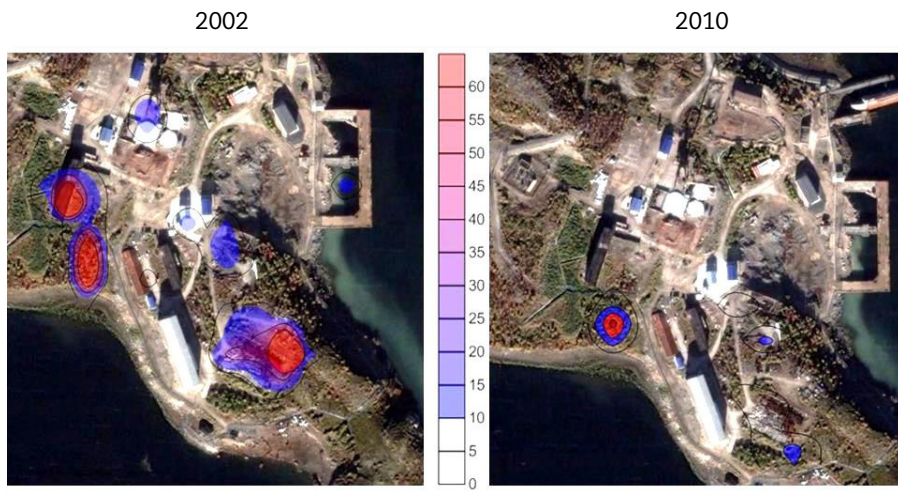
Table 1.1. Changed radiation situation following remedial actions

Location	Dose rate, $\mu\text{Sv/h}$		Action taken
	2002	2009	
Area near new pier	0.15-450	0.15-0.35	Old pier dismantled
Around building 50	0.3-1.5	0.25-0.57	Removal of scrap metal landfill
Various damaged buildings	0.58-2.7	0.38-1.1	Sand backfilling and asphalt covering
Motor transport decontamination area	2.5-30.7	0.57-0.7	Paving of the site

Source: Shandala and Sneve, 2014.

Figure 1.8 illustrates the change in radiation situations across the site over the period from 2002 to 2010 (from Shandala, Kiselev and Klimova, 2011, "Russian Experience in comprehensive regulatory supervision of former military technical bases", Presentation at WM2012, American Nuclear Society.)

The measures outlined above are not considered complete or final, but are part of the ongoing process of remediation, allowing for lower dose-rate working areas, and taking into account parallel developments of necessary regulatory requirements and guidance. It is regarded as vitally important that these regulatory documents and corresponding procedures have been put in place prior to the commencement of the main and most hazardous operations, due to take place in the coming few years.

Figure 1.8. Change in radiation situation across the site over the period of 2002 to 2010

Source: Shandala et al., 2011.

Regulatory developments

Above, the need was noted for clearly defined regulatory requirements for decommissioning of nuclear facilities. For planned situations, it can be expected that development of such requirements can be built into overall programmes of activities. For legacy situations as they are described in EC (1999), but also in the case of decommissioning of facilities which have suffered severe accidents, the requirements may not be adequate to address the abnormal and unplanned circumstances. This was the reasoning behind the regulatory threat assessment reported in Ilyin et al. (2005) carried out by the FMBA in co-operation with various support organisations as part of the NRPA-FMBA regulatory co-operation programme.

The threat assessment and subsequent regulatory developments took into account all aspects of radiation safety including:

- protection of workers involved in the most radiation-hazardous operations, including application of optimisation;
- protection of the public and the environment;
- emergency preparedness and response;
- requirements for environmental monitoring;
- regulatory identification of possible end-states for the territory of the STS and corresponding radiation description;
- monitoring of performance reliability of workers involved in SNF and RW handling activities.

Arising from the initial activities, it became very apparent that the second issue noted in Section 1.1 from EC (1999) also applied to operations at STS Andreeva Bay, i.e. clearly defined waste management and disposal routes. Accordingly, further guidance was developed to cover the following RW storage and disposal issues:

- waste acceptance criteria for packages and materials to be stored at the Saida Bay waste treatment and storage facility;
- criteria for re-designating waste containing fuel fragments as RW rather than SNF;
- criteria for on-site disposal of very low-level radioactive waste (VLLW).

Development of corresponding regulatory guidance and other documents was carried out in all three areas with the full co-operation of, and working dialogue with, the waste producers, while at the same time maintaining clear and separate responsibilities of the respective organisations. The details of the waste acceptance criteria and arrangements for managing fragments of fissile material are quite specific to the conditions at the STS Andreeva Bay, which are very different from those in a degraded and severely damaged reactor. However, it is highlighted that, at this site, the implementation of an effective regulatory basis for waste management occurred in parallel with waste characterisation work, prior to the most hazardous recovery operations which are due to take place in the coming few years (Grigoriev, 2015). This allowed the specification of detailed regulatory requirements to take account of new information as it became available as part of a carefully controlled step by step process.

An important feature of the VLLW criteria was that they were designed to take account of the site-specific nature of the materials chemical and other hazards that might be associated with VLLW.

The need to characterise and assess the chemical and other non-radiologically hazardous content of radioactive waste, alongside the radioactive content, has been examined internationally in NRPA (2015). It was concluded that, ideally, a holistic approach to assessment of radionuclides and hazardous materials should be internationally developed and applied, such that consistent assumptions are employed in assessments and consistent criteria used in the evaluation of risk. Currently, the basis for separation in approaches includes traditional behaviour, regulatory and institutional differences, lack of common language in addressing issues with respect to both waste types, lack of international guidance on criteria for assessments, as well as lack of supporting information from science. The development of a common set of objectives and, hence, assessment endpoints and time frames for the different waste types would be very beneficial. In particular, this would promote the proportionate allocation of resources to the different types of hazards associated with the waste. In cases where technical differences are necessary, a clear understanding of the reasons for the different approaches should be provided to allow differences to be understood and communicated. In the case of VLLW waste at Andreeva Bay, the way to achieve such coherent and intelligent risk management was to regulate its management within the framework of industrial hazardous waste management, but with due account given to management of the radiation hazards.

The list of documents developed within the regulatory co-operation programme related to STS Andreeva Bay and related waste management includes:

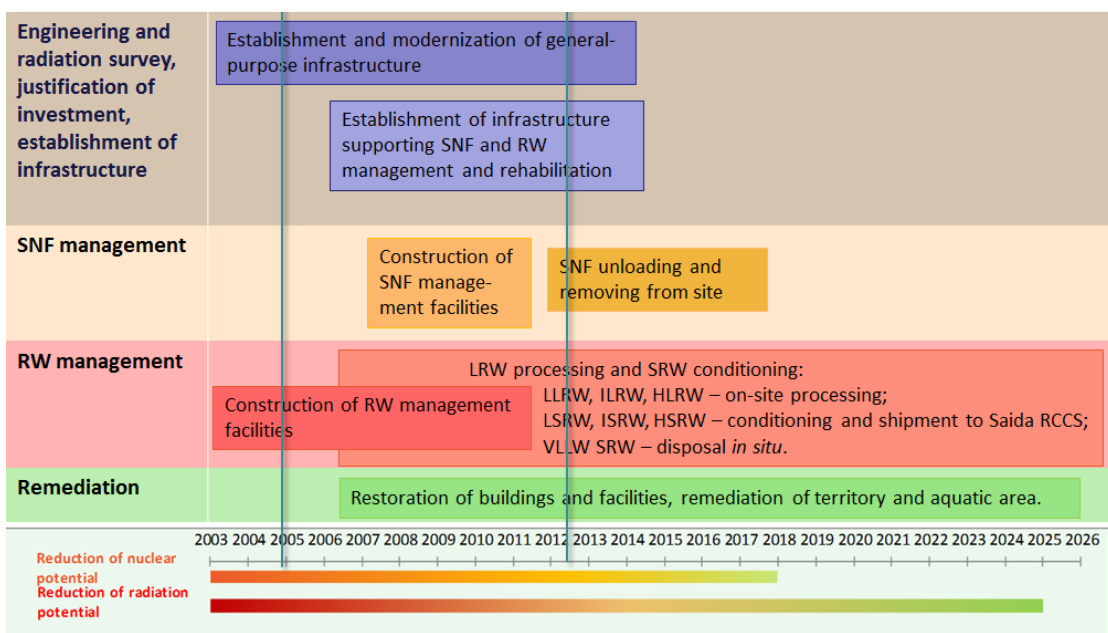
- “Criteria and Norms for Remediation of Sites and Facilities Contaminated with Man-made Radionuclides”, R 2.6.1. 25 – 07.
- “Hygienic Requirements for Radiation Protection of Workers and the Public during Planning and Arrangement of the SNF and RW Management at the SevRAO Facility-1 (R-GTP SevRAO-07)”, R 2.6.1. 29 – 07.
- “Arrangement of the Environmental Radiation Monitoring in the Operational Area of the Federal State Unitary Enterprise ‘Northern Federal Enterprise for Radioactive Waste Management’ of the Federal Atomic Energy Agency”, MU 2.6.1.37 – 2007.
- “Hygienic Requirements for the Industrial Waste (VLLW) Management at the SevRAO Facility (R ONAO SevRAO-08)”, R 2.6.5.04 – 08.
- “Personal Dose Monitoring of the Occupational Exposure at SevRAO Facility-1”, MU 2.6.5. 6 – 08.

- “Special features of ALARA principle application during the SNF and RW management at SevRAO Facility-1”, MU 2.6.5. 05 – 08.
- “Procedure of Radiation Monitoring at the SevRAO Facility-1”, MUK 2.6.5. 7 – 08.
- “The Operational Radiological and Medical Criteria for the Initiation of Emergency Protective Actions in the Case of Radiation Emergency at the SevRAO Facilities”, approved by FMBA of Russia, 2008.
- “Procedures of Radiation Monitoring and Individual Dose Monitoring during SNF and RW Management at SevRAO-2 Facility (Ostrovnoy)”, MUK 2.6.5.021 – 2009.
- “Requirements for Protection of Workers, Public and Environment during Arrangement of Radioactive Waste Management in the Centre for Conditioning and Long-Term Storage at the SevRAO”, R 2.6.5.028 – 2010.
- “Application of the Information and Analytical System for Prediction of Doses to Workers to Regulate Radiation Protection of Workers of NWC SevRAO”, MU 2.6.1. 064 – 12.
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- Guide on “Administrative Requirements Providing Safe Management of Objects Containing Nuclear Materials, while Transferring them to the Category of Radioactive Waste”.

All these regulatory documents are available in English.

Technical programme for Andreeva Bay remediation

The main stages of site remediation, from an engineering perspective, are illustrated in Figure 1.9 according to the schedule defined in Minatom (2004). The actual progress is delayed only by about two years, and trial SNF recovery operations have been carried out. It is notable that regulatory developments have been needed in all stages to address the abnormal conditions. The need for continued close regulatory supervision during future main operations to recover poorly stored SNF should be further discussed (Sneve et al. [2015]).

Figure 1.9. The main stages of remediation of STS at Andreeva Bay

Source: Minatom, 2004.

1.4. Windscale

General description of the accident

The 1957 fire in Windscale Pile 1 is rated as a level 5 accident on the International Nuclear Event Scale. It resulted in the accidental discharge of radioactive materials from the pile chimney and their deposition over the surrounding area. A brief description of the accident and waste produced is given below. The description of the accident is taken from Arnold, 2007.

The Windscale site⁵ is located in Cumbria, England and was first developed for nuclear purposes by the UK Ministry of Supply in 1947. Construction of two air-cooled graphite-moderated nuclear reactors (“piles”) for the production of plutonium commenced in that year. Pile 1 went critical in October 1950; Pile 2 in June 1951. On 10 October 1957, overheating was detected in Pile 1. Radioactivity was detected on-site in air samples and by the on-site meteorological station. Later that day, an area of the pile was found to be on fire. Fuel elements from the affected area were discharged from the pile and attempts were made to create a “fire break” around the affected area by discharging fuel elements from the surrounding channels. Later, an attempt was made to extinguish the fire by pumping CO₂ into the pile. These measures did not bring the fire under control, and the decision was then taken to extinguish the fire by pumping water into the pile. The water was turned on at 9 a.m. on 11 October, and pumping continued for 30 hours until the pile was cold. A total of approximately 7 000 m³ of water was added. The fire was reported to have abated by midday on the 11 October. Clean-up operations began on 12 October.

5. The Windscale site is now part of the Nuclear Decommissioning Authority’s Sellafield site.

Health Physics surveys and biological monitoring, especially of milk produced in the surrounding area, commenced on 11 October and the decision to stop consumption of local milk was taken on 12 October on the basis of measured I-131 activities. (I-131 was recognised at the time as the major radiological hazard arising from the accident during emergency phase.) The area of the milk ban was increased to 200 square miles (approx. 518 km²) on 15 October, and an extensive programme of district surveys commenced. Based on the results of these surveys, distribution of milk from an area adjacent to Windscale was restricted for a period of several weeks.

Most of the water that was pumped into the pile was contained in the engineered structure and was ultimately discharged to the cooling pond. Some water overflowed into the forecourt of Pile 1 and, because of the high levels of radioactive contamination, was pumped back into the ponds. Inevitably, some of the fire water was lost through the base of the pile building into the underlying ground and into the surface water drainage system, and led to contamination of the surrounding ground. After the accident, fresh water was fed into the ponds and the contaminated water was discharged to sea.

The fire resulted in the interior of the pile becoming contaminated with damaged fuel and production isotopes. Potentially, uranium hydride had been formed when the water injection reached damaged fuel elements. Damaged fuel was observed on the discharge of the reactor (Ervin, 2008).

General description of the waste produced

Decontamination of buildings and removal of topsoil following the accident

Over the two years following the 1957 Windscale fire, extensive decontamination of buildings around the pile (e.g. blower houses, control room) was carried out and the topsoil from the immediate vicinity of the pile was removed (Arnold, 2007). The inventory and physical nature of this waste is poorly known, but it is expected to include significant volumes of contaminated soil and building materials/contents.

Waste produced from decommissioning the pile

The pile was defueled as far as possible by the beginning of November 1957. It is estimated (Ervin, 2008) that about 15 tonnes of the uranium fuel (out of a total of 180 tonnes) and up to 2 000 isotope cartridges (used to irradiate materials in the reactor) remained in the pile after this process. No attempt was made to clear the blocked channels or to remove debris from the air and water ducts beneath the pile. Following the sealing of Pile 1 in 1958, the pile was placed under long-term care and maintenance, with periodic camera surveys to check there was no serious degradation. Phase 1 decommissioning of the pile, to secure the safety of the facility, commenced in the early 1980s and was complete by 1999 (Cross, 2013). The work included putting dams in the original water ducts and air ducts to seal the bioshield. Monitoring systems were installed in and around the core to measure temperatures, radiation levels and airflow. Outside the core, accumulations of fuel debris, sludge and other radioactive waste were removed from the water duct (Cross, 2013).

At this time, a significant project milestone was achieved in that the Pile 1 Operational Safety Case was approved by the regulator. Approval from the regulator allowed surveys of the fire-affected zone of the pile to be undertaken for the first time. These surveys, undertaken in 2007 (Ervin, 2008), have allowed detailed plans for the final phase of decommissioning Pile 1 to be developed.

Waste produced from decommissioning the pile chimney

Following the fire, the contaminated filters were removed from the filter gallery at the top of the 110 m tall pile chimney, the air inlet ducts to the chimney were isolated and the top of the chimney was sealed. Decommissioning of Pile 1 chimney commenced in 1998, but was stopped in 2003 following a fatal accident. Decommissioning restarted in

2007, with removal of 78 tonnes of steelwork, rubble, lead and aluminium within the chimney base (Slater, 2013). Work to demolish the pile chimney itself began in 2013. The filter gallery and ancillary equipment has been removed; steel, brick and concrete waste has been produced. Preparations are underway to remove the diffuser, which sits below the filter gallery. The chimney will then be demolished down to about the 35 m level. The remaining structure will be placed in care and maintenance and demolished following the final decommissioning of the pile reactors.

Summary

Although the Windscale fire occurred nearly 60 years ago, the management of the accident provides lessons and insights that are still relevant today. In particular, extensive environmental monitoring was put in place during and after the Windscale fire; this provided the evidence that was used by the authorities when deciding whether to restrict milk consumption in the surrounding area. The prompt action in banning milk consumption is recognised to have reduced the radiological impact of the accident to members of the public.

However, the age of the accident means that it is not appropriate to present experience from the Windscale case study in the following areas: review of the techniques and approaches used for chemical, physical and radiochemical characterisation of waste at the time of the accident; dialogue with regulators and stakeholder engagement at the time of the accident; waste classification at the time of the accident; and waste conditioning and volume reduction at the time of the accident. More recent decommissioning activities, which started in the 1980s and are ongoing, have been undertaken as part of Sellafield Ltd's wider decommissioning programme, which includes decommissioning reactors and legacy ponds and silos. A short section on the Windscale case study is included in Section 8.2 to emphasise that all accident-related waste has been managed using the waste routes in use at the time for normal operational and decommissioning waste. No new storage or disposal facilities were developed specifically for fire-related waste.

1.5. Fukushima Daiichi nuclear power plant accident

Introduction

Outline of the Fukushima Daiichi nuclear power plant

The Fukushima Daiichi nuclear power plant (hereinafter referred to as "Fukushima Daiichi") is located at approximately the middle of the Pacific coast of Fukushima Prefecture, and straddles the towns of Okuma and Futaba of the Futaba District. The site is semi-elliptical in shape, extending lengthwise along the coastline, and the site area is approximately 3.5 million m². The power plant has six boiling water reactors. When the disaster occurred on 11 March 2011, units 1 to 3 were in rated output operation. Units 4 to 6 had been shut down for outage.

Overview of the Fukushima nuclear accident

At 2:46 p.m., on 11 March 2011, as a result of the Tohoku-Chihou-Taiheiyo-Oki Earthquake,⁶ whose focal area widely ranged from offshore of Iwate Prefecture to Ibaraki Prefecture, all of the operating reactors were automatically shut down. The distance to

6. Time and date of the occurrence of the earthquake: 11 March 2011 at 14:46, Hypocenter: off the Sanriku coast (focal depth of 24 km), Magnitude: 9.0.

epicentre and hypocentre from Fukushima Daiichi NPP is 178 km and 180 km, respectively.

At the Fukushima Daiichi NPP, the subsequent arrival of the tsunami, which is one of the largest in history, caused flooding of many cooling seawater pumps, emergency diesel generators (EDGs) and power panels. It caused the station black out (SBO) of units 1-5, and all the cooling functions using alternating current power were lost in these units. Consequently, the fuel in each unit was exposed without it being covered by water, and the fuel cladding was thereby damaged. Radioactive materials in the fuel rods were released into the reactor pressure vessel, and the chemical reaction between the fuel cladding (zirconium) and steam caused the generation of a substantial amount of hydrogen.

Later, in units 1 and 3, explosions, which appeared to be caused by hydrogen leakage from the primary containment vessel, destroyed the upper structures of their reactor buildings. In addition, another explosion occurred at the upper structure of the reactor building in unit 4 where all the fuel had been removed from the reactor, stored in the spent fuel pool (SFP) and kept under water in the SFP.

In Fukushima Daiichi units 5 and 6, one of the EDGs for unit 6 was in operation. By tying a power cable to unit 5, water could be supplied into the core of both units. After the recovery of the residual (decay) heat removal function from the reactor to the sea, units 5 and 6 reached cold shutdown.

The characteristics of the waste generated after the Fukushima Daiichi accident

Most of the waste that has been produced by the Fukushima Daiichi accident has been surface-contaminated, including contaminated water penetration. It is assumed that the radionuclides originated from the damaged fuels at units 1-3, and mainly consist of fission products with long half-lives. It is difficult to evaluate the future waste volume and its inventory correctly because of the kinds of contaminated depths at this point. Future decommissioning and dismantling plans have not yet been decided. The radionuclide dataset is insufficient to draw up a nuclide inventory, especially for long half-life nuclides.

Due to the hydrogen explosion that occurred at units 1, 3 and 4, massive waste was generated as concrete debris and metal debris, and also radioactive contamination was widely dispersed. Experience is also limited in the treating and disposal of contaminated trees and soil.

Inside reactor buildings, there is a massive volume of waste which is contaminated with a high dose rate. In addition, it is difficult to access the interior of reactor buildings to research and collect data.

At the Fukushima Daiichi NPP, contaminated water is a significant issue for the recovery and decommissioning of the plant. For treatment of the contaminated water, which has remained on the underground floor of reactor buildings and has been recirculated as reactor cooling water, different kinds of water filtering facilities have been installed. In the process of the water treatment, a certain amount of absorbing materials have been used and would be generated as waste or so-called secondary waste of water treatment. Secondary waste of water treatment and solid waste contaminated by high contaminated water should be studied to determine the treatment and disposal methods because an unprecedented water treatment and cooling system was installed emergently. It is difficult to directly pick secondary waste of water treatment and high-level water contaminated waste up for the analysis of activity, but estimation of radionuclides included in waste is possible through evaluation of radionuclides in cooling water.

There are three kinds of water treatment systems at the Fukushima Daiichi NPP: i) a caesium adsorption device has been installed to remove the caesium from contaminated water. This system generates spent zeolite used as absorbing materials; ii) A desalination

device, evaporative concentration and reverse osmosis have been installed as a part of the process for the recirculation of contaminated water as cooling water, and iii) a multi-nuclide removal system is being used for the removal of radionuclides from contaminated water (Table 1.2); ferric co-precipitation slurries and resin waste are generated as a result of this process.

Table 1.2. List of nuclides removed by the multi-nuclide removal system

No.	Nuclide	No.	Nuclide	No.	Nuclide	No.	Nuclide
1	Rb-86	17	Sn-126	33	Ce-141	49	Pu-240
2	Sr-89	18	Sb-124	34	Ce-144	50	Pu-241
3	Sr-90	19	Sb-125	35	Pr-144	51	Am-241
4	Y-90	20	Te-123m	36	Pr-144m	52	Am-242m
5	Y-91	21	Te-125m	37	Pm-146	53	Am-243
6	Nb-95	22	Te-127	38	Pm-147	54	Cm-242
7	Tc-99	23	Te-127m	39	Pm-148	55	Cm-243
8	Ru-103	24	Te-129	40	Pm-148m	56	Cm-244
9	Ru-106	25	Te-129m	41	Sm-151	57	Mn-54
10	Rh-103m	26	I-129	42	Eu-152	58	Fe-59
11	Rh-106	27	Cs-134	43	Eu-154	59	Co-58
12	Ag-110m	28	Cs-135	44	Eu-155	60	Co-60
13	Cd-113m	29	Cs-136	45	Gd-153	61	Ni-63
14	Cd-115m	30	Cs-137	46	Tb-160	62	Zn-65
15	Sn-119m	31	Ba-137m	47	Pu-238		
16	Sn-123	32	Ba-140	48	Pu-239		

Waste storage volume

As of the end of 2015, about 173 000 m³ of concrete and metal waste and about 85 000 m³ of felled trees have been stored at the Fukushima Daiichi NPP (TEPCO, 2016). Felled trees are produced by clearing areas for installation of facilities for water treatment facilities, tanks, etc. Contaminated water that is stored in the tanks and in the buildings at Fukushima Daiichi NPP has reached 788 541 m³. Approximately 1 million m³ of all forms of radioactive waste have been stored at the Fukushima Daiichi site. Details of the waste storage at Fukushima Daiichi NPP are provided in Chapter 7.

Summary

As a result of the Fukushima Daiichi NPP accident that followed from the tsunami, a significant amount of radionuclides were released. Radioactive waste currently generated at the Fukushima Daiichi site can be characterised as secondary waste from contaminated waste treatment, debris and felled trees. TEPCO is now promoting the R&D programme designed to accumulate analysis data to predict and/or assess waste characteristics and to evaluate the inventory.

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2. Regulator/implementer interaction

2.1. General description

The NEA Radioactive Waste Management Committee (RWMC) has been discussing the regulator-implementer dialogue as part of the process of developing a deep geological disposal facility (GDF). The importance of the process of interaction between regulator and implementer has been pointed out since the 1997 Cordoba Workshop of the RWMC. The overall objective of RWMC activities on the interaction between regulator and implementer was to explore diverse perspectives and expectations and to come to a common understanding of the main objectives and bases of the long-term safety criteria for disposal of the long-lived, high-level waste (NEA, 2014). There is no doubt about the importance of the dialogue between regulator and implementer because it is a part of a national decision process for developing and implementing a geological disposal system.

The RWMC decided to draft a questionnaire on this topic at the 47th plenary meeting in March 2014 and collected answers from the RWMC member countries before the 48th plenary meeting in April 2015 (NEA, 2015). This questionnaire focused on the issues and challenges on regulator-implementers' dialogue in the national programme of radioactive waste management, and included these items; i) factual position and decision framework, ii) mandated and voluntary dialogue, iii) international channels, iv) experiences and lessons learnt, and v) opportunity for other remarks.

Insights from the summary of the questionnaire of the regulator-implementer dialogue (NEA, 2015) are:

- In most countries surveyed, mandated regulator-implementer dialogue only takes place after the submission of a licence application from the operator. In many countries, however, the regulator and implementer have arrived at a voluntary process for information exchange and discussion, e.g. to clarify and resolve issues that both sides find useful. In Japan, such informal exchange is not allowed and this is considered problematic by the implementers.
- The independence of the regulator from the implementer is important, but this should be balanced with the promotion of dialogue between the regulator and the implementer, which is also important.
- A framework for pre-licensing dialogue is desirable – early and periodic review reduces the risk of later difficulties at the licensing stage.

After a nuclear accident or radiological contamination, it is assumed that the relationship between national regulator and implementer would change from the normal situation. An organisational restructuring might happen, the relationship with stakeholders may change, or the business base of the implementer may be affected after the accidents. A variety of situations might change. The dialogue between regulator and implementer is important even in normal situations, and even more so after an accident has occurred.

This chapter outlines some experiences on the dialogue between regulators and implementers after nuclear or radiological accidents, and lessons learnt from them.

2.2. Case studies

Ukrainian experience

Regulatory activity related to the shelter object

All activities at the shelter object, as well as at other radiation-hazardous objects should be carried out according to the laws, regulatory standards and rules in force in Ukraine that regulate safety in the field of nuclear power use. Moreover, it is also reasonable to apply recommendations of international organisations that do not contradict these laws, regulatory standards and rules in force in Ukraine to such activities.

State safety regulatory authorities shall establish regulatory and legal provisions for the activities at the shelter object whose consequences affect or can affect the safety of personnel, the public and the environment. State regulatory authorities shall make their decisions in accordance with the legislation, safety regulatory standards and rules in force, and taking into account assessments of the safety analysis performed by the operating organisation – the Chernobyl nuclear power plant (ChNPP). In particular, these authorities shall note the completeness of the application of safety regulatory standards and rules at the shelter object.

At present, specific regulatory requirements generally do not cover activities at the shelter object. In particular, a significant part of the technical requirements similar to the requirements that are established in normative documents (ND) in force for nuclear power plants and other objects is not established for the shelter object.

However, it is not reasonable to give specific technical requirements for the shelter object, and to stipulate these requirements as obligatory regulations, for the following reasons:

- To develop regulatory standards and rules for activities concerning the shelter object, it is important to have accurate data on the shelter object, on the activity underway there and on experience gained. There are no such data regarding the shelter object.
- The establishment of regulatory standards and rules provides that the shelter object and the activity there will be unchangeable as a whole during the long term. At the same time, the shelter object and activities there will be permanently transformed.

Thus, the development of strict requirements for the shelter object without due consideration of the data gained through the transition activities could be ineffective. A number of overly specific technical requirements established for the ND would not give “freedom of optimum choice for safety assurance” to the operator of the ChNPP. Moreover, a procedure for changing these requirements would take a long time and cause delays in ChNPP activities.

Another approach to safety regulation could be applied to this object. In this approach, the regulatory authority would establish general principles and criteria for ChNPP safety purposes and the ChNPP would have the possibility to independently select an optimum method for achieving these purposes taking into account these principles and criteria. The ChNPP should demonstrate in safety analysis reports that safety purposes are achieved by means of the selected method, and safety principles and criteria are observed.

Safety purposes, principles and criteria specified in the ND in force in Ukraine can be applied to the shelter object, taking into account the specific character of the object.

Moreover, these purposes, principles, and criteria are established in the documents dedicated specifically to the shelter object, namely:

- Law of Ukraine “On General Principles of the Further Chernobyl NPP Decommissioning and Destroyed Fourth Power Unit of this NPP Transformation into Ecologically Safe System”.
- “National Programme for Chernobyl NPP Decommissioning and Shelter Object Transformation into Ecologically Safe System”.
- “Shelter Implementation Plan of 31 May 1997”, approved in framework of the Law of Ukraine “On Ratification of Framework Agreement between European Bank on Reconstruction and Development and Ukraine Concerning Activities of “Shelter” Chernobyl Fund in Ukraine”.
- “National Strategy related Transformation of Shelter Object into the Ecologically Safe System”, approved in March 2001 by the Governmental Commission for Complex Solution of Chernobyl NPP Problems.
- Regulatory Bodies “Statement on the Regulatory Policy of Shelter Object Nuclear and Radiation Safety”.

The regulatory framework in Ukraine contains numerous different technical requirements which are directly associated with nuclear and radiation safety (NRS) or are significantly related to NRS. This legislative base consists of documents of the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) (the regulatory authority in the nuclear field), the Health Ministry, the Ministry of Ecological Issues, the Ministry of Construction and other regulatory authorities.

Technical requirements for structures, systems and elements of the Shelter Object and activities on their transformation should, on the one hand, be specific taking into account the uniqueness of the object as a technical system, and on the other hand should not differ in principle from the similar requirements established for NPPs and other objects. Hence, in activities concerning the shelter object, including the development of projects within the Shelter Implementation Plan (SIP), it is reasonable to use the regulatory framework in force with NRS requirements of a technical nature. Requirements should not be “mechanically” applied to the shelter object. However, the expediency of the application of specific technical requirements should be considered depending on specific SIP projects.

The aforesaid approach provides for an urgent solution for specific technical aspects of NRS assurance in co-operation with the regulatory authority and the ChNPP within the established licensing process.

The licensing process for SIP activities established depending on the aforesaid approach allows the ChNPP to obtain individual permissions for the implementation at the shelter object for each project developed within the SIP. In this way, co-operation with the regulatory authorities during the basic design stages ensures that the method for achieving safety purposes was selected correctly.

In this case, the key factor in safety regulation is not only regulation but also licensing. Wide application of licensing for activities on the shelter seems to be more effective than strict individual regulation or licensing at each step.

Licensing activity

- Licence that authorises shelter-related activities

The SNRIU issued a licence that authorised the ChNPP to carry out the SIP activities at the shelter object. The licence will be valid until commissioning of the new safe confinement (NSC).

The licence establishes certain conditions and rules for shelter-related activities. It requires, among other things, that the licensee develop, substantiate and agree that appropriate technical decisions (TD) are valid after their approval by SNRIU for any activity. Practices within SIP phase 2 designs may be implemented only on the basis of individual permits to be issued by the SNRIU.

The most important issue is to ensure safety during the implementation of SIP projects/designs. In order to ensure safety effectively, appropriate safety programmes and plans are required, which would cover the whole shelter object activity, including: a radiological protection programme; emergency planning; a programme for RW management generated during SIP. These programmes and plans should be submitted to SNRIU.

- Issuing separate permits

A procedure of interrelations between the licensee and the SNRIU, including issues of agreement or permits, has been determined in the regulation “Order for Separate Permits Issuing in the frame of Shelter Implementation Plan”. This document has been developed to effectively streamline the authorisation process at SIP phase 2 and to obtain SNRIU permits for specific projects/designs/operations at the shelter or its site. All permits to be issued by the ChNPP are described in the shelter object licence.

- Development of licensing plans for the SIP

The ChNPP has developed the “Licensing Plan for Shelter Implementation Plan Designs at Chernobyl NPP Phase 2” (LP). The LP was agreed by the SNRIU. The LP is aimed at providing effective SIP management as regards its licensing process and also ensuring the quality of the licensing document and timely agreement of documents by regulatory authorities. This document outlines certain interrelations between participants of the licensing procedure during SIP phase 2. It specifies types and a list of regulatory stages and types of documents to ensure the effectiveness of the authorisation activity.

Based on the LP mentioned above, a set of more detailed licensing plans were developed *inter alia* for such projects as the stabilisation of the shelter object, a new safe confinement, an integrated automated system of control and others.

Drafts of licensing plans have to be reviewed and approved by the all regulatory authorities involved in the licensing process.

Some aspects related to the supervision (inspection) of activity at the shelter object

Ukrainian legislation provides for state supervision of the ChNPP shelter object related activity. The goal of this activity is to check the NRS in the SIP process and assess adherence of this activity to shelter object nuclear and radiation safety improvement. The state supervision in accordance with this goal is organised and carried out pursuant to the specific procedure “Order for State Safety Supervision for Shelter Implementation Plan Designs”.

State safety supervision for individual shelter object stabilisation, reconstruction, modernisation and technical re-equipment designs and NSC design includes: i) inspection monitoring before construction (installation) and/or commissioning and operation; ii) inspections during construction (installation), commissioning and operation.

Inspection monitoring prior to construction (installation) and/or commissioning and operation is conducted to verify information provided by the operator (ChNPP) and the regulatory body (SNRIU) in relation to a permit for specific activity to ensure compliance with the actual status and conditions provided for the safety of this activity. Inspection monitoring, among other things, includes verification of detailed procedures described in design-engineering and operational documentation to ensure the NRS of personnel in construction (installation), commissioning and operation.

Inspections during construction (installation), commissioning and operation are conducted to check structures, systems, or equipment, activities and personnel qualifications for their compliance with NRS requirements. Inspections, among other things, include verifications of ChNPP documentation, status of work, preparedness of personnel to perform appropriate functions, direct monitoring of work, testing and measurements.

Functioning of the Joint Coordination Group for SIP Licensing

The licensing process is co-ordinated by the Joint Coordination Group for SIP Licensing (JCG). JCG consists of representatives of SNRIU and its technical support organisations and representatives of the ChNPP.

JCG is an organisation for online interaction between the operator (ChNPP) and the regulatory body (SNRIU) in the licensing process and should promote the efficiency and quality of this process. In doing so, the following main tasks are performed:

- analyse and assess the progress of the licensing process including the licensing plan;
- monitor schedules for submitting licensing documents to the SNRIU, including schedules for their review;
- solve procedural issues of the licensing process;
- reveal potential problems in the licensing process and determine ways to resolve them.

Lessons learnt

- A change in the regulatory approach is needed. Taking into account the uniqueness and peculiarities of the shelter object, the Ukrainian regulatory body had to define i) the status of the shelter object and ii) the activity to be licensed (to use the new term “transformation”/“conversion” of the shelter object into an ecologically safe system).
- The licensing approach was to issue one general licence and after that a number of separate permissions. The licence included a detailed description of the essence and contents of the activity, namely:
 - Activity related to maintenance of shelter safety through operation of all the needed systems and components identified in the technical specifications for shelter operation.
 - Activity related to transformation/conversion of the shelter into an ecologically safe system (the scope of this activity was defined in the initially approved revision of the “Shelter Transformation Strategy”).
 - List of permissions to be issued for implementation of the most significant projects (including stages for some of them).

- No need to establish specific safety standards for the shelter object because it is practically impossible to do for a “unique” facility such as a shelter object. The regulatory body shall establish safety objectives, principles and criteria for the ChNPP in general, and the ChNPP shall have the possibility to select independently an optimum method for achieving these objectives, taking into account principles and criteria. Also, the ChNPP should demonstrate in safety analysis reports that safety objectives are achieved by means of the selected method, and safety principles and criteria are observed.
- State supervision and inspections:
 - special procedures need to be developed taking into account the activity/projects to be implemented;
 - inspections before the start of the activity and during the implementation are needed;
 - the design documentation should be well received because of the variety of projects and different works on the site of the shelter object.
- The regulatory body and operator (ChNPP) put into practice the development of licensing plans, both general and detailed (for specific projects) in order to improve efficiency of the licensing process. The licensing plans were subject to review and agreement by not only the “nuclear” regulatory body, but also other regulatory bodies in accordance with the procedure envisaged by law. Considering implementation of activities at the shelter, it should be pointed out that development of licensing plans has been and remains useful from a practical point of view.
- The Joint Coordination Group between the regulatory body and operator proved to be equally important and efficient because it allowed immediate regulator–operator interaction during the licensing of different projects approved in the licensing plans.

US experiences

Debris removal regulatory approval strategy

At the time of the accident at Three Mile Island 2 (TMI-2), the major legislation applicable to regulation of commercial nuclear power in the United States consisted of the following articles:

- Atomic Energy Act of 1954 (as amended) – This act is the fundamental US law on uses of nuclear materials. It provides for both the development and the regulation of the uses of nuclear materials and facilities in the United States, declaring the policy that “the development, use, and control of atomic energy shall be directed so as to promote world peace, improve the general welfare, increase the standard of living, and strengthen free competition in private enterprise.” The act requires that civilian uses of nuclear materials and facilities be licensed, and it empowers the US Nuclear Regulatory Commission (NRC) to establish by rule or order, and to enforce, such standards to govern these uses as “the Commission may deem necessary or desirable in order to protect health and safety and minimise danger to life or property.” Commission action under the act must conform to the act’s procedural requirements, which provide an opportunity for hearings and federal judicial review in many instances.

- Energy Reorganization Act of 1974 – This act established the Nuclear Regulatory Commission. Under the Atomic Energy Act of 1954, a single agency, the Atomic Energy Commission, had responsibility for the development and production of nuclear weapons and for both the development and the safety regulation of the civilian uses of nuclear materials. The act of 1974 split these functions, assigning to one agency, now the Department of Energy, the responsibility for the development and production of nuclear weapons, promotion of nuclear power, and other energy-related work, and assigning to NRC the regulatory work, which does not include regulation of defence nuclear facilities. The act of 1974 gave the commission its collegial structure and established its major offices.
- National Environmental Policy Act of 1969 as amended – Every proposal for a major federal action significantly affecting the quality of the human environment requires a detailed statement on, among other things, the environmental impact of the proposed action and alternatives to the proposed action. The statement is to accompany the proposal through the agency review process. The act also established in the Executive Office of the President a Council on Environmental Quality, which has issued regulations on the preparation of environmental impact statements and on public participation in the preparation of the statements.

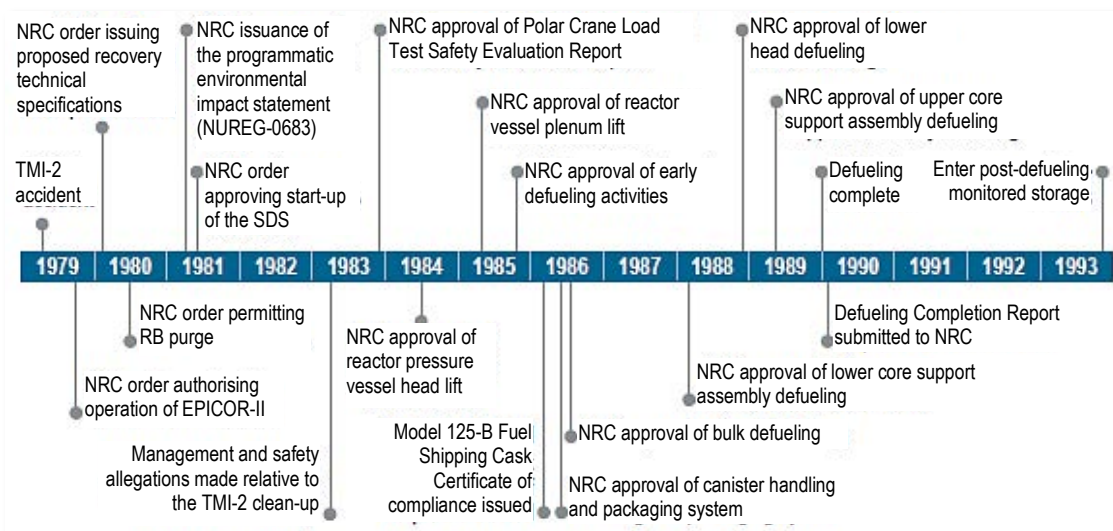
NUREG-0980 Nuclear Regulatory Legislation includes this legislation as well as other legislation affecting the regulation of commercial nuclear power in the United States. Although this legislation provided the basic framework for regulating nuclear power, more detailed provisions were required:

- To implement the requirements of the legislation, the NRC issued regulations under Title 10 of the United States Federal Code of Regulations Parts 1 through 199.
- To supplement the regulations and to provide licensees with methods acceptable to the NRC to meet certain regulations the NRC has issued Regulatory Guides.
- NRC reports information in NUREGs, which the NRC describes as reports on regulatory decisions, results of research, results of incident investigations, and other technical and administrative information.
- The NRC issues policy statements to inform the public of the commission's intent in taking an action.
- The NRC issues orders to pronounce commission decisions in adjudications and other matters before the commission.

At the time of the accident NRC regulations were not written with accident recovery in mind. As a result recovery from the accident at TMI-2 required a new approach to licensing and NRC oversight of a licensee. Therefore, the licensee and the NRC had to work closely together to develop a licensing approach for recovery from the accident. This approach consisted of significantly increased NRC on-site presence and oversight in the form of the NRC Three Mile Island Program Office (TMIPO).

The TMIPO had staff both at NRC Headquarters and at the Three Mile Island site; this greatly enhanced communications between the NRC and the licensee. As a result, during the recovery there were almost daily meetings and discussions between various members of the plant staff and the NRC to discuss plans, actions, activities, and mistakes and the proposed corrective actions. Additionally, there was step-by-step, in-line NRC approval of each new, major recovery activity. This programme is outlined in NUREG 0698 Revision 2 and discussed in more detail in the following sections, but the basic process was consistent throughout the recovery programme.

Figure 2.1 provides a timeline of the significant licensing-related events associated with the TMI-2 clean-up.

Figure 2.1. Timeline of licensing for the TMI-2 clean-up

Source: Bechtel, 2001.

NRC review and approval process development

■ 11 February 1980 Order

The first step in formalising an approval process for TMI-2 clean-up activities came on 11 February 1980 with an NRC Order issuing proposed “recovery technical specifications.” In addition to revising the existing TMI-2 technical specifications to recognise the post-accident condition at TMI-2, these specifications prohibited venting or purging of the reactor building atmosphere, discharge of water decontaminated by the EPICOR-II system, and the treatment and disposal of high-level radioactively contaminated water in the reactor building, until each of these activities was approved by the NRC, consistent with the NRC Commissioners Statement of Policy and Notice of Intent to Prepare a Programmatic Impact Statement.

In addition to these generic restrictions, the order also imposed a requirement that the NRC would approve certain procedures including Recovery Operations Plan (this section was titled Surveillances in Operating Plant Technical Specifications) and Recovery Mode (this was the newly defined condition of the TMI-2 Facility) Implementation Procedures.

The specific procedures affected by this requirement were those which:

- specifically related to core cooling;
- could cause the magnitude of releases to exceed limits established by the NRC;
- could increase the likelihood of failures in systems important to safety and radioactive waste processing or storage;
- alter the distribution or processing of significant quantities of stored radioactivity or radioactivity being released through known flow paths.

■ Programmatic Environmental Impact Statement (NUREG-0683)

As a result of litigation brought against the NRC during the licensing of the EPICOR-II System, on 21 November 1979, following start-up of the EPICOR-II system, the NRC issued a statement of policy announcing its intention to prepare a programmatic environmental impact statement (PEIS) for the “decontamination and disposal of radioactive waste

resulting from the 28 March 1979, accident at Three Mile Island Unit 2". The PEIS, prepared by the NRC, would satisfy NRC responsibility under the National Environmental Policy Act (NEPA). A contention in the litigation was that NEPA required a complete review of the Environmental Impact of an action, TMI-2 clean-up in this case, and not a piece-by-piece approach which would only look at incremental impacts, which are small but when taken in total could be large.

After an extended period of development and public consultation the PEIS was issued in March 1981. In its Policy Statement accompanying the PEIS, the commission directed the staff to determine whether specific licensee clean-up proposals and the associated potential impacts fall within the scope of those already assessed in the PEIS. With the exception of accident-generated water disposal which the commission reserved to itself, if the proposed actions were within the PEIS scope and any supplements, the Director, TMIPO, has been delegated the approval authority, while keeping the commission informed of the staff's actions on each major proposal. If the licensee's proposal was not within the PEIS scope, the commission is notified and additional reviews by the TMIPO staff are undertaken in accordance with the NEPA. The staff, based on an environmental and safety review, makes a recommendation on the proposed action to the commission. With the exception of accident-generated water disposal which the commission treated as a special case, the commission review option did not have to be exercised.

The Policy Statement further stated that at any time the staff determines that the conclusions presented in the PEIS have substantially changed, the staff would issue a supplement revising the PEIS in accordance with NEPA. During the course of the clean-up programme three supplements were issued to the PEIS. Supplement 1 was issued in October 1984 prior to the start of defueling as the original estimates of occupational radiation exposure were believed to be too low and did not bound the expected doses to personnel for the remainder of the clean-up process. Supplement 2 was issued in June 1987 and addressed the disposal of accident-generated water and Supplement 3 was issued August 1989 and addressed the completion of the clean-up programme and placing TMI-2 in post-defueling monitored storage.

The issuance of the PEIS simplified the review process for TMI-2 clean-up actions, the approval process prior to issuance of the PEIS were lengthy and litigious and involved multiple layers within the NRC. With the issuance of the PEIS and the accompanying NRC Policy Statement approval for clean-up activities was delegated to the TMIPO. This action is likely to have greatly accelerated the clean-up.

▪ NRC Plan for Cleanup Operations at TMI-2 (NUREG-0698)

NUREG-0698 was first issued in July 1980 and as the name suggests details the process used by the NRC to review and approve clean-up activities.

As described in NUREG-0698 the NRC described its role as being responsible for the regulation of IMI-2 clean-up operations to ensure the protection of the health and safety of the public and the environment. For all post-accident operations at TMI2, the NRC stated the following regulatory objectives:

- maintain reactor safety and reactor building integrity;
- ensure that environmental impacts are minimised, and that radiation exposures to workers, to the public, and to the environment are within regulatory limits and are as low as reasonably achievable (ALARA);
- ensure the safe storage and/or disposal of radioactive waste from clean-up operations.

The NUREG went on to state that implementation of clean-up activities was the responsibility of the licensee. Each revision of the NUREG discussed which actions it had approved, which actions were under review and which actions it expected to approve

prior to implementation. Each revision also described the interface with other Federal and Commonwealth of Pennsylvania agencies. This document provides a window on the NRC mindset with respect to its role and responsibilities at TMI-2.

Revision 0 to NUREG-0698 discussed the procedures which required NRC approval as described in the February 1980 Order, previously discussed. For review of clean-up actions proposed by the licensee it noted that a PEIS was in the process of being written and discussed in very general terms how clean-up actions will be approved once the PEIS was issued. The NUREG also noted the length of time required to approve the EPICOR-II and reactor building purge applications, discussed in more detail later, and opined that with the issuance of a PEIS approval of subsequent applications could be performed more quickly.

Revision 1 to NUREG-0698 was issued in February 1982 although the role of the NRC had not changed it updated the NUREG to reflect the authorisation given to the TMIPO for approval of clean-up activities by the commission following the issuance of the PEIS. In addition the NUREG also contained a copy of the original Memorandum of Understanding (MOU) between the NRC and US Department of Energy (DOE) for the disposition of waste resulting from the clean-up of TMI-2 (this MOU is discussed in the debris transportation regulatory strategy).

As described in the revision to NUREG-0698 clean-up actions proposed by the licensee and the appropriate level of TMIPO review of these actions fell into two categories.

1) If the proposed action involved a request for a licence amendment or an unreviewed safety question, the TMIPO staff would first determine if it was within the scope of the PEIS. A proposed clean-up activity would be considered to be within the scope of those already addressed in the PEIS if the following conditions were satisfied:

- The proposed method was similar to the general type of activities discussed in the PEIS for the clean-up and/or disposal of radioactive waste from the TMI-2 facility.
- Its potential environmental impacts were not significantly different (qualitatively and quantitatively) from those environmental impacts associated with this type of activity as assessed in the PEIS.

In addition to the PEIS scope of review, a significant hazards determination was performed by the TMIPO staff, in accordance with 10 CFR 50.91 and 92 (see the section on key NRC regulations affecting approval of clean-up activities details of this requirement) and a safety evaluation was prepared. If significant hazards were found to exist, an opportunity would have been given for a public hearing prior to approval of the proposed action. With the exception of accident-generated water disposal, a prior public hearing was not required for any clean-up activity. In accordance with NRC regulations, if no significant hazard exists, a notice for an opportunity of a hearing prior to approval and implementation of the proposed action, would not be given. For either case, TMIPO review of the proposal would be accompanied by its review and approval of the procedures to implement the proposed activity as required by the technical specifications promulgated by the February 1980 Order discussed previously.

If it was determined that any major activity or predicted environmental impacts fell outside the scope of those already assessed in the PEIS, the TMIPO staff would complete the necessary reviews in accordance with the NEPA and NRC requirements, as described above three supplements to the PEIS were eventually published. If it was determined that a supplement to the PEIS was appropriate, the supplemental environmental statement will be prepared under the direction of the TMIPO. In the event a proposed activity falls outside of the scope of the PEIS, but does not require the preparation of a supplemental environmental impact statement, the TMIPO staff would have publish a negative declaration to that effect and provided an environmental impact appraisal in support of the negative declaration. Action on proposals which are outside the scope of the PEIS will

be taken by the commission itself, as described above except for the special case of accident-generated water disposal this course of action was never needed to be followed.

For the three supplements to the PEIS an opportunity for the review of a draft supplement was afforded the public during a defined comment period. Other government agencies having an interest in the review, monitoring, and in some cases, participation in some phases of the proposed clean-up operation were also involved in the review of the supplement to the PEIS.

2) If the action, although major, did not involve the need for a licence amendment and the action did not involve an unreviewed safety question as described in 10 CFR Part 50.59, the TMIPO performed a safety review of the proposal and approved the detailed implementation procedures prior to implementation. In this case, the TMIPO review also determined if the proposed action and its potential environmental consequences were within the scope of that discussed in the PEIS. If they were outside of the scope of activities evaluated in the PEIS, the TMIPO would have proceeded with the review in accordance with NEPA and NRC requirements as outlined in above, however, this option did not have to be exercised during the clean-up programme as all actions were determined to be bounded by the PEIS.

Revision 2 of NUREG-0698 was issued in March 1984 and was issue to describe changes in the functional role of the NRC in clean-up operations, the clean-up schedule, and the current status of the clean-up activities and provided a revision to the MOU between the DOE and NRC on waste disposal. The fundamental NRC review and approval process however remained unchanged from Revision 1.

- Key NRC regulations affecting approval of clean-up activities

Two regulations would have significant impact on obtaining NRC approval for clean-up activities. The first of these regulations was 10 CFR 50.59 Changes, Tests and Experiments. Paragraph (a) of the regulation, which was in place during the TMI-2 clean-up, stated:

(a) (1) The holder of a license authorizing operation of a production or utilization facility may (i) make changes in the facility as described in the safety analysis report. (ii) make changes in procedures as described in the safety analysis report and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval unless the change, test or experiment involves a change in the technical specifications incorporated in the licence or an unreviewed safety question (emphasis added).

(2) A proposed change test or experiment shall be deemed to involve an unreviewed safety question (i) if the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be increased; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The second of these regulations did not exist at the time of the accident but were brought about as a result of litigation brought against the NRC during approval of the reactor building purge. These regulations, 10 CFR 50.91 and 50.92 described the process for amending a licence; this would include changes to a technical specification or action on an unreviewed safety question. The key component of this regulation is the determination of whether a public hearing must be held prior to approval, termed a “no significant hazards determination”. If a proposed amendment would not:

- involve a significant increase in the probability or consequences of an accident previously evaluated;

- created the possibility of a new or different kind of accident from any accident previously evaluated;
- involve a significant in a margin of safety.

Then a prior public hearing was not required and the NRC could issue the change even if adverse public comments have been received or a request for a hearing has been filed.

▪ Licensee impacts and response

The elements of the NRC review and approval process described above defined how approval for the clean-up process would proceed. For each major clean-up action, as described in NUREG-0698, the licensee would need to seek NRC approval even if an unreviewed safety question as defined by 10 CFR 50.59 was not involved. The TMIPO could approve the activity if it was bounded by the PEIS. After an activity was approved the procedures needed to implement the activity would also require NRC approval as required by the technical specifications imposed by the February 1980 Order.

As a result a Safety Evaluation Report (SER) was prepared, by the licensee, for each major clean-up activity and a Technical Evaluation Report (TER) was prepared for each system. In preparing these documents beyond the basic description and safety analysis two important sections were added.

The first of these sections was the 10 CFR 50.59 review to determine if a technical specification change was needed or an unreviewed safety question was involved. If neither of these concerns applied then the TMIPO could simply issue an approval via a letter after their review was completed. However, when either of these concerns applied then the licensee amendment process would need to be invoked. During the course of the clean-up careful attention was paid to these aspects such that if either of these concerns existed a licensee amendment request, also known as a technical specification change request, could be submitted to the NRC prior to submittal of the impacted SER or TER so that the SER/TER could reference that request. In this manner only a specific issue would be open to public comment and not the entire programme described in the SER/TER.

The second of these sections was an environmental assessment which would demonstrate that the SER/TER was bounded by the PEIS. As described above if the action was bounded by the PEIS then the TMIPO could approve it, if it was not bounded then a further review by the commission would be required. Thus, in the planning of the clean-up activities a careful assessment was made to ensure the action was bounded by the PEIS. Only two clean-up actions were not bounded in the PEIS the first was accident-generated water disposal which the NRC treated as a special case as described elsewhere in this document, the second was for the entry of TMI-2 into post-defueling monitored storage as the end of the clean-up process was not considered in the original PEIS.

Thus, by careful consideration of these two aspects in the planning process and ensuring that, to the extent possible, that the proposed activity was bounded by existing safety analyses and the PEIS the administrative burden on the TMIPO was reduced and the time required to obtain NRC approval to perform an activity was reduced to the extent possible.

▪ Prelude to core debris removal

In order to prepare to remove the core debris from the TMI-2 reactor a multiple step incremental approach was taken in the licensing process. This approach was commonly referred to as taken a bite of the elephant (from an old joke "How do you eat an elephant?" "One bite at a time"). Each new licensing submittal built on the knowledge gained from the previous action. The pre-core debris removal licensing followed two main paths "core debris removal preparations" and "remote characterisation". Each submittal built on knowledge gained from previous activities.

Special nuclear material accountability and criticality safety analysis

■ Introduction

Following completion of defueling the licensee needed to account for the special nuclear material (SNM) remaining in the reactor vessel and to demonstrate that there was no potential for a redistribution of the residual material in the reactor vessel that could lead to a criticality. The purpose of this section is to describe the TMI-2 SNM Accountability Program and summarise the criticality safety analyses presented in the TMI-2 DCR Defueling Completion Report and in TMI-2 letter, C312-92-2080, dated 18 December 1992. This section identifies the methods and sequence of events for residual SNM accountability; the Quality Assurance Programme applied to the SNM measurements; the areas, systems and components that were assessed for residual quantities of SNM; and the areas, systems and components that did not require SNM assessment.

The quantity of fuel (i.e. UO_2) remaining at TMI-2 is a small fraction of the initial fuel load. As a result of TMI-2 defueling and decontamination activities, approximately 99% of the fuel was removed and transferred to the DOE and/or licensed burial facilities (note >99% fuel removal was a goal the requirement was to ensure there was no possibility for a criticality anywhere in TMI-2).

The final results of the SNM Accountability Program are based on a comprehensive post-defueling survey of the TMI-2 facility. The post-defueling survey consisted of a review of the TMI-2 plant to identify areas that could contain SNM and areas unlikely to contain SNM. The quantity of SNM was determined in each area that was identified to have SNM present. This section describes the process by which the post-defueling survey was conducted and summarises the results of the survey.

Finally, this section presents a summary of the criticality safety analyses which demonstrated that a criticality event could not occur in TMI-2.

■ Background

The March 1979 accident resulted in significant damage to the reactor core with a subsequent release of fuel and fission products into the reactor coolant system and other connected systems. The core was reduced to fractured fuel pellets, resolidified fuel masses, structural metal components, loose rubble and partial fuel assemblies. The generic term used to refer to the post-accident core material is core debris.

The core debris removed from the TMI-2 facility was loaded into special canisters for shipment to the DOE Idaho National Laboratory facility in Idaho. Each shipment was accompanied by a Nuclear Material Transaction Report (DOE/NRC Form 741) which recorded the net weight of the contents of each canister. Fuel accountability by the normal method, i.e., accounting for individual fuel assemblies, was not possible. Since the canisters were filled with a mixture of SNM, other materials, and water, there was no practical or feasible method to determine the exact SNM content in each canister. A statement to that effect was included on each DOE/NRC Form 741.

In October 1985, GPU Nuclear, the DOE and NRC entered into an agreement that final SNM accountability for TMI-2 would be performed after defueling was completed. The accountability would be based upon a thorough post-defueling survey of TMI-2 which would quantify the amount of residual SNM in plant systems and components. Implied in this agreement was an understanding that the post-defueling survey would involve all areas, structures, systems and components where SNM could reasonably be suspected to have been deposited as a result of the 1979 accident and subsequent clean-up activities.

■ SNM accountability process

The entire TMI-2 plant was reviewed to determine where SNM could have been deposited as a result of the 1979 accident and subsequent clean-up activities. Each area was classified into one of three categories:

- Category 1 – Locations where SNM was highly probable;
- Category 2 – Locations where it was possible that SNM could be deposited;
- Category 3 – Locations where it was unlikely that SNM was deposited.

Category 1 locations required that measurements or, in selected cases, analysis, be performed for SNM. Category 2 areas were considered to have a lower probability for fuel deposits, but were assessed in the same manner as the category 1 areas. Category 3 areas were determined not to require SNM assessment based on analyses of the TMI-2 accident and review of clean-up activities.

■ SNM accountability methods

SNM accountability for TMI-2 was completed in accordance with the SNM Accountability Plan. Several plant areas and components were characterised for SNM deposits prior to initiation of the formal SNM Accountability Program. In some cases, ALARA considerations, the quality of the previous measurements, and lack of actions potentially affecting SNM deposits warranted their use. These measurements were independently reviewed to ensure sufficient data existed to meet SNM accountability quality assurance standards. In all cases, the quantity of residual SNM was determined through measurements, sampling, inspection, or engineering analysis. The NRC contracted with the Pacific Northwest National Laboratory to perform an independent assessment of the SNM accountability programme which includes reviews of methods and calculations and independent measurements in a few locations in TMI-2 to ascertain the quality of the programme.

Measurements

In most cases, measurements were performed in individual locations after planned clean-up activities were completed within the area. In some areas, as stated above, it was determined that the clean-up activities did not materially affect the original SNM measurements which were then used for SNM accountability. The post-defueling survey required the application of several measurement techniques. Technique selection for an individual measurement depended upon the geometry of the component/system or area to be assayed, physical access limitations, radiological conditions, personnel exposure considerations and the probable quantity of SNM in the area. Where required or desirable, the measurements also involved use of more than one measurement technique. Since the final SNM accountability activities were classified as “important to safety”, measurements conducted for SNM accountability were performed using quality assurance-approved procedures.

Gamma scintillation spectrometry using sodium iodide detectors accounted for the majority of the early work. Later measurements involved the use of high-purity germanium detectors, which allowed greater resolution for the tracer isotopes of interest. Other measurement techniques included alpha scans using proportional detectors and gross gamma measurement techniques using collimated Geiger-Mueller detectors. The endfitting and dry reactor vessel measurements were completed using neutron interrogation techniques.

Sampling

To obtain additional isotopic and volumetric information for use with the other analysis techniques, sampling of suspected fuel locations was performed. Solid and liquid samples were obtained from various areas and components to obtain isotopic,

composition, and density data for use with measurements and visual inspections. Scrape samples were taken of metal surfaces (i.e. manways, piping, filter housings) to determine film depositions. These samples were analysed using either on-site or off-site facilities, applying quality assurance-approved procedures.

Visual inspection

In areas where measurement was not practical, video camera probes were used to estimate the volume of material remaining in the subject area. Using the volumetric data generated through sampling, a fuel quantity was assigned.

Engineering analysis

In the latter part of the project, several areas that had not been measured were estimated using a flow-path analysis. The flow-path analysis was performed by examination of possible SNM introduction pathways into an area through plant systems during the accident or subsequent clean-up activities.

Documentation

The quantity of residual SNM in each location was documented in a GPU Nuclear engineering calculation. The engineering calculations were based on geometric configuration, analysis of the measurement data, instrument calibrations, capabilities and performance. Also included in the calculations were any specific assumptions made based on review of earlier measurements and the relevant history of that location during the accident and clean-up. All SNM engineering calculations were produced and approved in accordance with approved procedures.

The engineering calculations, in turn, provide the quantity of SNM for a specific area, system or component that is outlined in the post-defueling survey reports (PDSRs). Each PDSR contains:

- a detailed description of the area, system or component;
- its role in the accident and/or clean-up activities;
- the rationale supporting a conclusion as to whether contained residual SNM exists and, if so, a summary of the appropriate SNM engineering calculations;
- applicable photographs and/or drawings of the area;
- an assessment of residual fuel.

The PDSRs were forwarded to the NRC. The completed PDSRs formed the basis for the final TMI-2 SNM inventory.

■ Final SNM Accountability

Final accountability was performed by summing the residual fuel quantities identified in the PDSRs and reporting the results as the remaining plant inventory of special nuclear material. The amount of fuel shipped to the DOE Idaho National Laboratory was determined by subtracting the sum of the final plant inventory and the amount of SNM shipped as radioactive waste from the pre-accident plant inventory of SNM, as corrected for decay in the most recent SNM Material Balance Report.

Pre-accident reported inventory (corrected for decay)

- Final in-plant inventory
- SNM shipped as samples/radwaste

= SNM shipped to Idaho National Laboratory (in canisters)

The resulting SNM inventory was reported on the PDMS SNM Material Balance Report (DOE/NRC Form 742). This was the method used to demonstrate to the NRC that approximately 99% of the original TMI-2 core had been removed from the site.

- **Criticality analysis**

The inherent criticality safety of the residual fuel during the PDMS period has been demonstrated in TMI-2 letter, 4410-90-L-0012, "Defueling Completion Report, Final Submittal", dated 22 February 1990 and by GPU Nuclear letter, C312-92-2080, "TMI-2 Reactor Vessel Criticality Safety Analyses", dated 18 December 1992 which evaluated reactor vessel (RV) subcriticality based on an increase in the estimated RV fuel inventory due to the neutron interrogation method used during final reactor vessel draindown from the original visual measurements following defueling. The criticality analyses addressed the quantity of residual fuel in each defined location and the potential for fuel relocation. The analyses estimated the quantity of fuel remaining, its location, its dispersion within the location, its physical form (i.e. film, finely fragmented, intact fuel pellets, resolidified), its mobility, the presence of any mechanism that would contribute to the mobility of the material, the presence of any moderating or reflecting material, and its potential for a critical event. Each issue was addressed to the extent appropriate for a given quantity of fuel. The NRC staff concurred with the criticality analyses presented in the defueling completion report (DCR) via their 26 April 1990 letter stating "no objections" to the TMI-2 transition from Facility Mode 1 to Facility Mode 2.

As stated above, a reanalysis of the RV steady state and accident criticality safety evaluations was necessitated by an increase in the estimated quantity of fuel remaining in the RV above that assumed for the DCR. A conservative criticality model was used to bound the most credible fuel configuration.

These analyses have demonstrated that criticality has been precluded as a result of the extensive TMI-2 defueling effort. This conclusion was based on three evaluations: the safe fuel mass limit determination, the bounding reactor vessel steady state criticality calculations, and the potential for criticality under accident conditions. In fact, it was demonstrated that no physically achievable quantity of residual core debris could result in a critical fuel configuration. Therefore, criticality is precluded for all credible conditions. Although not needed to assure reactivity control over the long term, as an additional conservative measure, a stable and insoluble neutron poison, consisting of 1 400 lbs of Boron Silicate glass shards, was added to the bottom head of the RV.

Control of SNM during post-defueling monitored storage

Control of SNM at TMI-2 during PDMS relies upon isolation boundaries and control of access to components which contain SNM. Isolation boundaries will be maintained, as necessary, to prevent relocation of significant SNM quantities. The reactor coolant system, which contains the largest quantity of SNM, was drained to the extent practical and isolated within the containment building.

There will be no physical inventory of SNM quantities at TMI2 during PDMS because the remaining materials are of low enrichment, highly radioactive and relatively inaccessible. The NRC has granted TMI-2 an exemption from the 10 CFR 70.51(d) physical inventory requirements. However, any shipments of accountable quantities of SNM from TMI-2 during PDMS will be reported as required on DOE/NRC 741 Nuclear Material Transaction Reports.

Conclusions and lessons learnt

The establishment of on-site regulatory presence helped improve the timeliness of regulatory actions. The licensee and the regulator need to work closely together to develop a licensing approach for recovery from an accident.

During the accident recovery almost daily meetings and discussions between various members of the plant staff and the regulator to discuss plans, actions, activities, and mistakes and the proposed corrective actions are valuable for smooth recovery process.

An overall analysis of the accident recovery is needed early in the recovery process. Then breaking the accident recovery into major activities for specific regulatory approval allows the safety analysis for the next phase to build on the learnings from the previous.

The ability to approve recovery activities should be delegated to on-site regulatory presence to the extent allowable under national regulation. This action is likely to accelerate the clean-up.

Fuel accountability by the normal method, i.e., accounting for individual fuel assemblies, is not possible following an accident. Containers filled with a mixture of SNM, other materials, and water, provide no practical or feasible method to determine the exact SNM content in each canister thus a more accurate accountability can be determined by measuring fuel remaining after the conclusion of the defueling process.

Norwegian experiences

Regulator view of regulator/operator interaction

For legacy situations as they are described in EC (1999), but also in the case of decommissioning of facilities which have suffered severe accidents, the existing requirements may not be adequate to address abnormal and unplanned circumstances.

The range of radiation and nuclear safety and security issues arising at accident and legacy sites is very large, encompassing issues of worker, public and environmental protection, in planned, existing and accident exposure situations. The condition of many of the facilities and materials such as spent nuclear fuel (SNF) and radioactive waste (RW) may not be in compliance with either original requirements or requirements as they exist today. That is to say, the situation at these sites is, generally speaking, abnormal. Therefore, even planned situations require special consideration and the development and application of new techniques and corresponding regulatory requirements and guidance.

A typical situation that can arise is that in order to avoid continuing degradation of an already poor storage facility, a hazardous operation has to be undertaken. Proper planning can reduce the risks and associated with the remediation operations, and, while not completely eliminating all risks, bring them to within acceptable limits. However, the nature and scale of the existing hazard may indicate a degree of urgency. Early action may reduce continued degradation and avoid possible acute releases from acute failure of containment. However, the remediation action itself, may create its own accident risks, and lead to exposure of workers, or generate effluent discharges affecting the public and the environment, or generate a much larger volume of radioactive waste, or all of these things.

To solve this problem requires an effective prognostic assessment capability that allows the implications of different management alternatives to be evaluated. In turn, this has to rely on sufficient characterisation of the source terms and of the environments into which radioactive material may be released. However, it also needs to rely on clear and coherent guidance on radiation risks and their control within the context of all the other legacy issues. Apart from radiological protection and radioactive waste, it has to be recognised that there are other physical and pollution hazards to take into account, such as asbestos, heavy metals and organic compounds.

As well as the generally understood issues of environmental and human health protection, there are also legitimate concerns over security, including the control of large sources and nuclear material. The security aspect adds additional constraints to the selection and justification of appropriate management decisions. The resolution of many

of these legacy issues involves military and civilian authorities, including those involved in safety, security and environmental and human health protection.

To complete the picture, it is necessary to mention the challenge of fitting the management of these accident and legacy sites and situations into still evolving international recommendations on radiological protection, waste management and so on, which in turn have implications for nationally based regulatory requirements. In Russia, for example, there is a new Federal Law on the Management of Radioactive Waste, which was adopted by the State Duma on 29 June 2011, and approved by the Council of the Russian Federation on 6 July 2011. Further regulatory experience and lessons learnt have been explored at an international workshop on “Regulatory Supervision of Legacy Sites: from Recognition to Resolution” (Sneve and Strand, 2016).

The range of disciplines and relevant experts involved is very large. NRPA staff and their colleagues in sister organisations in Russia take the view that opportunities for cross co-operation between regulators have been relatively limited in the past and should be increased. Practical work should be encouraged to improve such opportunities through joint technical meetings:

- between managers and shop workers;
- among different operators – e.g. waste producers and waste disposal organisations;
- between operators and regulators;
- among nuclear safety regulators, radiological protection regulators and other pollution and safety regulators;
- among scientists, policymakers and wider stakeholders; and among all of those mentioned above.

The material above seeks to illustrate the complexity of legacy management and the challenges that complexity raises for regulatory authorities. One of the starting points for discussion within the IAEA Regulatory Supervision of Legacy Sites (RSLs)¹ was to understand what is meant by a (nuclear) legacy site, bearing in mind that the IAEA safety and waste glossaries do not mention legacies. The approach was taken at RSLs to be inclusive and to adopt a working definition that a legacy site is a facility or area that has not completed remediation and is radioactively contaminated at a level which is of concern to regulatory bodies. The status of a site has implications for its radiological supervision, as regards, for example, whether exposures at a site should be considered as existing or planned exposures, which in turn affects the application of reference levels (Hedemann-Jensen and McEwan, 2011).

The NRPA’s bilateral co-operation experience has shown that it is vital that each regulatory authority has its own clear lines of responsibility and for those lines of responsibility to be clearly communicated to all stakeholders. This should be obvious, but

1. The IAEA has been implementing a number of actions associated with legacy sites. To support the integration of these efforts, from a regulatory perspective, the International Forum for the RSLs was set up by IAEA in 2010. Through resolution GC (54)/RES/7, the IAEA General Conference has endorsed the creation of the RSLs and encourages member state participation. In the context of the RSLs, regulatory supervision refers to the full scope of activities that regulatory authorities would be engaged with for legacy sites (e.g. regulations, review and assessment activities, licensing, inspection and public outreach). Whether Fukushima Daiichi is considered as a legacy or has some other status is not the critical issue. What is critical is that the factors affecting regulatory supervision at Fukushima Daiichi NPP are just the same as at these other legacy sites.

our experience shows that it takes some time to develop a common understanding of accident and legacy situations, so that those responsibilities can be recognised.

Optimisation is a major feature of radiological protection and its regulation. It includes consideration of economic and social factors, which raises difficult questions such as:

- How should regulators include economic and social factors in its decision-making process, without being or appearing to be involved in political issues?
- How practically can you separate the scientific and the social value judgements?
- How should a regulator compare or balance short and long-term risks to different groups of people, which assessment of options present as alternatives?
- Assuming that one adopts the current advice not to use assessment of radiation doses to estimate health effects, how can a regulator, or anyone else, compare the radiological consequences with other human health consequences associated with legacy remediation options?

These questions raise their own challenges regarding the regulatory decision-making process, such as:

- Development of consistent protection objectives and regulatory approaches for radioactive and other contaminants, for humans, non-human biota and in special areas such as groundwater protection.
- Corresponding development of consistent derived standards relevant to those protection objectives, and approaches for their assessment.
- Development and application of transparent methods to support decisions on choices between options, and maintaining a balanced and proportionate response to all risks.
- Improved communication of uncertainties so that decisions are taken on a risk-informed basis.

Prospective radiological assessments are used to assess outcomes of alternative management options for legacies. Bearing in mind the uncertainties involved, through what process does a regulator evaluate the adequacy of:

- waste and contamination characterisation;
- site and environmental characterisation.

In addition, the appropriate protection objectives to apply in a particular exposure situation can be difficult to determine, i.e. it is not always clear in any particular case if it is an emergency, existing or planned radiation exposures. This is acknowledged in the IAEA Basic Safety Standards and the issue is discussed further in Sneve and Smith (2014) and Copplestone et al. (2016).

Lessons learnt

- Encouraging early interaction between the operator and regulator for timely, safe and effective decommissioning, especially in the case of decommissioning after an accident.
- Closely linking the decommissioning strategy and programme and developing it jointly with, the waste management strategy and programme.
- Considering that early decisions on remediation, without due consideration or final disposal, can make final disposal more difficult.

- Applying normal regulatory requirements and procedures is preferable, as far as possible, for example as set out in the Phase 1 NEA report (NEA, 2013) and earlier documents on decommissioning (NEA, 2003). So a key question to ask is, “What characteristics of the Fukushima Daiichi NPP accident waste are such that they cannot fit into normal requirements and procedures?”
- Taking into account the current Japanese safety requirements and waste acceptance criteria (WAC) for interim storage and disposal, and considering whether the waste arising from Fukushima Daiichi NPP decommissioning fits comfortably within that categorisation. If it does not, it should be asked how it is different and which, if any, regulatory or other changes to specifications are needed.
- Taking into account transport regulations and packaging requirements. An early question to address is if new package types will be needed.
- Considering who in Japanese system will be responsible for development of WAC for storage and disposal, and what will be procedure for their development? For example, concerning deep disposal in the United Kingdom there has been joint development, but technical development was led by the organisation responsible for managing radioactive waste, not the waste producer, nor the regulator. However, the waste management organisation sought approval from the other organisations.
- Being aware of the use of fingerprint techniques developed in normal situations, which may not be applicable to accident waste.

Japanese experiences

Regulation for waste from ordinary operation and decommissioning of NPPs

- Law and rules for radioactive waste management

Japanese regulation for waste disposal from ordinary operation and decommissioning of NPPs are outlined in the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (hereinafter referred to as the act), the category 1 waste disposal rule, category 2 waste disposal rule, the technical standards notification, etc.

- Regulation for the each operational stage

Interaction between the regulator and the implementer starts from the safety review of the licensing application. At this stage, siting, basic design, capability of the implementer and safety assessments during and after the operational period are reviewed. At the operational stage, safety confirmation of waste packages is performed, in which each waste package is shown to comply with the technical standards and their application. Concentration of radionuclides of each waste package is also confirmed. So-called scaling factors (ratio of composition between a hard-to-measure nuclide and a key nuclide) for each NPP are set by the implementer and confirmed by the regulator beforehand.

Safety confirmation for a waste disposal facility is also performed at the appropriate step of the construction of the pit or the cover, in which the disposal facility is shown to comply with the technical standards and the design shown in the application.

During this step, environmental dose rates and radionuclide concentrations are monitored, and the preservation plan is reviewed. A periodic safety review is performed reflecting those monitoring data and the state-of-the-art knowledge in ten years, and is reported to the regulator.

The implementer is to perform institutional control for 300 to 400 years after the closure of the disposal. Preservation work of the disposal facility and the periodic safety review are to continue to the step of the termination of the licence. At this step,

remediation work is to be performed if it is necessary. The decommissioning plan, including the safety assessment at that time, is to be reviewed at the step of the termination of the licence.

With these regulations, basic design before the operational step, adaptabilities of the waste forms and disposal facilities at the operational step, monitoring data and the periodic safety review and the final safety assessment at the decommissioning step, are reviewed and permitted or confirmed by the regulator.

▪ The framework of the regulation of the radioactive waste disposal

The framework of the regulation of the radioactive waste disposal is as follows.

1) The safety concept of the waste disposal

In article 51-3 of the act, it is written as “the location, structure and equipment of the waste disposal facilities or waste storage facilities are such that they will not hinder the prevention of disasters resulting from nuclear fuel material or material contaminated with nuclear fuel material.”

The dose limit for trench and pit type disposal (shallow land burial) is listed below.

- Operational period and active institutional control period after the closure of the disposal:
 - normal condition : 50 $\mu\text{Sv}/\text{y}$;
 - accident condition: 5 mSv/y .
- After the termination of the licence (i.e. 300 to 400 years after the closure of the pit type disposal):
 - Likely scenario with most probable condition: 10 $\mu\text{Sv}/\text{y}$;
 - Less likely scenario with range of uncertainty: 300 $\mu\text{Sv}/\text{y}$;
 - Other natural scenario and human intrusion scenario: 1 mSv/y .

2) Safety requirement

To ensure the safety concept, safety requirements are imposed:

- Design requirement for disposal – prevention of the dispersion, shielding, containment and control the migration:
 - as a control requirement, prevent personnel from restricted area, patrol and inspection;
 - consideration of natural events for siting and design (earthquake, tsunami, fire).
- Design requirement for waste package – uniaxial compressive strength, containment.

3) Technical standards

To ensure the safety requirement, technical standards for some barriers are established:

- Disposal facilities: prevention of seepage of rain water, prevention of dispersion, cover soil, structural bearing force, area of opening of facilities, volume of facilities.

4) Confirmation of waste package and disposal facilities

It must be confirmed that each waste package complies with the technical standards and that the radioactive concentration of each nuclide is below the maximum permitted.

5) Safety review

At the safety review, it is checked that siting and design of the disposal comply with the safety requirements and technical standards. Safety assessment should show that the effect to the public is below the dose limit during the operational period, institutional period and after the termination of the licence.

■ Regulation for the Fukushima Daiichi accident waste management

The Fukushima Daiichi NPP is designated as the “specified nuclear power facilities pursuant to the article 64 2 paragraph 1”. The Act Specified Nuclear Power Facility System is a designation system for controlling nuclear facilities in an appropriate manner responding to the prevailing circumstances at the facilities where nuclear emergencies occur. The Nuclear Regulation Authority (NRA) indicated TEPCO the “items required for measures” and directed TEPCO to submit an “implementation plan”. The NRA is to control the facility based on the “implementation plan”. The NRA is to arrange appropriation of the Nuclear Regulation Act relating to the specified nuclear power facilities according to the government decree defining the special cases (such as exception of appropriation of the law).

With this system, permission for and notification of changes, approval of the design and construction method, and pre-service inspection are treated as the exception of the act, and the rule for TEPCO’s Fukushima Daiichi NPP is applied. Moreover, other methods can be applied with the approval of the NRA. These systems allow flexible regulation to minimise the total risk. These systems are applied for the demolition, transport, treatment and storage of waste at the site. The act is applied to the decommissioning of the NPP and the final disposal of the waste.

Japanese implementer view

Generally, the interaction between regulator and implementer in Japan is open.

For example, nuclear power plant operators’ communication with regulators is posted on the NRA website because of the meeting rule that all meetings for more than five minutes must be open to the public.

Concerning current Fukushima Daiichi supervision, the Fukushima Daiichi NPP has been designated as a specified nuclear facility, and regulatory investigation and approval of operator’s application concerning the Fukushima Daiichi, such as construction project permits on-site, depends on the time and situation, compared to that of normal nuclear facilities.

On the other hand, regarding future issues, for example, waste disposal rule, given the current situation of Fukushima Daiichi as a specified nuclear facility, it is important to have policymaking and engineering (safety evaluation method, etc.) discussions which lead to disposal rule-making in future, sharing expert views, specific information and data for Fukushima Daiichi waste.

This discussion is likely to be extended in time and be very complex. Therefore, it is better that a discussion framework be established as soon as possible where a neutral body oversees relations between the regulator, research and development (R&D) institutions, the implementer, other experts and stakeholders.

2.3. Recommendations

- A number of overly specific technical requirements established by the regulator will not allow freedom of options for safety, cost benefit and their optimisation for the operator. Furthermore, it is assumed that it will take a long time for such specific requirements to become a national procedure of the regulator. It might lead to unnecessary delay of implementation. Development of an overall

framework for waste management is recommended to ensure the safety of workers and the public, and the protection of the environment after an accident or contamination.

- The final goal of waste management after an accident or contamination is the same for the regulator and the implementer – the safety of the public and workers and protection of the environment. It is imperative to discuss and create a dialogue between the regulator and the implementer to clearly define the roles and the responsibilities of each stakeholder.
- Mandatory interaction between regulators and implementers, e.g. the licensing process, is important at key decision-making point(s) for the waste management activities within an overall strategy. Early interaction, however, is always useful in order to avoid returning hand. If decisions on remediation or decommissioning are made without consideration or a strategy on disposal, the final disposal may be more difficult. Early interaction is recommended and an overall strategy on waste management including the final disposal should be considered at an early stage.

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3. Stakeholder involvement

3.1. General description

Who is the stakeholder?

It is not clearly defined who the stakeholder is in the situation of a post-nuclear accident. In the case of the Chernobyl accident or Fukushima Daiichi accident, large areas and numbers of people have been affected by the accidents. It is not difficult to imagine that there are many stakeholders after such accidents. The dictionary defines a stakeholder as “any actor – institution, group or individual – with an interest or a role to play in the societal decision-making process around radioactive waste management. Different stakeholders may have different interests. Engagement strategies should thus be adjusted to context: differing needs, programme phases, formal requirements, as well as national process and national and local culture” (NEA, 2015). There is no doubt that there are many people who should share information and share responsibilities after a nuclear accident and not only the current generation, but also future generations. Stakeholders may have many standpoints, many thoughts, many reactions on radiological exposure, and different levels of understanding in terms of radiation. However, stakeholders should be engaged in the process of recovery from the situation created by the accident.

Based on accident experiences, stakeholders may include:

- national administration (government);
- regional administration (prefecture, regional government);
- local administration (local government);
- national expert body (national laboratory, technical support organisation for the government);
- experts from universities;
- independent experts;
- local liaison commission around nuclear installations;
- non-governmental organisations (NGOs);
- groups of citizens;
- nuclear operators;
- waste managers;
- international experts.

Stepwise process on stakeholder engagement

Based on the experiences during communication with stakeholders in the post-accident situation in Belarus and Fukushima, there are a great deal of similarities in the consequences of the accident:

- a loss of confidence of the authorities and experts;
- worry on the part of inhabitants about health and especially children;

- a general feeling of discrimination and exclusion in the public;
- feelings of helplessness and abandonment among some people;
- a loss of control of daily life for some people and apprehension about future.

Why should the stakeholders be engaged in the process of the recovery from nuclear accident at all? It goes without saying that people or parties are affected directly and are interested in the recovery process. Further reasons are listed below:

- to take into account more effectively stakeholders' concerns and expectations and the specificity of the context at state;
- to adopt more effective and fairly protective actions;
- to maintain stakeholders' vigilance;
- to empower stakeholders in order to encourage autonomy.

Engaging stakeholders for a "common evaluation" of the situation and for a discussion on ways forward is essential, as well as the organisation of follow-up and monitoring in the perspective of ensuring vigilance and building a common future.

The management of such a situation is different from a "normal situation", although it is difficult to adopt a long-term perspective to establish criteria or protection objectives different (or significantly different) from a "normal situation". There is a need to establish a stepwise process, including flexibility but providing a long-term perspective and addressing concerns for future generations. The process for engaging stakeholders and could be summarised in five steps, as the experiences in Belarus and Fukushima have shown:

- establishing places for dialogue between experts and affected people;
- listening and learning from the inhabitants about their concerns, difficulties and wishes;
- developing a "common evaluation" of the local radiological situation;
- implementing projects to address the problems identified at the individual and the community levels with the support of local professionals, experts and authorities;
- evaluating and disseminating the results.

To implement the necessary measures for such a stepwise process, there are three essential systems for inhabitants engaging after a nuclear accident:

- **Inclusive radiation monitoring system** allowing individuals to regain self-control on their direct environment, i.e. to understand where, when and how they are exposed and what can they do in order to adapt their behaviour and take appropriate actions to protect themselves.
- **Health surveillance system** relying on the participation of the inhabitants.
- **Education system on radiation/radiological exposure** which is based on the practical radiological protection culture and its transmission to future generations. The radiological protection culture must take root in the communities affected by the nuclear accident. This culture may be developed together by the experts and the inhabitants as knowledge and skills on radiological protection during the initial stage, and may be what enables the inhabitants to make decisions or to behave wisely in the situations involving potential or actual exposure to ionising radiation. This culture may allow people to interpret results of measurements, to orient themselves in relation to radioactivity in everyday life, and to provide information to make decisions and take actions.

3.2. Case studies for stakeholder involvement

Three Mile Island 2 (TMI-2) accident

In response to public concern about the clean-up of the Three Mile Island, unit 2 (TMI-2) facility after an accident on 28 March 1979 involving a loss of reactor coolant and subsequent damage to the reactor fuel, 12 citizens were asked to serve on an independent advisory panel to consult with the US Nuclear Regulatory Commission (NRC) on the decontamination and clean-up of the facility. The panel met 78 times over a period of 13 years (12 November 1980 to 23 September 1993), holding public meetings in the vicinity of TMI-2 (Harrisburg, Pennsylvania) and meeting regularly with commissioners from the NRC in Washington, DC.

NUREG/CR-6252 “Lessons Learned From the Three Mile Island Unit 2 Advisory Panel” August 1994 describes the results of a project designed to identify and describe the lessons learnt from the advisory panel and place those lessons in the context of what we generally know about citizen advisory groups. A summary of the empirical literature on citizen advisory panels is followed by a brief history of the TMI-2 Advisory Panel. The body of the report contains the analysis of the lessons learnt, preliminary conclusions about the effectiveness of the panel, and implications for the NRC in the use of advisory panels. Data for the report include meeting transcripts and interviews with past and present panel participants.

The areas of concern identified through the literature review and examination of meeting and interview transcripts were used to organise the information into a lessons-learnt analysis. The lessons learnt include the following:

Panel objectives

- Original objectives were well-known to all panel participants and used effectively to keep panel meetings on track.
- Participants believed that panel objectives were met although there was concern that reduced public participation also reduced the ability of the panel to represent the public.
- Participants perceived that implicit panel objectives included reducing public anxiety about the accident and clean-up of TMI-2 and believed these objectives were met.
- Panel members were able to reduce growing antagonism and conflict between members of the public and other panel participants by expanding the original objectives to include issues of great concern to the public.

Characteristics that support implementation of advisory panels

- Successful advisory group implementation requires a high profile problem with a specific focus.
- Without an appropriate focus, an advisory panel is unlikely to attract quality participants or hold their attention for long.
- Maintaining a successful advisory group requires a continuing high public interest in the event or topic.

Panel composition

- A range of expertise increased the capability of the panel members to participate in technical and political discussions.
- Panel members educated both the public and each other across different areas of expertise and capability.

- Diverse perspectives and capabilities increased conflict among panel participants. This conflict, however, appeared to contribute to the perception of the panel as a credible and legitimate forum for discussion of the clean-up activities.
- The wide range of panel members' perspectives also appeared to increase the credibility of the panel with other participants and observers.
- Although some panel members were unable to contribute directly during certain technical discussions, they did participate by providing additional perspectives to the issues under consideration.

Meeting structure

- Consistently applied speaking rules created a perception of fairness among panel participants.
- An informal atmosphere provided the appropriate flexibility for wide participation.
- Impersonal methods for meeting control maintained respect for individual perspectives.
- A mid-meeting public comment period increased the range of public response and reduced increasing tensions between citizens and panel members.
- Frequent, but controlled, periods for public participation increased the quality and quantity of input and reduced ongoing conflict over meeting procedures.
- Recommendations and reports to the NRC commissioners were most often developed through informal consensus building among panellists.
- Respondents believed that improvements could be made to the advisory panel by increasing resources for the panel, increasing the technical aspects of the NRC Designated Official role, and reassessing how panel members are selected.
- Term limits for panel members did not appear feasible to most participants due to the complexity of clean-up issues.

Panel influence on the clean-up

- The most crucial panel influence on clean-up activities was the increased public scrutiny of both NRC and licensee decisions and activities.
- The panel facilitated communication with the public for both the NRC and the licensee. This communication helped sensitise the agency and the licensee to public concerns.
- The level of technical influence on clean-up activities was modest and, in any case, difficult to untangle from other pressures put on the licensee. Most respondents agree, however, that panel and public questions expanded the range of alternatives considered by the NRC and the licensee.

Role of the media

- Local media covered the advisory panel meetings throughout the years.
- In the early years, front page coverage of meetings was common. During later years, stories about the meetings moved to back pages with other, less controversial, news.
- Media coverage disseminated clean-up information to a wider audience than was reached through the panel meetings.
- Media coverage encouraged high-quality presentations about the clean-up.

- Some participants believe that media coverage provided opportunities for grandstanding and irresponsible claim-making to wide audiences.
- Media coverage may have reinforced the significance of panel activities to panel members and encouraged their continued participation.

Panel longevity

- Many participants continued with the panel in spite of initial concerns about its efficacy because it was the only forum available for participating in discussions about the clean-up.
- The longevity of the advisory panel served to smooth over divergent views of panel participants, allowed enough time for individuals to learn about the complicated technical issues involved in the clean-up, and created an almost universal perception that the panel was an effective communication forum.
- Although interpersonal trust between panel participants was generally quite high, this trust has not typically been translated into increased trust of the institutions or organisations that other participants represented.
- All past and present panel members expressed surprise that the panel survived for 13 years. Even those panel members who believed the panel should continue thought the panel had only a few issues left to address.

Chernobyl nuclear power plant accident – Ukraine

Involvement of “government”, “regulatory, supervisory and managing authorities” as stakeholders

To speak about stakeholders involved in the planning and implementation of activities at the destroyed Chernobyl nuclear power plant (ChNPP) unit 4, one should start from the government and state regulatory and supervisory authorities, as well as the managing authorities. The extent of their involvement in the processes at ChNPP commonly depends on allocated functions and duties.

However, the issue of an effective mechanism for involving these categories of stakeholders in the planning process (first of all, *strategic planning*) appeared together with an understanding, at the governmental level, of the need to develop a national long-term strategy for shelter object transformation. For this purpose, the “Interdepartmental Commission for Comprehensive Solution of the Chernobyl Nuclear Power Plant Problems” was established in 2000. The commission is headed by the Vice Prime Minister and includes top managers or deputy top managers of various regulatory and supervisory bodies, including the ChNPP Director, Mayor of Slavutych and representatives of the ChNPP trade union organisation. The expert working group including representatives of various organisations and scientific institutions was established for detailed consideration of issues and preparation of draft decisions of the Interdepartmental Commission. It should be mentioned that the primary task of the Interdepartmental Commission was to develop and approve the National Strategy for Shelter Transformation.

“The Interdepartmental Task Force of Regulatory Authorities” (ITFRA) was established by the regulatory authority (State Nuclear Regulatory Committee of Ukraine) to co-ordinate activities of other regulatory authorities (RAs) involved in the shelter transformation process.

ITFRA is an advisory working structure for the online interaction of the above-mentioned state regulatory authorities in the regulatory process. The mission of ITFRA is to: first, organise mutually agreed actions of regulatory authorities to avoid overlapping activities, reduce the time spent on reviews and agree on designs to be created on the

ChNPP site; second, to promote the solution of issues that are not regulated by standards and rules in force and require effective resolution during development and implementation of these designs. In doing so, the following main tasks are performed:

- analyse and assess the progress of development and implementation of designs and RA review of appropriate design documents;
- co-ordinate RA actions in reviewing design documents;
- analyse procedural issues in reviewing design documents and support their solution;
- agree RA proposals on the procedure for expert reviews of design documents;
- reveal potential problems in reviewing design documents and make proposals on avoiding these problems (proposals should be agreed in future with RA management).

The first favourable experience took place in the co-ordinated work of regulatory authorities and their technical support organisations (TSO) in reviewing the terms of reference (TOR) for the development of the safe confinement conceptual design. Under the co-ordinating role of the State Scientific and Technical Center for Nuclear and Radiation Safety (TSO of the regulatory body in the nuclear field), five expert organisations of different RA carried out an expert review (technical evaluations) of this document during a short period. The ChNPP was provided with a summarised document with comments and recommendations of all expert reviews. Conclusions of state expert reviews (technical evaluations) allowed the regulatory authority to make well-grounded regulatory decisions concerning the above-mentioned TOR.

It should be noted that implementation of projects and different activities within the shelter transformation caused a number of issues related to regulation, control, licensing and oversight, which have to be solved by various regulatory authorities within their competence and powers. The experience shows that it is necessary to ensure clear working procedures and constant interaction of regulatory authorities involved into the licensing process to ensure effective implementation of projects. This is needed to reduce/eliminate the so-called regulatory risks arising during implementation of large projects at the destroyed facility. For example, the regulatory authority develops and approves a new regulatory document. Accordingly, the operator of that facility analyses the feasibility and cost of implementing changes related to introduction of the new/revised regulation. The problem arises when the implementation of measures requires additional money and time, but the operator has to keep to the schedule and remain within envisaged funding. In this case, the operator has the right to apply to the regulators group, which in turn has to address the problem and find an acceptable solution or recommend specific actions and possible ways to achieve the objectives and results envisaged through implementation of the new regulation. Another example might be a problem when the operator recognises that implementation of the regulatory requirement would cause significant doses on personnel associated with difficulty of access and high radiation fields in places of operations. Such cases are also brought to the regulators group for discussion, and recommendations are prepared for each case taking into account the specific issues.

The regulator's group still operates in Ukraine, though it gathers much less frequently than it did earlier. This is because the main issues have been resolved at early stages of project implementation. The main issues that were discussed during the last meetings were connected with the need to comply with regulations and rules on fire safety during implementation of the New Safe Confinement (NSC) project and high doses that would be received by personnel involved in actual activities required to take fire safety measures.

Involvement of the operator – “Chernobyl NPP” as stakeholder

The Joint Coordination Group for SIP Licensing (JCG) between the RA and ChNPP was established to co-ordinate the licensing process related to shelter transformation. The JCG consists of representatives of the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) and its technical support organisation and representatives of the ChNPP, including Project Management Unit for Shelter Implementation Plan (SIP). The JCG is an organisation for online interaction between the ChNPP and SNRIU in the SIP licensing, and should promote the efficiency and quality of this process (see Chapter 2). At present, the above-mentioned commission and groups continue to operate.

Involvement of national and international experts/scientific, research and technical organisations

The State Scientific-Technical Centre for Nuclear and Radiation Safety (SSTC NRS) is an official technical support organisation for Ukrainian regulatory authority in the area of nuclear energy use. The SSTC NRS supports the regulator in the licensing activity related to the transformation of shelter object into an ecologically safe system.

Different scientific, research, technical organisations are involved with the operator (ChNPP) at different stages of projects related to the shelter implementation plan (SIP).

Involvement of foreign experts and active international co-operation were always included in the planning and implementation of projects at the shelter. Indeed, the International Consortium, which included Ukrainian scientific institutes also, developed the “shelter implementation plan” (SIP). SIP is a large international project that includes a number of tasks related to the updating of shelter object safety, stabilisation of structures and components of the shelter object, creation of the new safe confinement and management of radioactive waste and fuel-containing materials. The financial support of SIP is provided from the “Shelter Implementation Fund”. This fund accumulates the money of donor countries. The European Bank for Reconstruction and Development is administrator of this fund. Ukraine is also obliged to donate money to the “Shelter Implementation Fund”.

It is important to say that during the period of implementation of SIP, both the “operator” and “regulator” have been supported by the international and national experts/specialists in different fields of knowledge from different organisations, institutes and companies.

During the licensing activity related shelter transformation, the Ukrainian regulatory authority was supported by international experts from Germany and France (Riskaudit Company) and from the United States (Sciencetech Company) and national experts from the technical support organisation of the regulator. Involvement of international and national knowledge and experience was practically useful and important to support the decision-making process provided by the regulatory authority.

The “co-ordinating” of activity organisations/experts involved in the process of project implementation is very important because of the many organisational procedures established in contracts, and agreements of involved parties. Another important aspect was the clear charring of responsibility between involved parties, which meant that the national and international experts/organisations played an important role in supporting the operator or regulator. At the same time, the expert opinion could not replace the decision of operator or regulator.

Involvement of the “public” as stakeholder (based on the materials from the National Report of Ukraine “25 Years after Chernobyl Accident. Safety for the Future”)

A national policy related to people affected/suffering from Chernobyl accident was established in the Law of Ukraine “About Status and Social Protection of Citizens Suffered from Chernobyl Accident”. This law determines the main provisions regarding implementation

of the constitutional right of citizens who suffered from the Chernobyl accident for protecting their lives and health, and it demonstrates a single procedure for determining the status of people who suffered. When this law was accepted, the government started to work on preparing and putting into force subordinate legislation to implement provisions determined by the legislation, first of all, to determine the status of people who suffered and organise their social protection.

Categories of affected people were defined depending on the status of contaminated territories (so-called “zones”) established in another law of Ukraine – “About Legal Status of the Territories being Radioactively Contaminated due to Chernobyl Accident”. The main purpose of this law is to implement the legislative determination of legal status for territories having different radioactive contaminations and measures on its provision.

The compensation policy relating to all categories of people who suffered from Chernobyl accident was introduced in Ukraine. Compensation was implemented as payments, free and extraordinary access to different services causing a significant increase of costs in the state budget. When the country gained independence, the political institutes being at the initial stages, on behalf of their electorate, they actively addressed the problems caused by the Chernobyl accident. As a result, the parliament repeatedly agreed to the recovery of damages without an appropriate assessment of resource abilities. Many liabilities were not fulfilled, and “Chernobyl payments” fell onto the state budget.

Because of these facts, a significant part of measures prescribed by the law “About Status and Social Protection of Citizens Suffered from Chernobyl Accident” were never performed and did not meet the expected outcome.

Moreover, after approximately 30 years since the Chernobyl accident, there is still a category of “affected/suffering people” waiting for “Chernobyl payments” and not interested in losing these payments and other social benefits established by the state for those living in contaminated areas. It is important to add that this category of people believes that the state is obliged to provide these compensations forever because people do not believe that the territory where they live can be safe, taking into account radiation risks. Such an attitude exists because of many years of implementing “policy of compensation”, not effective activity related recovery of contaminated territories including not sufficient policy related information for the population about radioactive contamination of lands, water, air, products, food and real exposure doses of members of the public.

“Chernobyl syndromes” of people affected/suffering from Chernobyl accident and needs for future

Thus far, treatment of the following variety of social syndromes that have been imposed on the affected community by the disaster and its aftermath have been a failure, whether by medical means the expenses of material compensations or environmental rehabilitation:

- “syndrome of a victim” – a large part of the affected individuals refer to themselves as a community of victims during their entire lifetimes;
- “syndrome of social exclusion” – absence of initiative, paternalism, demands for “eternal rent from the government” dominate in the collective consciousness of affected individuals;
- “syndrome of evacuation and resettlement” is driven by a disturbed picture of the world and weak adaptation to new conditions that are typical for the affected people;
- “syndrome of lost health” is a combination of adults’ and children’s’ health deterioration and a fact of the disaster and its overpowering consequences;

- “syndrome of uncertainty and confusion” is a paradox reliance of the affected individuals upon the government in terms of solving their problems combined with simultaneous almost complete distrust of authorities and recognition of a real support from a family;
- “syndrome of ignorance” is the affected individuals’ unfamiliarity with laws and rules of activities for daily living in the post-accident environment, thus guidance by subjective risks and not by actual situation in a daily life.

In summarising the practical experience related to the elimination of the socio-psychological consequences of the Chernobyl accident during an approximately 30-year period, some important conclusions can be made:

- need to revise the social policy with the objective to revive life within radioactively contaminated areas and involve national, non-governmental, business, and public effort into the process of recovery, i.e. it is necessary to join administrative and self-government controls in an integral system;
- need to reorient the programme of recovery within radioactively contaminated areas from contamination hazards, i.e. radiation risks, towards activation of people and communities, search for innovative chances of life-sustaining activity and behaviour within radioactively contaminated areas;
- need to develop an evolutionary way for the transformation of people from “affected” status to a status of a full-fledged citizen of Ukraine.

Activities of the socio-psychological rehabilitation centres and distribution of information to the affected individuals

The centres for socio-psychological rehabilitation and distribution of information among the affected people were established in 1994-2000 under the support of the United Nations Development Programme (in cities such as Borodianka, Boyarka, Ivankiv, Korosten and Slavutych) to be involved in tasks oriented towards the elimination of the social and psychological consequences of the Chernobyl accident to the population.

The following were the field of concerns covered by the centres for socio-psychological rehabilitation and distribution of information among people: social and psychological support to people; development of personal responsibility for one’s own life; orientation towards affirmative addressing of existing problems; development of communities and interpersonal relations; formation of efficient behavioural models in line with new living conditions. While ensuring continuous interface with the communities in terms of developing their self-government and local upgrowth potential, the centres warrant the stability of the projects’ outcomes and ensure the acquisition and dissemination of their positive experience.

Social and psychological problems of affected people to date are still urgent, as is overcoming the “syndrome of a victim” and the negative perception of radioactively contaminated areas as a potential place to live. Hence, experts of the Centres for people’s socio-psychological rehabilitation endeavour to find new approaches to address the above problems. Also, the centres’ important work area is developing civic engagement among youth. The objective is to involve new generations in the social and political life of the native habitat, foster leadership skills in youth as well as interest in decisions taken to have an effect on the life of the communities, environmental outlook and healthy lifestyle.

Dissemination of information on the Chernobyl disaster consequences to the public is still the most efficient method for overcoming social and psychological problems. Informational, analytical, and educational activities of the centres are aimed at identifying the key issues regarding general environmental situation in the region, socio-economic processes, as well as public needs for information about safe living within the

radioactively contaminated areas. A top priority of the centres' research activities are the following issues of "Chernobyl": public attitudes towards various problems; level of awareness; psychological, social, and ecological aspects of life in the regions.

In order for the centres to efficiently implement initiatives and programmes, introduce advanced international techniques into the recovery processes and sustainable development of a strong psychological and social immunity among the affected population, the institutions collaborate with international organisations and programmes, promote attraction of charitable funds for the implementation of projects targeted at socio-economic recovery of the affected regions and improvement of the local life standard.

The centres for rehabilitation demonstrated their highly efficient activity while helping all age brackets of people; disseminating information about opportunities for social risk mitigation among all interested groups; extending their activity to entire districts (rayons); facilitating formation of active communities in population centres that are targeted at overcoming their most pressing problems.

The regulatory authority and involvement of the public

The regulatory authority established the Public Council. It includes representatives of public organisations (including green) and independent experts in nuclear energy. The Public Council has the right to discuss different issues (long-term operation of power units, diversification of nuclear energy use, Chernobyl aspects). Issues on transformation of the destroyed ChNPP unit 4, both current and future, are regularly initiated and considered by the Public Council.

Since recently, legislative documents have been approved in Ukraine in order to extend the opportunity for the public to receive information from executive authorities and the government. Therefore, in addition to information received via set mechanisms (from printed media, official Internet pages of agencies and organisations, official correspondence, participation in workshops, public hearings, etc.), members of the public can apply to a specific organisation with a certain issue via e-mail and get the answer in a short time (7-14 days). It is important to note that the authority is responsible for the failure to reply within the established time frame.

Lessons learnt

- Interdepartmental commissions and working groups are most important and productive in the period when strategic documents need to be developed and approved by the government. After the strategic documents are developed and approved, experience shows that the role and activity of these groups gradually decrease.
- The Co-ordination Group between the regulatory authority and the operator proved to be equally important and efficient because it allowed immediate regulator/operator interaction during licensing of different projects approved in the licensing plans.
- Establishment of the Public Council is an effective mechanism for the involvement of the public in consideration of issues related to shelter object activity and building a constructive dialogue with members of green organisations and journalists.
- The national policy should be aimed at reducing the "victim" feeling. One should diverge from the stereotypes that only the state could solve all issues associated with mitigation of the accident consequences. People suffering should also try to overcome obstructions on their own.
- It is necessary to transit from risk compensation policy to compensations for damage done in fact.

- The society of “Chernobyl” victims (2.6 million people) is in a state of social depression and social exclusion. Paternalistic orientations towards governmental rent for the lost health and broken lives do take place. However, it is a mistake to limit the social policy for eliminating the consequences exclusively to social assistance. Large-scale recovery activities are needed in order to return the affected individuals to active life.
- Long-term keeping of the affected communities in a state of an information vacuum is unacceptable. It is necessary to continuously disseminate information about environmental conditions and ways of adequate behaviour and living; the information is to be recipient-oriented and specific.
- In contrast to the risk concept that causes fear, stress, and various “social syndromes” in the affected people, it is necessary to increase productivity of a chance concept, i.e. focus on search for and implementation of efficient behavioural models and life activities in post-accidental situations. A leading role in the chance concept is to be given to the idea of social health and returning consciousness towards the future.
- Distribution of complete, timely, and targeted information about the risks and chances usually helps affected people to return to an actual situation spacing and real behaviour.
- Keeping the Chernobyl-affected people in a “stopped life” atmosphere for such a long time is unacceptable. It is necessary for a large-scale recovery and development programme for the affected individuals and communities to initiate (although with a great delay) a “roadmap” in overcoming the social, sociocultural, and socio-economic impact of the disaster.
- For all the activities aimed at recovery and development of the affected communities and areas, an allowance is to be made for the nationwide trends of retargeting active models of behaviour and living activity.
- Organisation of “socio-psychological rehabilitation centres” is a good practice to deal with social and psychological problems of the public associated with consequences of the Chernobyl accident.

Chernobyl NPP accident – Belarus

The Nuclear Evaluation Protection Centre (Centre d'étude sur l'Evaluation de la Protection dans le domaine Nucléaire, CEPN), France, had developed, in co-operation with other French research teams, the stakeholder engagement approach in the field of radiological protection after the Chernobyl nuclear accident in Belarus. In this approach, public meetings were organised to listen to the concerns of inhabitants about radiation and its effect for the health. Major concerns of the inhabitants were the effect of the radiation on their health and on the agricultural products in the area affected by the accident. These meetings with inhabitants and other stakeholders pointed out the importance of developing a “common evaluation” of the radiological situation and its impacts on exposure of the public. To develop a “common evaluation” between the public who lived in the affected area and the radiological protection experts, inhabitants had measured their own radiation level in their everyday life environment. It is a most important point that the radiation (dose rate) should be measured by the inhabitants themselves. In this way, they can feel and understand their exposure situation.

In addition, the information on the radiological situation of inhabitants had been collected. Through communicating with the inhabitants and collecting the information of the exposure situation (in other words “life style”), the issues and problems for developing the “common evaluation” were identified. Information is based on the local traditions, habits and diet, and distribution of local productions. Through these activities,

individual exposure estimations have been collected and individual ways to grip and improve the environment of daily life have been promoted. Furthermore, the heterogeneity of the contamination in each area and the distribution of exposure dose with relation to the individual behaviours was also revealed in this way.

Figure 3.1. Communication with stakeholders and measurement of radiation in Belarus



Source: Jacques Lochard.

The interpretive activities on radiological exposure and radiation for the inhabitants, and the actual actions for avoiding unnecessary exposure, had been implemented at the same time in Belarus. There were some activities for better understanding of the contamination of local production, especially milk production, and identifying together the way forward to improve the situation.

Figure 3.2. Communication with stakeholders and measurement of radiation in Suetsugi, Fukushima



Source: Jacques Lochard.

Fukushima accident

After the Fukushima accident, the CEPN has been involved in the practice on stakeholder engagement in Suetsugi area, Iwaki City, Japan. Suetsugi is located in south of Fukushima Prefecture and 30 km from the Fukushima Daiichi NPP. In Suetsugi, the decontamination activities have been performed and the soil has been stored temporarily in an area located in the village.

Dialogue between the habitants and international/national experts has continued in Suetsugi, and many kinds of radiological measurements also have been done by the inhabitants themselves with the support of the experts as in Belarus. It is most important point that the radiation (dose rate) should be measured by the inhabitants themselves for feeling and understanding their own exposure level and situation. This is the “common evaluation” that a concept developed by the experiences in Belarus.

Lessons learnt

The Chernobyl and Fukushima experiences demonstrate that the contribution of local actors through self-help protection is the “engine” of long-term recovery from nuclear accidents.

The role of experts is to serve local actors and to facilitate the development of their ability to assess and manage their own situation. Experts must evolve from the explanation of phenomena to the resolution of problems together with the affected people (co-expertise).

The pluralism of sources of measurement (public and private; local and national) is important for ensuring confidence of the population in the results. National resources must be mobilised to support community projects and local producers to improve living conditions in the areas affected. Places of dialogue to exchange experiences are essential to engage stakeholders and diffuse the practical radiological protection culture.

3.3. Recommendations

- Practical experience shows that the level of trust and confidence of the population (main stakeholder) in governmental authorities and operational organisations can be lost when taking into account the significant influence of accident consequences (radiological risks and health, social and psychological effects). It is very important to define, create and support an organisation/commission or special group of people (experts, consultants, scientists) to provide all information (including reporting) needed by the population. Such information should be true, timely and easily understood. At the same time, the information to be provided to the population should be adequate and based on the level of risks to avoid panic.
- Creation of a public organisation to help speak with people from contaminated territories is useful practically. The activity of such organisations should be supported by the government. Moreover, international co-operation and support should be encouraged.
- Identification of the main groups of stakeholders is needed especially for non-specific situations (e.g. during the post-accidental period when long-term strategy related contaminated territories is under development). A number of different governmental and non-governmental organisations (technical and scientific) should be involved, such as local authorities from territories affected by the accident.
- Creation of new mechanisms (procedures) for stakeholder involvement is needed. The regulatory body and operator should review the current procedure and develop new mechanisms for stakeholder involvement taking into account the specific situation and tasks to be performed (e.g. licensing of damaged nuclear facility, solving a problem related to the management of waste accidental origin).
- Absence of full, adequate and true information for a long period creates additional social and psychological problems in the population affected by the accident (e.g. creation of psychological syndromes of victim or social exclusion).

3.4. Reference

NEA (2015), *Stakeholder Involvement in Decision Making: A Short Guide to Issues, Approaches and Resources*, OECD, Paris.

4. Physical and chemical nature of the waste

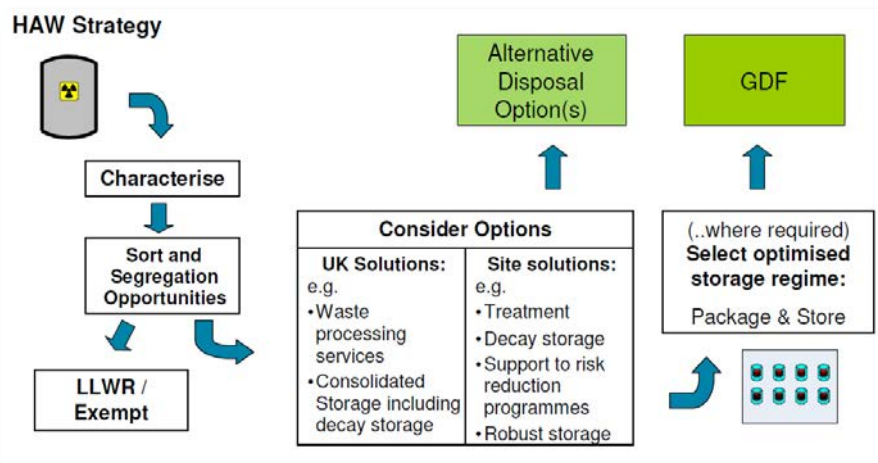
4.1. Introduction

This chapter first summarises some aspects of state-of-the-art physical and chemical characterisation approaches and techniques that could be relevant to the characterisation of accident-related waste. It then presents the experience of the Three Mile Island 2 (TMI-2), the Chernobyl nuclear power plant (ChNPP) and the Fukushima Daiichi programmes in the physical and chemical characterisation of waste produced by the accidents. These case studies also include descriptions of the volumes and types of solid and liquid waste produced, as well as presenting some information on the temporary storage of such waste and its radiological characteristics. There are therefore several common themes between this chapter and Chapters 5 and 8, which consider radiological characterisation and destination (storage/disposal) of accident waste. The chapter concludes by making recommendations for physical and chemical characterisation of materials and waste in future accident situations.

4.2. State of the art

Introduction

Research and development (R&D) programmes to support reactor decommissioning and waste management are well-developed worldwide and key themes have been identified: improved decommissioning technologies; waste characterisation; waste minimisation, treatment and conditioning; interim storage, and; long-term waste management, including transport and ultimate disposal in suitable repositories. These themes enable the development of an integrated waste management strategy. The figure below is taken from the UK Nuclear Decommissioning Authority’s report “Integrated Waste Management Strategy Development Programme” (NDA, 2012).



Source: NDA, 2012.

Physical and chemical characterisation of materials and waste is important for several purposes. For example:

- to better apply the waste hierarchy by minimising waste volumes through sorting, segregating, reusing and recycling materials where possible;
- to enable appropriate waste conditioning, volume reduction and packaging solutions to be developed;
- to enable the radionuclide inventory in reactor components to be estimated, based on an understanding of their irradiation history;
- to enable the performance of the waste during storage and after disposal to be assessed;
- to enable appropriate interim storage facilities to be developed;
- to ensure that the waste is suitable for disposal, when such disposal routes become available.

In the following sections, we summarise some aspects of “state-of-the-art” physical and chemical characterisation that are relevant to the characterisation of accident-related waste. We consider the following areas:

- development of the materials inventory;
- hazardous non-radioactive substances in waste;
- chemical complexing or chelating agents;
- calculation of inventories of radionuclides produced by neutron activation reactions.

More detail on some of the topics presented below is given in the NEA report *R&D and Innovation Needs for Decommissioning Nuclear Facilities* (NEA, 2014).

Development of the materials inventory

The materials within the waste will strongly influence waste behaviour in the short term, medium term and long term. Therefore, a materials inventory should be developed to enable future waste behaviour to be assessed. An example of the type of information that could be collected is shown in Table 4.1, based on the 2013 UK Radioactive Waste and Materials Inventory (NDA, 2014). The list should be tailored to meet the requirements of the relevant decommissioning and waste management programme. For example, information should be collected to enable implementation of waste minimisation and conditioning approaches, in addition to ensuring the “disposability” of the waste in the long term.

Table 4.1. Material components of waste

Metals: stainless steel; other steel; aluminium; other
Organics: cellulose; plastics; rubbers; other
Inorganics: asbestos; concrete, cement and sand; graphite; glass and ceramics; sludges, flocs and liquids; other
Soil and rubble

Information presented in the 2013 UK Radioactive Waste and Materials Inventory.

Hazardous non-radioactive substances in waste

It is likely that the operator of any facility for the storage or disposal of radioactive waste produced from an accident will need to demonstrate the safety of the facility against non-radiological hazards. The first stage will be to identify those non-radiological substances deemed to be hazardous; this may be done by reference to national regulations or may require review of the approaches taken in other countries. For example, the EU Groundwater Directive (EU, 2006) sets out objectives for groundwater quality and specific measures to prevent and control groundwater pollution. It distinguishes between “hazardous substances” and “non-hazardous pollutants”, and requires member states to develop a list of substances that should be determined as hazardous. Member states are required to “prevent” inputs of hazardous substances into groundwater and to “limit” the input of non-hazardous pollutants into groundwater.

The inventory of toxic metals in alloys within the waste is relatively straightforward and is to be determined from knowledge gathered from the materials inventory. Information on other non-radioactive hazardous substances will need to be identified for accident waste.

Chemical complexing or chelating agents

The presence of any chemical complexing or chelating agents in waste streams should be established and compared with the likely limitations on such materials in a future disposal site. Depending on the outcome of this comparison, it may be appropriate to assess the impact of chemical complexing or chelating agents on the safety performance of the disposal facility and, if necessary, restrict the presence of such materials in the waste.

Calculation of inventories of radionuclides produced by neutron activation reactions

Two different characterisation approaches are used to estimate compositions and concentrations of activation products in irradiated reactor components. The first involves direct measurement of all relevant radionuclides, including hard-to-measure radionuclides, as discussed in Chapter 5. The second approach involves calculation or modelling based on knowledge of the neutron fluence and the composition of the irradiated material; this is the approach generally used for reactor internals, which cannot be easily accessed ahead of dismantling. The most highly activated components are the stainless steel reactor internals, and knowledge of trace element compositions in these steels is required to enable calculation of activation products. The NEA report on *R&D and Innovation Needs for Decommissioning Nuclear Facilities* (NEA, 2014) identifies the need for a better understanding of the range and statistical distributions of cobalt and trace contaminants in irradiated reactor components to improve estimation of radionuclide inventory through calculations.

4.3. Case studies

TMI-2

Accident-generated water

One of the most significant issues was public intervention in the form of a suit filed by the City of Lancaster to block the release of any TMI-2 water to the Susquehanna River, even if the water met all regulatory criteria. An out-of-court settlement was reached early in 1980, known as the City of Lancaster Agreement. The agreement, signed by the City of Lancaster, the US Nuclear Regulatory Commission (NRC), and the Licensees of TMI-2, placed significant restrictions on the discharge of accident-generated water.

“Accident-generated water” was defined as:

- water that existed in the TMI-2 auxiliary, fuel handling and containment buildings including the primary system as of 16 October 1979 with the exception of water which as a result of decontamination operations became commingled with non-accident-generated water such that the commingled water had a tritium content of 0.025 micro Ci/ml or less before processing;
- water that had a total activity of greater than 1 micro Ci/ml prior to processing, except where such water was originally non-accident water and became contaminated by use in clean-up;
- water that contained greater than 0.025 micro Ci/ml of tritium before processing.

Approximately 8 700 m³ of processed water eventually was defined as accident-generated water. None of the ion exchangers employed at TMI-2 could remove tritium from water, and thus almost all water used at TMI-2 after the accident had to be stored.

Additionally, boric acid and NaOH were added to the reactor coolant system to maintain a sufficient boron poison concentration to ensure the damaged fuel would remain subcritical under all conditions. At the end of the clean-up process, these chemicals were removed from the processed water by use of an evaporator used to discharge the tritium-contaminated water from TMI-2. The solid material produced by the evaporation process was sodium tetraborate which was dried, packaged in 55 gallon drums and shipped for disposal.

Abnormal waste

▪ Epicor-II

The Epicor-II system was designed to handle intermediate-level liquid waste (from 1 to 100 microcuries/ml radioactivity) generated by the accident, such as the water which covered the unit 2 Auxiliary and fuel handling building floors and filled the reactor coolant bleed tanks. Beginning in October 1979, the EPICOR-II demineraliser system was used to filter and remove radionuclides from approximately 2 100 m³ of accident-generated water in the basement of the auxiliary and fuel handling buildings. The EPICOR II system in the configuration used to process this water consisted of three carbon steel liners in series (two 4 ft (1 m) x 4 ft (1 m) liners followed by one 6 ft (2 m) x 6 ft (2 m) liner) followed by cartridge filters to trap any released resin fines and particulates. The process was completed in August 1980; it resulted in 72 contaminated filters (65 4x4 liners [50 prefilters and 15 demineralisers] and 7 6x6 liners). The 50 prefilters contained 1 430 ft³(40 m³) of organic resin, 273 ft³ (8 m³)of inorganic resins (zeolites) and 12.5 ft³ (0.5 m³) of charcoal. The prefilters contained an average of approximately 1600 curies of activity; these radionuclide concentrations precluded commercial disposal. Instead, these materials became part of the abnormal waste inventory. The 22 second-stage filters were disposed of commercially as low-level waste.

▪ Submerged demineraliser system

The submerged demineraliser system (SDS) was designed to process the high-level liquid waste (greater than 100 microcuries/ml) in the reactor building basement and the reactor coolant and makeup and purification systems. The original design of the system called for two filter vessels in series (which were replaced by sand filters shortly after system start-up), feeding two parallel trains of zeolite ion exchanger vessels (three vessels per train for a total of six), and in turn feeding two parallel cation exchanger vessels. Each stainless steel ASME code pressure vessel held approximately 0.3 m³ of media. SDS operated in various modes from 10 July 1981 to 21 July 1988, processing 159 batches of water totalling 17 300 m³ through the ion exchangers. During its lifetime the operation of SDS generated 37 zeolite vessels with a total of 8.4 m³ of zeolite used. The system was first used to remove caesium and strontium from accident-generated water in the

basement of the reactor building, primary reactor coolant system and several miscellaneous tanks. Three years later the system was used to remove caesium eluted from the makeup and purification system demineralisers. A total of 19 stainless steel vessels resulted, each containing inorganic zeolites loaded with as much as 112 600 Ci of radioactivity. During this processing effort, there was no attempt to control the activity within these vessels to comply with commercial burial standards. These vessels became a part of the abnormal waste programme.

▪ Damaged fuel

The original TMI-2 core inventory included approximately 94 000 kg of UO₂ and 35 000 kg of cladding, structural and control materials. Accounting for the oxidation of core materials during the accident and for portions of the upper plenum structure that melted, the total amount of post-accident core debris was estimated to be approximately 133 000 kg.

During the accident, peak temperatures ranged from approximately 3 100 K at the centre of the core (molten UO₂), to 1 244 K immediately above the core, and 723 K at hot leg nozzle elevations. Approximately 50% of the original core became molten. Following the accident a cavity existed at the top of the original core region. Below that, a bed of loose debris rested on a resolidified mass of material that was supported by standing fuel rod stubs.

The stubs were surrounded by intact portions of fuel assemblies. Of the original 177 fuel assemblies, 42 partially intact assemblies were standing at the periphery. Only two of these fuel assemblies contained more than 90% of their full-length cross-sections with the majority of fuel rods intact. The other assemblies suffered varying degrees of damage ranging from ruptured fuel rods to partially dissolved fuel pellets surrounded by once-molten material. A previously molten, resolidified mass was encapsulated by the distinct crust of material in which other fragments and shards of cladding could be identified.

Approximately 30 000 kg of molten materials flowed from the core to the core bypass region and through the lower internals. Approximately 19 000 kg came to rest on the reactor vessel lower head.

Lessons learnt from the TMI-2 accident

The TMI-2 case study highlighted the following:

- Accident-generated water was cleaned to radionuclide concentration levels below the site discharge permit, but because of its association with the accident, disposal of this water was opposed by local stakeholders and alternate disposal methods were needed.
- Accident-generated water was cleaned by resin-containing systems. Because of the urgent need to process this water early, resin radionuclide loading levels exceeded burial ground acceptance limits and alternate disposal options were needed.
- Damaged fuel existed in several forms (fine particles, rubble, solidified mass, partial fuel assemblies) such that different types of damaged fuel containers were needed to contain the fuel eventually removed from TMI-2.
- A significant quantity of fuel was displaced from the original core region. Therefore, methods to locate and quantify this fuel were needed.

Chernobyl radioactive waste arising from the accident

As a result of the Chernobyl accident, a considerable amount of radioactive materials including radioactive waste (RW) is concentrated in the exclusion zone and zone of

absolute resettlement. The main places of RW location in the exclusion zone are the following:

- the shelter object (SO), which will provide temporary storage for unorganised RW on the ChNPP site;
- sites for disposal of radioactive waste (SDRW). “Buryakivka, Pidlisny, the 3rd stage of the ChNPP” (the name of the site for planned reactor units 5 and 6);
- sites for temporary localisation of radioactive waste (STLRW);
- waste located on both SO and ChNPP industrial sites and adjacent territory.

The total amount of RW in the exclusion zone (excluding the SO) is about 2.8 million m³. Of this, over 2 million m³ RW with total activity of about 7.4E+15 Bq are located in the SDRW and STLRW. The RW consists mainly of short-lived low-level waste (LLW) and intermediate-level waste (ILW). The total activity in the natural environment of the exclusion zone (in the surface layer of soil, bottom precipitates of water reservoirs, vegetation, etc.) is over 8.5E+15 Bq. The total amount of radioactively contaminated materials in the exclusion zone is equal to 11 million m³.

RW of Chernobyl origin varies greatly in radionuclide composition, specific activity and physical/chemical composition. In contrast to other technological types of RW, Chernobyl accident waste is characterised by the presence of a wide spectrum of radionuclides, including those having considerable half-lives. Most Chernobyl RW is kept under conditions that do not meet the requirements of modern radiation safety norms. At the majority of RW repositories in the exclusion zone (except for SDRW Buryakivka and Pidlisny) radionuclide release from the facilities (for example, contamination of groundwater with radionuclides) is observed. This is a result of the absence of a proper system of engineering barriers and periodical flooding of STLRW.

The areas of RW disposal

SDRW Pidlisny was built for RW with an exposure dose rate (EDR) of up to 50 R/h, but, according to the decision of the government commission, RW with an EDR of up to 250 R/h were located there. The total amount of RW is 1.1E+04 m³, according to the data of 1990; the accepted estimation of total activity is 2.6E+15 Bq. The results of SDRW external investigations, which take account of more recent information, estimate the inventory to be 2.6E+18 Bq, suggesting that RW activity is considerably understated. All the RW in SDRW Pidlisny (see Figure 4.1) contain long-lived radionuclides and are liable to require geological disposal (see Chapter 8).

The presence of many cracks in the concrete foundation and walls of the structure calls for investigation of its condition. The main goal of the investigation should be an assessment of SDRW safety and development of a design for its stabilisation for the whole period up to the construction of a deep geological repository.

Figure 4.1. SDRW “Pidlisny”: Before and after reparation works



Source: Kilochytska, 2015.

SDRW “The 3rd stage of the ChNPP” was built for RW with an EDR of up to 1 R/h, but waste with much higher EDR was located in it. According to the data from the 1995 investigation, SDRW contains $2.6E+03 \text{ m}^3$ of low- and intermediate-level RW including long-living radionuclides, with total activity of $4.7E+14 \text{ Bq}$. Atmosphere and groundwater have free access to the depository owing to the absence of engineering isolation. SDRW requires investigation aimed at development of a project for its stabilisation and prospective liquidation.

SDRW “Buryakivka” (Figure 4.2) was created in 1987 for disposal of RW with EDR of up to 1 R/h. The decision of the government commission allowed placement of waste with EDR of up to 5 R/h. This trench-type repository for disposal is practically full. Options related to reconstruction are under review – a decision about additional trenches should be made.

Figure 4.2. SDRW “Buryakivka”

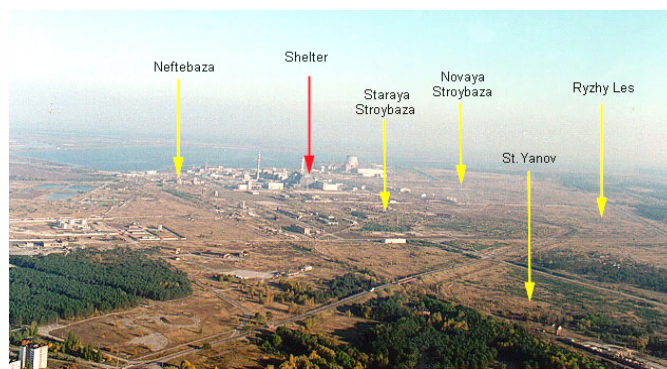


Source: Kilochytska.

Areas of RW temporary storage location

The sites for RW temporary location (STLRW) are in the territories adjacent to the ChNPP (see Figure 4.3) where in 1986-1988 decontamination of the area was conducted with localisation of decontamination waste in simple trenches, with no engineering barriers. It is considered that about 1 000 trenches and clamps are concentrated in nine STLRW on a total area of about 10 km^2 . More than half of the area of the STLRWs was not investigated. STLRW waste includes: contaminated soil, equipment, metal, concrete, building materials, remains of houses and rubbish.

Figure 4.3. Areas of radioactive waste temporary storage location



Source: Kilochytska, 2013.

According to the existing estimations, about $1.3E+06 \text{ m}^3$ of waste with total activity of $1.7E+15 \text{ Bq}$ is localised in STLRW. See Figure 4.4 for examples. Generally, this is low-level waste and waste with activity below the exemption level. Practically all the waste contains alpha nuclides; some parts of the waste are classified as long-lived. All the STLRW is situated in a territory with a high groundwater level; about 100 trenches with waste are flooded constantly or periodically, and radionuclides freely enter groundwater because of the absence of protective barriers.

Figure 4.4. Sites for temporary localisation of radioactive waste



Source: Kilochytska, 2015.

Conclusions and lessons learnt

- Before the Chernobyl accident, there was no previous worldwide experience in managing large amounts of emergency radioactive materials. Disposal of radioactive waste from the ChNPP accident was conducted in extreme conditions without adequate waste isolation technology and classification and registering of waste (its amount and activity). The possible environmental impact of storage sites was not considered. Even today, the majority of storage facilities require in-depth investigations.
- SDRW “Pidlisny” and SDRW “The 3rd stage of the ChNPP” were constructed immediately after the accident. They are not in operation now; RW retrieval and re-disposal are needed.
- Re-disposal is possible in a deep geological disposal facility (GDF); before this facility is ready, updating/development of safety barriers in existing facilities are needed.
- There are many sites for RW temporary location – STLRW. Not all waste from these sites should be re-disposed of. Decisions related to retrieval of RW from STLRW should be based on investigations data and safety assessments.

RW that are concentrated in the natural and artificial objects of the OS and the ChNPP industrial sites and adjacent territory

According to the existing estimations, about $15\,000 \text{ m}^3$ of RW remain in an active layer of soil of the local zone of the SO after completing the work on decontamination of the territory. According to the data from drilling and gamma-ray logging investigations, RW are concentrated mainly in the layer of disposed soil with a thickness of 10-30 cm (and in some places, considerably more).

Low- and intermediate-level waste includes contaminated and mixed pre-accident soils, contaminated concrete blocks and plates, metal structures, fill (crushed stone, sand, etc.) and construction waste.

In total, $500\,000 \text{ m}^3$ of RW of low- and intermediate-level waste are on the ChNPP industrial site. They are contaminated soils, metal, concrete, equipment, various materials, etc.

A considerable amount of radioactive material is concentrated in the cooling pond of the ChNPP. Its bottom sediments contain over $0.2\text{E}+15$ Bq. Certain parts of the cooling pond sediments are categorised as RW.

In the temporary waste storages on the ChNPP site are located:

- solid RW – $2\,500\text{ m}^3$, with activity of $1.40\text{E}+14$ Bq;
- liquid RW – $20\,000\text{ m}^3$, with activity of $3.85\text{E}+14$ Bq.

RW of the shelter object

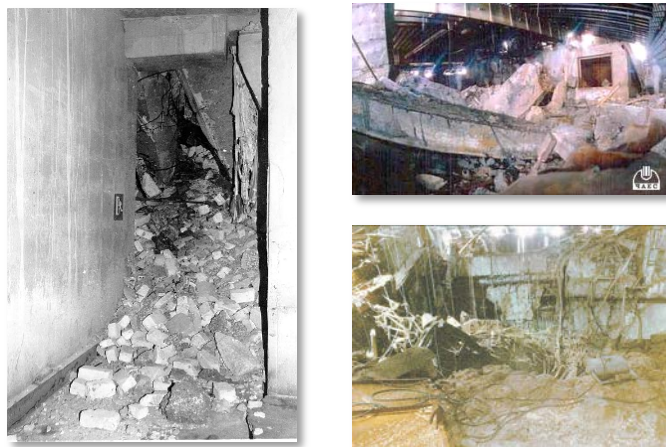
The shelter object was constructed under extreme post-accident conditions and has been performing its protective functions for almost 20 years. The key feature of the shelter is its potential hazard, which is significantly greater than permitted by regulations and rules for facilities containing nuclear-hazardous and radioactive materials. Generally, from the point of view of radiation safety, the shelter is actually an open source of alpha, beta, gamma and neutron radiation, which, with respect to its radiation characteristics, has no analogies in world practice. It can be considered as an interim barrier to fissile nuclear-hazardous materials and high-level waste (HLW), with a practically uncontrolled situation inside the facility.

The current status of the shelter is specified in Annex RSSU-97 “Radiation protection from sources of potential radiation” (RSSU-97/D-2000) – sites for surface storage of non-arranged RW.

From $400\,000$ to $1\,740\,000\text{ m}^3$ of RW are located in the shelter object and at its site. See Figure 4.5 for examples. At the beginning of 2005, their total activity was about $4.1\text{E}+17$ Bq.

Over 10% of the total amount of the SO RW is HLW, large amounts of which are concrete, metal structures and equipment, and backfill materials of the reactor. Over $2\,800$ t of HLW are fuel-containing materials (FCM), including lava-like FCM, fragments of the reactor active zone, reactor graphite and fuel dust.

Figure 4.5. RW inside the shelter object



Source: National Report of Ukraine, 2011.

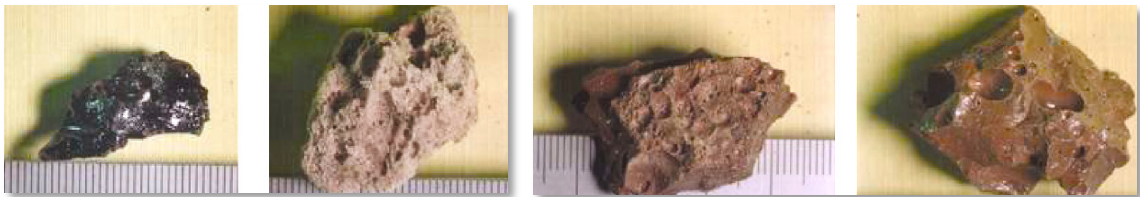
At the SO, constant accumulation of water from the atmosphere, condensation and technological origin takes place. Liquid radioactive waste (LRW) was produced from the interaction of water with radioactive materials. Annually, up to 900 m^3 of LRW are pumped from the accessible SO rooms, and transported to the system for treatment and storage of liquid RW at the ChNPP.

Fuel-containing materials located currently inside the shelter

Varieties of nuclear fuel formed during the active stage of the accident are currently in the shelter object. There are three varieties of FCM, containing the bulk of irradiated nuclear fuel (INF):

- fragments of the reactor core (FRC);
- fuel particles (fuel dust);
- lava-like fuel-containing materials (LFCM).

Figure 4.6. Varieties of fuel-containing materials: Black LFCM, pumice and brown LFCM



Source: National Report of Ukraine, 2011.

Some examples are shown in Figure 4.6. Most of the FCM is found in the central hall and premise 305/2 under the reactor.

A significant part of nuclear fuel got into the reactor vessel and premise 305/2 under the reactor, where conditions for fuel heating up to high temperatures were created. Fuel fragments entered into reaction with structural materials: zirconium, metalwork, serpentinite filling of biological protection, sand and concrete and formed high-level LFCM.

LFCM spread over the premises, corridors, cable passages and other free channels and, when hardened, formed accumulations (Figures 4.7 and 4.8) at different elevations in the destroyed ChNPP unit 4. LFCM may contain up to 130 t of uranium INF, and a significant part of the inventory of radionuclides generated in the reactor. Therefore, LFCM is still present and is the main source of nuclear, radiation and radiological hazard.

Figure 4.7. Fuel-containing materials



Source: Kilochytska, 2013.

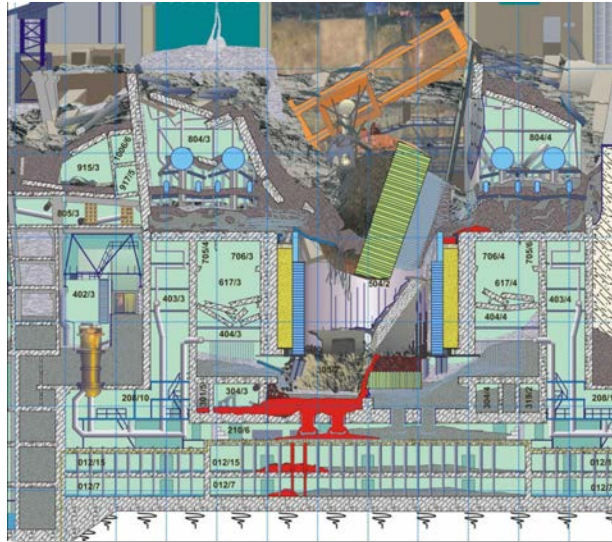
Assessment of the overall nuclear fuel amount that remained in destroyed unit 4 was based on studies on radiation fall-out, and now it gives grounds to consider that about 95% of nuclear fuel of the initial reactor loading is in the shelter object. Consequently, total activity of the radionuclides in the shelter object currently makes up approximately $4.8E+17$ Bq.

Recent investigations refining the geometry and spatial arrangement of LFCM accumulations have shown that there are two areas at the top elevations of ChNPP unit 4 where LFCM accumulation can be found (highlighted area in Figure 4.8).

Calculations have shown that the minimum amount of LFCM (1 t of UO_2) at the top elevations of the destroyed unit 4 is 15 t. This fact must be considered while developing strategies of INF removal from the central hall, in construction of the new safe confinement and in the course of further activities on the shelter object to convert it into an ecologically safe system.

Data suggest that high-uranium FCM accumulations are located in the south-east part of premise 305/2 (near the gap opening into premise 304/3), as well as in the vicinity of the burn-through towards premise 307/2.

Figure 4.8. Areas where FCM have accumulated



Source: National Report of Ukraine, 2011.

FCM are the main source of environmental emissions of radionuclides and, hence, the main source of radiological hazard within the shelter. It is well known that UO_2 pellets exposed to the air deteriorate in about 20 years. However, the most critical factor for the shelter can be deterioration of LFCM because most radionuclides are in this form of FCM.

Currently, the LFCM are demonstrating clear changes in strength properties, which are manifested by their cracking, destruction of big LFCM fragments and enhanced dust-formation capacity. Hence, a challenging problem is what critical changes can occur in LFCM over a prolonged period such as the next 50 years. Currently, there are two fundamentally different approaches to predicting changes in LFCM characteristics with time. In one approach, it is assumed a priori that LFCM characteristics are similar to silicate glass used for containing radioactive waste. Based on this assumption, a conclusion is drawn that radiation damages caused by alpha decay will initiate LFCM strength property changes no earlier than in 10 000 years. The authors attribute the basic causes of evident changes in LFCM to temperature drops, interaction with water, dust suppression compounds and other factors including external influences.

In another paper, scientists have investigated the basic characteristics of LFCM, and the influence of these factors in causing changes in LFCM properties with time. The basic conclusion is that there appears to be disordered areas created by recoil nuclei due to inner self-irradiation during alpha decay of transuranium isotopes. The increasing concentration of disordered areas (which are a source of occurrence of micro-fissures) can lead to sudden total destruction of LFCM. Such catastrophic destruction might occur in the next 50 years.

In addition, it was shown that submicron aerosols are generated on the surfaces of LFCM and INF, which can present a serious radiation hazard. The mechanism responsible for this phenomenon in LFCM can be a Coulomb explosion, which occurs during deceleration of alpha-particles. In spite of extensive investigations, to date there is no well-grounded prognosis on FCM behaviour. Hence, follow-up studies in this area are essential.

Conclusions and lessons learnt

- The most serious problem of shelter object is FCM in an uncontrolled state and with associated nuclear and radiation risks.
- Neutron activity and temperature of FCM should be monitored permanently to avoid a potential criticality incident (despite its low probability of occurrence).
- FCM were studied carefully during the first years after the accident. After that, there were no detailed investigations of physical and chemical characteristics because of high levels of dose exposure rates and costs.
- Investigation of degradation and destruction of FCM are real challenges for Ukraine, taking into account that the National Strategy of Shelter Object transformation into an ecologically safe system is that FCM should be retrieved during the lifetime of the new safe confinement (NSC).
- A strategy of long-term monitoring shall be developed and implemented including:
 - monitoring of physical and chemical characteristics of FCM to study the dynamics of destruction processes;
 - investigation of submicron aerosols production because of FCM destruction;
 - investigation of the behaviour of radioactive aerosols inside the shelter, especially after NSC is installed above the old shelter and temperature/humidity will change.
- Knowledge of physical, chemical and other characteristics of FCM is essential for further development of technology and methodology of FCM extraction.

Liquid radioactive waste in the shelter object

The process of moisture ingress into the facility and accumulation at the bottom levels of the unit in the form of LRW present another factor capable of destabilising the current state of nuclear, radiation and ecological safety of the shelter object. Moisture penetrates into the SO as a result of precipitation, condensation and operation of the dust suppression system. Precipitation gets into the SO through cracks in the roof and facility walls.

Having reached lower elevations in the SO, moisture interacts with structural and fuel-containing materials, which leads to transfer of radionuclides to the water. Such uncontrolled leakages result in medium-level liquid radwaste accumulating at the bottom levels, which uninterruptedly escapes from the SO in the following two directions – to the north and north-east. The north stream is accumulated in SO premise 001/3. Up to 300 m³ of LRW, which is from 60% to 70% of the total amount of water in the shelter object, is permanently found in these premises. Leakages from the north zone of the pressure-suppression pool, central and south-east premises of the shelter object, as well as from the cascade wall, flow together here. The stream of 700-900 m³ annually leaks further through the dividing wall to unit 3 premises and is pumped to the ChNPP chemical shop for temporary storage and treatment.

Radionuclide concentrations in LRW from premise 001/3, including transuranium elements (TUE) tends to increase with time. The major contributor (up to 80%) of total alpha activity of LRW is americium-241. Contribution of plutonium isotopes is less than 30%.

The south-east LRW stream of 300 m³ gets into premises 017/2 and 018/2 and leaks to ChNPP unit 3 premises. The dynamics of the average annual radionuclide concentration, including TUE, in LRW of this stream is similar to the one observed in the north stream.

Part of the activity transported with water leakages is concentrated in the form of sludges. Their amount, for instance, in premise 001/3 is estimated as 100 m³, with a total weight of about 150 t. Radionuclide concentration in sludges is two to three orders of magnitude higher than concentrations in the water. Sludges drying out in the event of leakages ceasing and continued LRAW pumping from premise 001/3 may result in significant exceedance of permissible radioactive airborne particle concentrations in these and other SO premises.

Shelter object LRW are characterised by a high concentration of organic compounds, including oil products, surfactants and film-forming compounds, as well as TUE activity which does not allow their treatment using the existing chemical shop facilities. This leads to growth of organic compounds and TUE concentrations in the ChNPP LRW storage facilities. When particular TUE and organic compounds concentrations are reached, it will not be possible to process such waste at the Liquid Radwaste Treatment Plant which is under commissioning.

Conclusions and lessons learnt

- radioactive water from the shelter object requires removal of alpha nuclides (so-called transuranium elements) and organics to allow further treatment of this type of liquid RW;
- an additional RW treatment facility is under development for removal of alpha nuclides and organics from the water of the shelter object;
- secondary waste management (as a product of this facility) is an additional problematic issue for the ChNPP.

Shelter object radioactive aerosols

The radiological hazard of radioactive aerosols of Chernobyl origin lies in highly toxic transuranium and long-lived isotopes they contain, in particular plutonium and americium isotopes.

Contamination of air inside and outside the shelter object may occur due to the following set of processes:

- dust raised from the facility premises surface;
- dust generation in the course of construction and installation works;
- dust generation and resuspension caused by collapse of facility structural members;
- degradation of fuel-containing materials due to radioactive process and ageing of materials;
- dust generation and resuspension caused by leaching of radioactive substances, solution drying and salt deposit formation.

Regular monitoring of the radionuclide concentrations in uncontrolled airborne particulate releases has been done since 1992.

Reduction of uncontrolled releases from the SO has been observed over the recent years. Commissioning of the modernised dust suppression system (MDSS) in 2004-2006 played a significant role in this process, since it expanded the zone of radioactive dust control to the entire area under the SO roof. A total of 220 tonnes of dust suppression solution (48.8 tonnes of dry residue) were supplied to the SO inner space over the period 2006-2009.

MDSS commissioning has reduced radioactive aerosol release from the SO by more than a factor of two, and removable surface contamination in the SO inner space reduced by more than four orders of magnitude. The protective polymer coat covers almost the entire area inside the SO and functions as a containment, precluding transport of radioactive substances into the environment.

Evaluation of the SO impact on the environment presents a complex and multi-factor issue.

Radioactive aerosols that are being currently transported from the facility have been generated during the accident and located in the form of dust inside the SO or have been newly generated in the course of physical and chemical degradation of fuel-containing masses. Air contamination monitoring may serve as an indicator of FCM destruction, including the accumulations that are beyond direct control. Such information would be useful during the NSC construction and commissioning. Furthermore, it is important to know the physical, chemical and radiochemical compositions of aerosols, the processes that formed them, their migration pathways (transport and settlement), as well as types of dissolution in the human respiratory system for determination of individual and collective protection. Consequently, monitoring of radioactive aerosols both in the environment and inside the destroyed unit remains valid in terms of radiological protection and understanding of the processes ongoing in the SO, in particular assessment of the status of nuclear fuel remains and lava-like fuel-containing materials.

Conclusions and lessons learnt

On the one hand, organic liquids are needed for dust suppression purposes and to decrease the level of radioactive aerosol activity. On the other hand, they create problems related to management of water from the SO.

Temporary storage on Fukushima Daiichi

Temporary storage classification of collected debris

Debris (such as rubble from damaged reactor buildings) collected in clean-up operations on-site are classified by surface dose rate, with boundaries at 30 mSv/h, 1 mSv/h, and 0.1 mSv/h, and moved by heavy machinery to each storage area.

Four categories and five storage areas are shown in Figure 4.9 below, from the solid waste storage building for higher-surface dose waste, which existed before the earthquake, soil-covered temporary storage facility and temporary storage facility, which were installed after the accident, and open areas, such as sheet-covered storage area and outdoor collection area. The outdoor collection area is for waste of less than 0.1 mSv/h.

Each temporary storage provides shielding from direct radiation and prevents radioactive waste dispersion.

Figure 4.9. Debris collection at Fukushima: Waste categories and temporary storage areas



Source: TEPCO, 2014.

Table 4.2. Temporary storage categories at the Fukushima Daiichi nuclear power plant

		Surface dose rate of debris			
		Over 30 mSv/h	1-30 mSv/h	0.1-1 mSv/h	Less than 0.1 mSv/h
Storage approach	Shielding	Containers and building	Concrete wall, soil and containers	None	None
	Prevent dispersion	Containers	Tent, soil and containers	Sheet cover	None
Temporary storage method		Container storage, solid waste storage building	Container storage, temporary storage facility, soil-covered temporary storage facility	Sheet cover	Outdoor collection

Source: TEPCO, 2014.

For the >30 mSv/h debris, containers and the building wall provide shielding. Containers prevent dispersion.

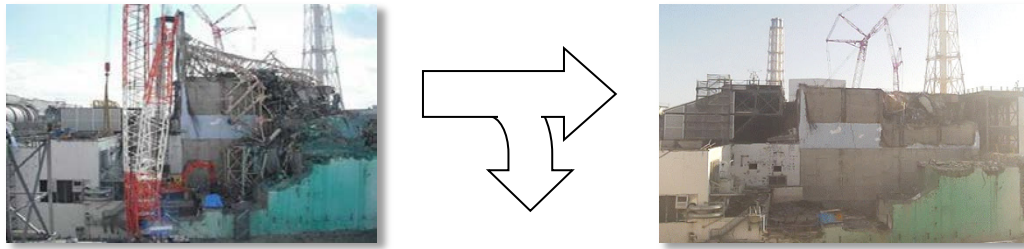
Concerning <30 mSv/h debris, shielding is provided by the concrete wall in the tent and soil on the roof of soil-covered temporary storage facility. The tent and cover of the storage prevents dispersion.

Prior to removing fuel from the spent fuel ponds, debris such as pieces of concrete and equipment have been removed in order to prevent further dispersion of radioactive materials and to improve working conditions. Debris removal and some of the waste storage facilities are shown in Figure 4.10.

Figure 4.11 shows details of the soil-covered storage facility. It is a trench with a bentonite layer and impermeable sheet at its base. Protective and impermeable sheets are placed on the debris and covered with soil. The soil cover greatly reduces external dose rate from this facility.

Figure 4.10. Images of debris removal work and some soil-covered storage facilities

Progress status of debris removal from the top of unit 3 building



Removed debris

Soil-covered temporary storage facility



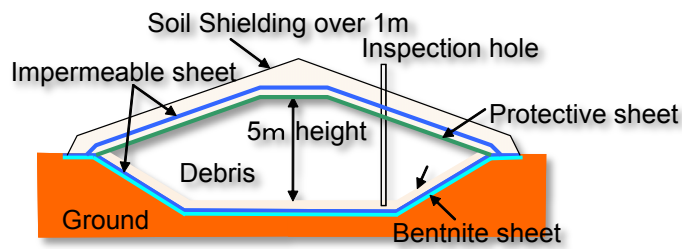
Solid waste storage building



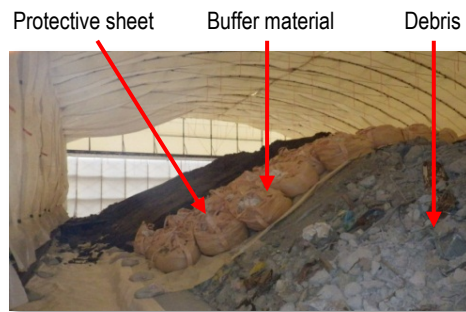
Source: TEPCO, 2014.

Figure 4.11. Details of the temporary soil-covered storage facility

(cross-sectional view)



Condition of 1st facility (photographed on 27 March 2014)



Interior of the 1st facility (as of 1 November 2014)

Source: TEPCO, 2014.

Storage areas in the overview map of Fukushima Daiichi

As of the end of 2015, about 173 000 m³ of concrete and metal waste and about 85 000 m³ of felled trees had been generated. Felled trees are produced by clearing areas for installation of facilities for water treatment facilities, tanks and so on.

The waste storage areas are scattered, because of the need to work urgently. There was not enough time to make and carry out a comprehensive and rational plan. A summary of the waste storage areas and their capacities is given in Table 4.2.

Estimation of waste generation and improving land use

More than 700 000 m³ of waste is expected to be produced on-site in the coming ten years. Installing waste volume reduction facilities (cutting and crushing materials) and additional solid storage buildings is planned. Locations of future storage areas will be chosen to use the site land effectively and to optimise the efficiency of work.

Lessons learnt from Fukushima Daiichi accident

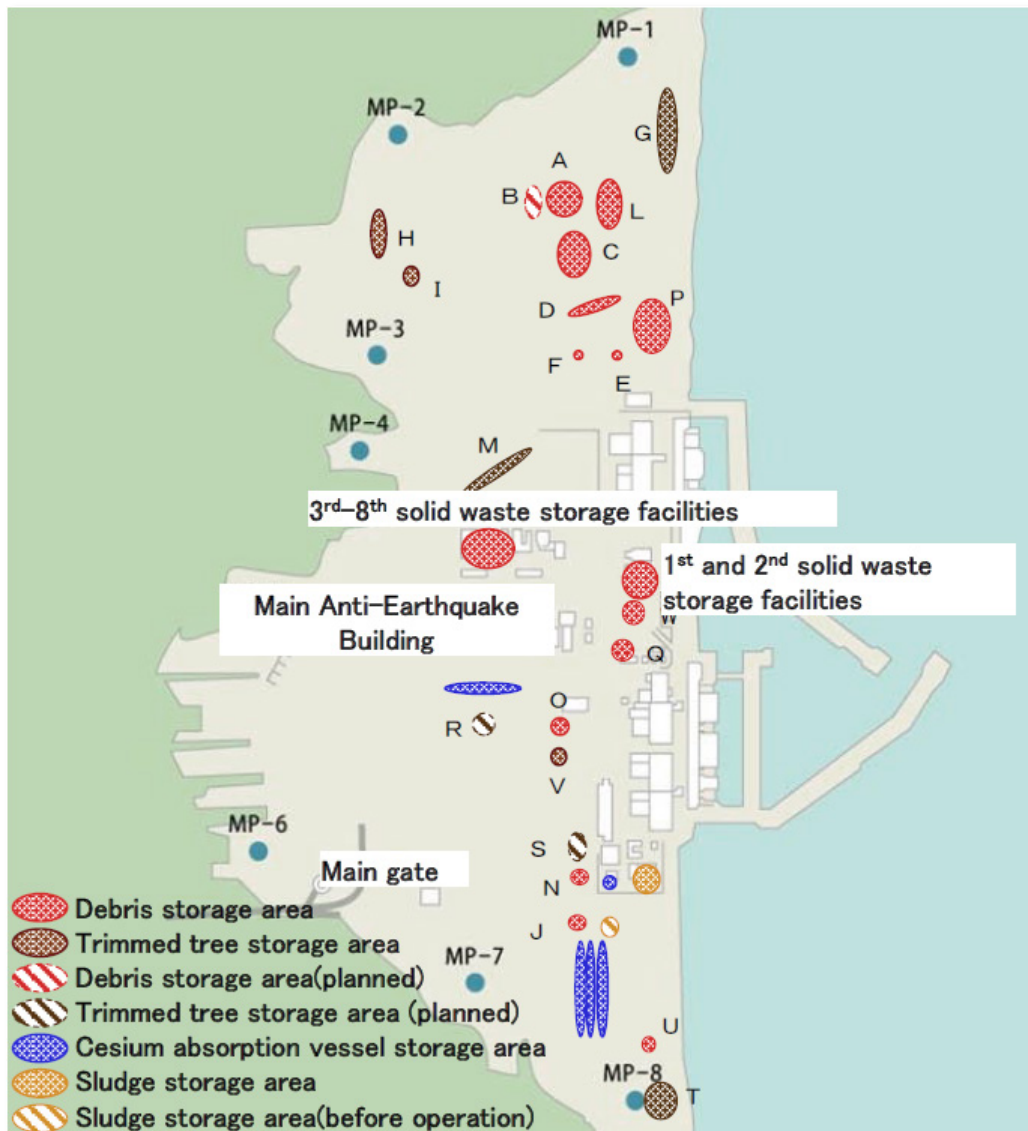
TEPCO should estimate the physical amount of solid radioactive waste expected to be generated in the coming ten years or so and take the necessary measures to suppress generation and reduce the volume of solid waste. Based on these efforts, TEPCO formulated a long-term storage plan within the fiscal year 2015 on the premise of adopting storage in a temporary storage area, systematic introduction of facilities equipped with shielding/dispersion-prevention functions, and appropriate storage supported by continuous monitoring.

Table 4.3. Summary of the waste storage areas and their capacities
(as of the end of 2015)

	Categories	Storage location*	Storage method	Storage quantity (m ³)	Storage capacities (m ³)
Debris	Less than 0.1 mSv/h	C	Outdoor accumulation	54 900	177 900
		F		5 000	
		J		3 000	
		N		3 800	
		O		26 200	
		P		22 000	
		U		700	
	0.1-1 mSv/h	D	Sheet covering	2 600	57 300
		E		7 200	
		P		600	
		W		21 000	
	1-30 mSv/h	L	Temporary storage facility	12 000	27 700
		A	Temporary storage tent	1 100	
		E	Container	300	
F		600			
Q		5 700			
Over 30 mSv/h	Solid waste storage building	Container	6 200	12 000	
Debris total				172 900	
Trimmed trees	Tree trunk roots	H	Outdoor accumulation	14 700	81 500
		I		10 500	
		M		39 100	
		V		2 400	
	Tree branch leaves	G	Temporary storage for trimmed tree	7 300	24 900
		T		11 100	
		Trimmed trees total		85 100	

* See Figure 4.12.

Source: TEPCO, 2016.

Figure 4.12. Waste storage area at Fukushima Daiichi NPP

Source: TEPCO, 2016.

4.4. Lessons learnt for post-accident physical and chemical characterisation

At the present time, some accident waste is being stored in temporary near-surface storage facilities on the Chernobyl and Fukushima Daiichi sites. The current policy for the long-term management of this waste is that it will be transported and disposed of in appropriate off-site surface or deep geological disposal facilities, depending on the categorisation of the waste. Disposal facilities for the accident waste have not yet been identified, but it is to be expected that each disposal facility will set requirements, in the form of waste acceptance criteria (WAC), on waste that can be accepted for disposal. The WAC will define both the radiological and physical/chemical nature of the waste or waste form. Recording the physical and chemical nature of the accident waste transferred to temporary storage facilities is valuable, as it provides information to enable the waste to be assessed against future WAC. It also provides information that can be used to assess approaches to volume reduction and waste processing/treatment.

Lessons learnt from the TMI-2 accident

The TMI-2 case study highlighted the challenges associated with: discharging accident-generated water, even when cleaned to radionuclide concentration levels below the site discharge permit; the need to develop methods to locate and quantify fuel that was displaced from the original core region, and; the need for a range of containers for the different forms of damaged fuel (fine particles, rubble, solidified mass, partial fuel assemblies) present.

Lessons learnt from the ChNPP accident

Management of FCM is identified as the most serious problem in the shelter object, as these materials are in an uncontrolled state with associated nuclear and radiation risks. FCM are the main source of environmental releases of radionuclides from the shelter object. FCM were studied carefully during the first years after the accident; after that there were no detailed investigations of their physical and chemical characteristics because of high dose exposure rates and costs. Physical degradation of some of these lava-like FCM is of concern because of the potential to produce radioactive aerosols, which would be highly mobile. The Ukrainian programme recognises the need to implement permanent neutron activity and temperature monitoring of FCM, to assess the likelihood (albeit low) of criticality, as well as monitoring of physical and chemical characteristics to study the dynamics of FCM destruction processes. In addition, the need to retrieve FCM during the lifetime of the new safe confinement, and process them to waste forms suitable for long-term management, has been recognised.

Liquid radioactive waste produced in the shelter object is characterised by high concentrations of organic compounds, including oil products, surfactants and film-forming compounds used for dust suppression and to decrease radioactive aerosols. Removal of organics (and elevated concentrations of transuranic elements) is required to allow further treatment of the water.

Lessons learnt from the Fukushima Daiichi accident

Key lessons for planning include the need to: estimate the physical amount of solid radioactive waste expected to be generated in the coming ten years or so; implement measures to minimise the volume of solid waste arisings, and; formulate a storage plan for this waste.

4.5. Recommendations for post-accident physical and chemical characterisation

The key issue for physical and chemical characterisation of accident waste is to determine the extent to which the accident has affected the physical and chemical characteristics of the solid and liquid waste that will be, or has been, produced. It will also be necessary to estimate the volumes of solid and liquid waste that will arise, and the times at which it is expected to arise. For example, any explosion within reactor buildings could result in debris being ejected from the reactors and in the release of radionuclides to the environment (to air and/or groundwater). This could generate large volumes of lower-activity contaminated materials around the reactor: for example, contaminated debris, vegetation, soil and groundwater.

Work should be undertaken to determine whether the physical and chemical characteristics of “accident” waste streams differ from those produced in by normal operations. We assume the physical and chemical nature of the latter waste will be well understood. This will be an important input into the Integrated Waste Management Strategy for any accident site.

To meet the requirements above, it will be necessary to acquire sufficient data as soon as possible to characterise materials and waste. To ensure appropriate data are

collected, there must be a clear understanding of what the data will be required for; for example, in the context of assessing how the physical and chemical characteristics of waste can affect disposability or waste treatment options. A degree of iteration in the process should be expected. At a detailed level, the following should be undertaken:

- determine the “materials inventory” for the waste that will be produced from the decommissioning programme;
- determine whether the waste contains any hazardous non-radioactive substances such as toxic metals or asbestos, and assess the implications of these hazardous substances on waste handling, conditioning, interim storage and disposal;
- determine whether the waste contains chemical complexing or chelating agents, and assess the impacts of such complexing or chelating agents on waste conditioning and disposal.

It is probable that new procedures will have to be developed to specify and collect this information for waste produced after an accident. Lastly, it is important that information on physical and chemical characteristics is generated and held in a way that allows it to be integrated with other characterisation data, principally radiological information.

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5. Radiological characterisation

5.1. Introduction

In this chapter, we first summarise some aspects of “state-of-the-art” radiological characterisation approaches and techniques that could be relevant to the characterisation of accident-related waste. We then present the experience of the Three Mile Island 2 (TMI-2), the Chernobyl nuclear power plant (ChNPP) and the Fukushima Daiichi programmes in post-accident radiological characterisation of materials and waste. The chapter concludes by identifying the lessons learnt from the accident case studies and makes recommendations for radiological characterisation of materials and waste in future accident situations.

5.2. State of the art

Radiological characterisation is important at many stages of the waste management and decommissioning process. For example:

- To determine the locations, identities and concentrations of radioactive contaminants in engineered structures and components, soils, vegetation and waters as part of the characterisation and survey stage of the programme. Such characterisation generally involves a combination of on-site mobile equipment and off-site analytical testing laboratories.
- As part of the waste sentencing process, to ensure correct sentencing is achieved. In the absence of good radiological characterisation at this stage, there will be a tendency to assign waste to higher categories than required, leading to impacts on the chosen interim storage and disposal solutions. Such an approach would not achieve the requirement to minimise the waste produced. This phase of characterisation generally takes place on-site, during decommissioning, and subsequently in the waste management facility.
- As part of the validation process, after completion of demolition, remediation or decontamination activities.
- To ensure safety throughout the programme. This can include issues as diverse as ensuring criticality safety, managing workforce doses and building understanding of any discharges to the environment.

In this section, we summarise some aspects of “state-of-the-art” radiological characterisation that are relevant to the characterisation of accident-related waste. We consider the following areas:

- identification of radionuclides for analysis;
- approach to defining the radionuclide fingerprint¹;

1. A radionuclide fingerprint is an estimate of the anticipated radionuclide mix, expressed as percentages of the various nuclides, in a material or waste stream. The radionuclide fingerprint is used to infer and quantify the presence of other nuclides by measuring only one nuclide, or a limited number of nuclides.

- building confidence in the radionuclide fingerprint and inventory;
- approaches to radiological characterisation;
- approaches to statistical modelling and sampling;
- geostatistical data processing.

More comprehensive treatments of radiological characterisation are given in existing NEA reports:

- “Radiological Characterisation for Decommissioning of Nuclear Installations” (NEA, 2013).
- *R&D and Innovation Needs for Decommissioning Nuclear Facilities* (NEA, 2014).

The reader is referred to these references for a more detailed description of some of the topics presented below.

Selection of radionuclides for analysis

It is unreasonable to expect waste producers to undertake costly and time-consuming programmes of work for the purpose of demonstrating the presence (or absence) of every known radionuclide in each waste stream. Therefore, the first step in the radiological characterisation programme is to identify the radionuclides for which information is required. Those countries that have accumulated (or are accumulating) radioactive waste have developed criteria that the waste must satisfy in order to be accepted at waste management facilities. The criteria differ in detail from country to country, but in general terms they are derived from the following requirements:

- to enable decisions to be made about conditioning and packaging waste;
- to ensure the safety of the waste during handling, transport, storage and disposal.

Radionuclides in the latter group are often termed “safety-relevant” radionuclides. Typically the list for this group will exclude radionuclides that are present in insufficient quantities to exceed safety criteria.

Radiological characterisation requirements are specified in waste acceptance criteria for existing storage and disposal facilities, and in the information requirements laid out by the organisations responsible for developing new disposal facilities. For new disposal facilities, the inventory information is used together with information on proposed waste conditioning and packaging solutions to minimise the likelihood that the waste package will be unsuitable for disposal to a future facility. As an example of safety-relevant radionuclides, Radioactive Waste Management Ltd (the organisation responsible for developing geological disposal facility [GDF] for higher-activity waste in the United Kingdom) considers a list of 112 radionuclides for intermediate-level waste. Similar lists have been derived in other countries and for other waste categories (e.g. by Low Level Waste Repository Ltd in the United Kingdom for low-level waste [LLW]).

Approach to defining the radionuclide fingerprint

The concept of a radionuclide “fingerprint” is generally applied to radioactive waste. The approach takes account of radioactive ingrowth, particularly for alpha nuclides. In normal operations, a fingerprint is generated to describe the average radionuclide composition of each waste stream. Once it has been established, waste is generally characterised and sentenced based on the analysis of a small number of radionuclides (for example, Co-60 or Cs-137). The concentrations of other radionuclides are derived

from knowledge of the ratios in the fingerprint between the unknown and measured radionuclides. The typical approach to estimating radionuclide activities in materials or waste streams involves a combination of:

- calculation of the expected radionuclide concentrations, given understanding of material chemistry, irradiation history of the material and fuel burn-up (“calculation”);
- radiochemical analysis of samples (“measurement”).

Many types of materials and waste are heterogeneous. For practical waste management purposes, the concept of an “averaging volume” is often used to describe a volume over which the contained radionuclides are considered to be homogeneously distributed. It is important to establish the averaging volume at an early stage in the decommissioning programme, particularly for high volume lower-activity waste, as it influences the approach chosen to radiologically characterise the material.

Building confidence in the radionuclide fingerprint and inventory

The total activity and radionuclide composition of many materials will be heterogeneous. It is important to characterise this heterogeneity in order to correctly assign a radionuclide fingerprint and inventory to the material. A prerequisite to designing a sampling and analysis plan to adequately characterise such materials is to have a good understanding of the processes by which the material has become radioactive. For example:

- “Hot spots” formed from fuel debris or similar will be present within some waste. Detection of such hot spots is required to reduce waste volumes or lower the category of waste in which hot spots occur.
- Some porous materials such as concrete have been contaminated at their surfaces. Depending on the mobility of the radionuclide and the duration of contact, some radionuclides will have penetrated into the porous material. This results in some of the subsurface material, as well as surface material, becoming contaminated. Characterisation will be required to determine the minimum depth of material to be removed from the surface, for example by high-pressure washing or scrubbing, to allow free release of the remaining material.
- Neutron fluence will have varied across structures in the reactor, leading to gradients in activation products in some materials, such as the concrete in the primary containment vessel. Knowledge of these activity gradients will enable waste to be minimised and appropriately sentenced.
- Contamination in soils will generally be highest close to the location at which contaminants enter the soil. More mobile contaminants, such as Sr-90 will move further from the point of entry than less mobile contaminants such as Cs-137 and actinides. Knowledge of depths of contamination and contaminant mobility will enable the minimum amounts of soil to be removed during site restoration.

Research and development (R&D) needed to build confidence in the radionuclide fingerprint fall into two broad areas:

- development and/or application of additional analytical techniques;
- development of sampling and data processing strategies.

These are discussed in the following sections.

Approaches to radiological characterisation

Laboratory-based radionuclide characterisation is well-established. The challenge for the decommissioning programme is to develop a laboratory analytical testing strategy that maximises sample throughput while ensuring the necessary information is collected.

Detailed radiochemical characterisation of representative samples should be undertaken as early as possible in the decommissioning programme to determine the radionuclide fingerprint of the material and for comparison with existing estimates of radionuclide inventories. Once the radionuclide fingerprint for a particular waste stream or material has been established, waste can be sentenced on the basis of analysing for a small number of “easy-to-measure” radionuclides. As discussed above, it is necessary to be aware that radionuclide composition may vary within the material; key radionuclide ratios that could provide information on the stability of the fingerprint should be identified and determined in a small proportion of samples. Examples of potentially applicable ratios include Co-60/Cs-137, which would measure the relative proportion of activation products and contamination, and Sr-90/Cs-137, which would measure the extent to which contaminants with different chemical behaviours have become separated by transport through materials such as concrete or soil.

The sample throughput requirements (by determinant and matrix) should be established by reference to the decommissioning programme and the numbers of samples required to adequately characterise materials/waste. Suitable techniques and sufficient analytical equipment should be put in place to meet these requirements.

International experience identifies that substantial cost savings and time savings may be achieved through the application of rapid, field-usable techniques to: i) locate, identify and quantify radioactive contaminants in reactor structures and components, demolition debris, soils and water; ii) survey areas during decommissioning to establish the decommissioning endpoint, and iii) to sentence the waste produced. Site-based radiological characterisation techniques that are currently routinely used for these purposes include:

- Static, mobile or hand-held instruments for measuring ionising radiation. Examples include: alpha/beta “contamination” probes; gamma spectrometers; passive and active neutron measurements, and; dose rate probes. Dose rate probes have sometimes been used to derive estimates of concentrations of the principal gamma-emitting radionuclides, although this is not a common application.
- Large area beta or gamma detectors, which can be deployed by foot or in vehicles to survey large areas of open ground. A number of systems are commercially available and used both at the characterisation/survey and dismantling/remediation phases of the decommissioning programme.
- Static detectors used for waste sentencing purposes: drum monitors, bucket monitors, conveyor systems, etc.
- Remotely operated vehicles with mounted analytical equipment, used at the characterisation/survey stage to work in high dose rate areas and inaccessible environments.

In addition, the use of gamma cameras to visualise the distribution of gamma-emitting radionuclides in decommissioning operations is becoming more widespread (for example, see IAEA [2011] and references therein).

The R&D strategy for any decommissioning programme should identify where suitable “off-the-shelf” field characterisation techniques are available to support decommissioning and waste characterisation, and should identify areas where further R&D is necessary. Approaches being developed by overseas R&D programmes should be reviewed against programme-specific requirements, and technology readiness levels

assigned to determine those technologies that are close to maturity. There are a number of areas where recent developments in analytical technologies or approaches could be appropriate to accelerate the decommissioning activities and improve characterisation information. See Section 2 of NEA, 2014 for additional information:

- Application and further development of rapid characterisation techniques for materials that do not have a reliable gamma fingerprint. As part of Nuclear Decommissioning Authority's Direct Research Portfolio, a literature review of potential technologies has been undertaken (Serco Technical Services, 2010a) and research priorities for the UK identified (Serco Technical Services, 2010b).
- Development of on-site mobile laboratories that have radiochemical separation facilities to allow rapid ex-situ analysis of key "difficult-to-measure" radionuclides.
- Development of dual beta/gamma probes (an emerging technology for the simultaneous measurement of beta and gamma radiation), which would enable identification of hotspots (for example fuel debris) in high volume waste in the presence of elevated gamma background.
- Application of approaches to derive estimates of radionuclide activity from dose rate measurement. The approach involves the application of radionuclide transport codes (shielding calculations) and knowledge of the radionuclide fingerprint.
- Development of visualisation techniques other than gamma cameras to image radiation fields. For example, alpha cameras (see IAEA, 2011), positron emission tomography and muon scans (IRID, 2015).
- Development of approaches to sample in and around difficult to access structures such as drains and pipes.
- Development of approaches to characterise the depth of penetration of radionuclides into concrete. Conventionally, depth distribution of radionuclides is determined by obtaining and sub-sectioning cores, followed by radionuclide analysis; this is generally slow process. More rapid destructive sampling and analysis solutions have also been developed, such as TruPro® (NMNT, 2015). Potential areas for R&D are discussed in Section 2 of NEA, 2014.
- Application and further development of in situ sensors for remote sensing of radionuclides in groundwater. In situ reusable radiochemical sensors have been developed and trialled in the United States for groundwater monitoring of Sr-90, Tc-99 and uranium on nuclear sites. Work is also ongoing to develop a laboratory prototype of a tritium monitor for proof-of-concept performance testing. See Section 6 of NEA, 2014 for further details.

Approaches to statistical modelling and sampling

There is substantial guidance available on the design of sampling programmes. Data quality objectives (DQO), developed by the US Environmental Protection Agency (EPA), is the most commonly used framework and is described in processes such as Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM, 2001). A series of logical steps is presented, which guides users on how to plan a campaign of data acquisition while making effective use of resource. The process is intended to be both flexible and iterative, and can be applied equally to decision-based studies (e.g. compliance/non-compliance) or estimation (e.g. ascertaining mean contaminant concentration). The DQO process forces data suppliers and data users to consider the following questions:

- What decision has to be made?
- What type and quality of data are required to support the decision?
- Why are new data required for the decision?

- How will new data be used to make the decision?
- How confident can data suppliers and data users be in those decisions?

Most stages in the DQO process are qualitative, and it is good practice to apply these aspects to any decommissioning activities. The final stage is quantitative and enables the number of samples required to reach a decision with a certain level of confidence to be calculated. A large number of statistical tests are available to do this; for example as implemented in software such as visual sampling plan (VSP), which is used to design sampling programmes and analyse data for MARSSIM. It is valuable to use statistical approaches in the design and analysis of sampling programmes, but it is important to ensure that the statistical model being used is an appropriate representation of the system.

The standard statistical approach, for example as described in MARSSIM, is to assume that the contaminant is randomly distributed throughout the material (“homogeneous distribution”). A key requirement for this approach is to choose a volume of material for survey such that this is a realistic representation. The challenge here is that it can sometimes be difficult to justify the assumption of homogeneity at scales larger than the averaging volume. As a consequence, additional “targeted” samples are sometimes taken to supplement the dataset and attempt to capture some of the known heterogeneity in the material. Typically, this approach involves collecting targeted samples from locations where radionuclides concentrations are known to be higher (i.e. it involves “expert judgement”). If not appropriately compensated for in the analysis, this can introduce bias in the calculated results.

Geostatistical approaches, which explicitly recognise spatial variability, have been developed in recent years and are discussed further in the next section.

Geostatistical data processing

Geostatistical techniques are designed for describing and modelling spatially correlated phenomena. Spatial correlations of relevance to radiological characterisation on nuclear sites include the distribution of radionuclides in engineered materials such as reactor graphite and concrete structures (Desnoyers and Dubot, 2011), and the subsurface distributions of radionuclides in contaminated soils. The aim of geostatistical techniques is to optimise radiological characterisation and improve understanding of the processes being investigated. In the context of contaminated materials, geostatistical approaches can provide estimates of waste volumes and associated uncertainties, and can determine the likelihood of encountering contamination at specific locations. The latter is of particular relevance when designing remediation schemes. Geostatistical techniques can also be used to optimise the sampling process (i.e. to ensure appropriate numbers of samples are taken from appropriate locations) and any subsequent long-term monitoring programme.

Kartotrak is an example of a computer program that has been developed for geostatistical analysis of data. It was originally developed by CEA and was subsequently commercialised by Geovariances (2015). The program includes an initial data quality control step to “clean” the dataset and remove erroneous data. For studies of contaminated materials (soils, graphite etc.), the program applies a range of geostatistical approaches to achieve the objectives described above (Desnoyers and Dubot, 2014).

It is important to recognise that geostatistical data analysis requires large amounts of data in order to build understanding of spatial variability. For example, a recent study of contaminated soil at the Sellafield site in the United Kingdom (Desnoyers et al., 2015) utilised over 14 000 measurements of each of gross alpha and gross beta activity. For this reason, geostatistical analysis may not be appropriate at the earliest stages of the programme, when only limited data are available.

5.3. Case studies

TMI-2 post-accident characterisation

Purpose

The TMI-2 post-accident assessment of plant conditions was a major challenge because conditions were unknown, unpredicted and unprecedented. Accurate information about plant conditions was the single most important factor in planning recovery and clean-up operations. Engineers were faced with the difficult task of determining the type and quantity of accident data needed to establish plant conditions amid pressures to acquire extensive data for the community and to avoid delaying clean-up activities. Management policy throughout the clean-up was that research should not significantly interfere with clean-up work. However, in many instances, research furthered clean-up progress by providing information crucial to planning and accomplishing clean-up operations. The EPRI Report “TMI-2 Post-accident Data Acquisition and Analysis Experience” (EPRI, 1992) contains a fuller understanding of how characterisation information was acquired and utilised (available at www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=NP-7156).

The general objectives motivating and affecting characterisation work were:

- Support clean-up operations – The first concern was the need to obtain data to support personnel protection, defueling, decontamination and waste management activities. This was in addition to the characterisation work required by normal plant operating specifications (required in diminishing degrees as the clean-up progressed).
- Research and development – When the opportunity and funding existed to extract invaluable information from this reactor vessel “full-scale” accident test case.
- Ensure safety – A large effort was also spent to measure parameters of various types to ensure that safe conditions existed; e.g. that no radiation was escaping to the environment or that no potential for a recriticality existed.

The balance between the three was sometimes difficult to strike because no one could be certain if the data gained from a particular characterisation task would provide information immediately useful to the TMI-2 project, useful in the long run to the nuclear power industry, or of no real use at all.

Reactor vessel

This work was the most fundamental and important part of all clean-up work from the accident onward. It began with analyses of computer codes and accident scenarios and continued with water sample analysis, video examinations, radiation and instrumentation readings, gamma scanning of an incore detector, debris sampling, topographic mapping by sonar, core stratification drilling, and removal of samples of the reactor vessel itself.

Details on the physical configuration of the post-accident core are available from numerous sources and are not repeated here. Some observations on the data are provided.

Samples of core debris particles from the upper head region indicated a significant depletion (up to 50%) of the zirconium content occurred and that less than 10% of the silver from the control rods was present. Ceramographic examinations showed extensive oxidation of fuel and cladding, molten oxygen-saturated alpha phase Zircaloy ($T > 2\,250\text{ K}$), molten $\text{UO}_2\text{-ZrO}_2$ ceramic ($T > 2\,800\text{ K}$), molten UO_2 ($T > 3100\text{ K}$) and relatively unaffected fuel ($T < 1\,900\text{ K}$). Debris samples were quite heterogeneous on a microscale but were fairly uniform from sample to sample in terms of fuel structure, elemental composition and uranium enrichment. The core debris retained approximately 94% of

the strontium, 53% of the antimony, 61% of the ruthenium inventories and practically all of the cerium. There was also evidence that molten steel tends to concentrate available ruthenium into a separate metal phase.

Debris samples from the lower head showed significant discrepancies between the concentrations of Ru-106 and Sb-125 in the two regions (upper debris bed and lower head). However, concentrations of Sr-90, Ce-144 and Eu-154 were more or less uniform in the two regions. In addition the lower head debris was substantially composed of a combination of the 1.98% and 2.64% enriched assemblies and that the 2.96% enriched assemblies did not participate in the melt that reached the lower head of the reactor vessel.

The TMI-2 Programmatic Environmental Impact Statement (NUREG-0683) suggested that pyrophoric materials might be present in the core debris and could be a safety concern during defueling. The metal of principal interest with respect to pyrophoricity at TMI-2 was Zircaloy-4, an alloy whose major constituent is zirconium (98 wt%). Following the accident it was judged that the principal compounds of zirconium associated with the core debris are most likely ZrO_2 and ZrH_2 and the formation of Zr-U solid solutions from Zircaloy – UO_2 fuel eutectics. As zirconium oxide, ZrO_2 is in its maximum oxidation state it is incapable of further oxidation. Based on a literature review it was determined that although zirconium hydride, ZrH_2 , can thermodynamically undergo oxidation it is less prone to ignition in air than zirconium for similar mesh sizes. Finally no data existed on the pyrophoric potential of uranium/zirconium eutectics. As a result of the literature review pilot ignition tests were performed to demonstrate the pilot ignition characteristics of TMI-2 core debris. All of the tests demonstrated no observed pilot ignition.

Reactor building

- Characterisation from outside the reactor building

Installed radiation monitors, water sample lines and existing penetrations were used to gather data in the building before the first entry. Samples allowed fairly accurate analyses of the water in the basement and the reactor coolant system, and a video camera inserted through a wall penetration provided a limited picture of the dark and dripping wet interior of the building. However, radiation levels were estimated to be several times higher than they actually were because in-plant radiation monitors gave false or misleading information, and the wall penetration used did not provide enough range to thoroughly survey the building.

- Containment Entry Programme

The entry programme was the first effort to characterise conditions in the reactor building by personnel inside the building. The first entry took place in July 1980 followed by entries of approximately one per month until November 1981, when preparations began for the Gross Decontamination Experiment, at which point the entry rate greatly increased. All entries were extensively planned for and had specific area or equipment characterisation goals.

- Gross Decontamination Experiment

This series of experiments was conducted in the containment in early 1982, and required months of preparation and pre-decontamination characterisation. The results showed that various decontamination techniques could be effective, but that recontamination would be a strong factor that could counteract much of the success.

Following the Gross Decontamination Experiment, the work of characterising the reactor building continued for many years. Numerous techniques were used, some familiar and some innovative. Uses of then existing technology included:

- Data acquired by using remote or remotely-transported survey instruments to obtain information in areas that were not accessible for human entry or where entry would have been non-ALARA.
- Directional survey data were gathered to differentiate between multiple radiation sources at each level of the containment and determine the contamination levels on walls and floors.
- Thermo-luminescent detector (TLD) strings were used to obtain vertical radiation profiles and were of particular value in obtaining data in inaccessible areas.
- Single-point TLD data were used for general characterisation and measurement of the effectiveness of decontamination for general area monitoring.
- Gamma spectral data were obtained to characterise the surface contaminants on the walls and floors.
- Self-reading dosimeter data was used to monitor and control personnel exposure during the clean-up effort.

Ex-vessel fuel

Fuel failure during the TMI-2 accident released fissile materials to the reactor coolant system (RCS) and to systems supporting the RCS. The most likely locations for reactor fuel in the nuclear auxiliary systems and the auxiliary and fuel handling building (AFHB) was determined by examining their use during the accident, early stabilisation and later clean-up phases. By examining these systems the most likely locations for reactor fuel concentrations were identified, i.e. tanks, filters, demineralisers and piping dead legs.

To ensure that the reactor fuel was being located and quantified correctly and in a timely manner, gamma spectroscopy was selected as the method to determine the presence of reactor fuel. Gamma spectroscopy was used to measure the quantity of Ce-144 and/or Eu-154 present in discrete locations. The quantity of cerium or europium present was converted to the quantity of residual fuel present based upon the calculated ratios and actual measurements of cerium/fuel and europium/fuel ratios.

In addition to quantifying ex-vessel, preventing the potential for a recriticality during decontamination operations was required. Therefore a programme was established to ensure that where a sufficient amount of fuel existed, system decontamination and subsequent material handling activities could be conducted without creating the potential for a criticality. The primary control parameter was that no more than 75% of a critical mass would be allowed to collect in any component. In addition any decontamination operations would be performed with borated water whose concentration had been determined to provide sufficient shutdown margin for TMI-2 fuel debris under all conceivable conditions of mass and geometry.

Waste

With the implementation of 10 CFR Part 61 “Licensing Requirements for Land Disposal of Radioactive Waste”, efforts were started to identify difficult-to-measure radionuclides. In late 1986 EPRI’s TMI-2 Technical Support Program conducted a detailed review of all TMI-2 liquid waste stream radionuclide analysis results with the objective to compile a comprehensive data file of all such results which contain measurements of the difficult-to-measure radionuclides requiring reporting by 10 CFR Part 61.

The objective of the investigation was to determine scaling factors for Tc-99, Ni-63, I-129, C-14 and transuranics. A single ratio of the nuclide to be inferred, to a tracer nuclide was desired. The tracer nuclide required the following attributes, with respect to the nuclide to be inferred:

- is readily measured using analytical techniques at TMI-2;
- is prevalent in TMI-2 waste streams;
- has comparable transport and removal behaviour.

The existence of a relationship between a specific nuclide and a preferred scaling isotope was evaluated by performing a linear regression analysis of the logarithm of the decay-corrected sample results.

For Tc-99, Co-60 was evaluated as the preferred scaling nuclide.

For Ni-63, Co-60 was again evaluated as the preferred scaling nuclide.

For I-129, Sb-125 was evaluated as the preferred scaling nuclide.

For C¹⁴ none of nuclides commonly measured at TMI-2 were found to correlate with C¹⁴. Consequently rather than developing a linear relationship to scale C-14 to another nuclide, an upper limit of observed C-14 in a given volume was specified.

For transuranics, sample results only reported low concentrations based on the very low solubility of these isotopes and that any particles containing these elements are very heavy and would tend to settle quickly. As a result the only realistic way to approach 10 CFR 61 limits is where actual particles of solid fuel are in the waste which is unlikely in any liquid waste stream.

When converting liquid waste concentrations (uCi/ml) to disposable solid waste concentrations (uCi/cc) due to processing, a volume reduction factor is used as follows:

$$\text{uCi/cc} = \text{uCi/ml} \times \text{Volume Reduction Ratio.}$$
 The maximum concentration thus results from the greatest volume reduction. Evaporation is one of the most efficient volume reduction processes; a value of 100 is considered to be excellent.

Conclusions and recommendations

Although TMI-2 post-accident characterisation preceded the EPA's DQOs process which was discussed in Section 5.2, a formal planning process similar to the EPA's DQOs process was used. TMI-2 characterisation focused the level of effort and purpose of the characterisation programme to primarily support the clean-up process but in addition provided valuable information for use of the industry as a whole.

The DQO process is a strategic planning approach based on the scientific method used to plan a characterisation activity. The process provides a systematic procedure for defining the objectives of a characterisation programme. The characterisation programme objectives developed via this process are qualitative and quantitative statements that:

- clarify the characterisation objective;
- define the most appropriate type of characterisation measurements to make, the number of measurements to make, and the most appropriate measurement methods to use;
- determine the most appropriate locations and times to make characterisation measurements;
- specify the level of measurement uncertainty that is allowable to support the decisions that are made based on the characterisation measurements.

Following this planning process ensures that the type, quantity, and quality of characterisation data used in decision making will be appropriate. In addition, using this process guards against committing resources to the characterisation effort not required to support a defensible decision.

The planning process consists of seven steps:

- state the problem;
- identify the decision;
- identify inputs to the decision;
- define the study boundaries;
- develop a decision rule;
- specify limits on decision errors;
- optimise the design for obtaining data.

The output from each step in the planning process influences the choices that are made later in the process and often leads to reconsideration and refinement of the outcome of previous steps. This iteration is desirable since it ultimately leads to a more resource-efficient (i.e. time, money and dose) characterisation programme.

During the first six steps of the process, the planning team develops the objectives that will be used to build the characterisation programme in the last step. The first six steps should be completed before the planning team attempts to develop the data collection programme design because this final step requires a clear understanding of the objectives identified in first six steps.

Documenting the characterisation programme in a plan that also incorporates all the elements of an industry-recognised quality programme such as ISO 9000 or ASME NQA-1 ensures that the data collected are defensible and of known quality.

Lessons learnt from the TMI-2 accident

- Different accident pathways have different characteristics particularly between volatile and non-volatile constituents.
- Ex-vessel fuel measurements were performed using surrogates and scaling factors based on isotopes which would react similarly to fuel, such as cerium and europium.
- After the TMI-2 accident, engineers were faced with the difficult task of determining the type and quantity of accident data needed to establish plant conditions amid pressures to acquire extensive data for the community and to avoid delaying clean-up activities. Application of a formal planning process for the acquisition of data proved valuable, and focused the level of effort and purpose of the characterisation programme to primarily support the clean-up process. This process is very similar to but preceded the MARSSIM, NUREG-1575 which is an excellent reference for designing and performing characterisation.

Experiences on Chernobyl radioactive waste management

Sorting of radioactive waste in Ukraine taking into account existing classifications and disposability

The main problem in Ukraine has been that the types and categories of radioactive waste used thus far do not take into consideration the basic disposal requirements. The Ukrainian “Law on the Radioactive Waste Management” and “Basic Sanitary Rules of Radiation Safety in Ukraine-2005” mention short-lived and long-lived waste as separate

types; moreover, short-lived waste can be disposed of in surface repositories while long-lived waste should be disposed of in deep geological repositories.

The practical question for the waste producer is “How to define which waste is ‘short-lived’ and can be disposed of in surface repositories?” National legislation provides only a very generic answer to this question.

To answer the question and implement the classification practically, the waste producer should know the waste acceptance criteria (WAC) of the repository in which the conditioned waste will be disposed of. Of course, the characterisation, sorting, treatment and conditioning of waste should be performed taking WAC into account.

Today in Ukraine there is an Engineered Near-surface Disposal Facility (ENSDF) for low- and intermediate-level (short-lived) solid radioactive waste licensed for disposal of conditioned waste from ChNPP site. Another near-surface disposal facility (trench-type) – “Buriakivka” was created and licensed only for disposal of waste of “Chernobyl origin”.

According to the applicable regulations for solid radioactive waste with unknown specific activity it is allowed to use sorting to low, medium and high activity, depending on the gamma dose rate at a distance of 0.1 m from the surface. The use of this criterion for separation of waste streams for many years (in particular, on nuclear power plants) resulted in accumulation on NPPs of large amounts of waste, the nuclide composition of which can only be estimated with a high degree of uncertainty. At the same time to transfer this waste for disposal, according to the regulatory documents, the manufacturer must demonstrate compliance with WAC for disposal in a specific repository. The passport of conditioned waste to be disposed of must provide information about radionuclide composition (list of nuclides, their specific and total activity) for a list of radionuclides including long-lived beta- and alpha-emitting radionuclides. This also applies to determination of chemically active substances in accordance with the limits established by the operator of the disposal facility.

As a result, the modern practice of waste characterisation and sorting is not sufficient enough for ensuring proper disposal. In many cases, the waste characterisation is limited to measuring the total (β/γ) activity and/or the corresponding rates of radiation dose from the waste packages. Thus, additional work on the measurement and sorting will be needed after the introduction of a new classification scheme.

It is preferable to characterise and separate waste immediately, at the site of waste generation. Waste types from this waste stream are usually similar in structure and radionuclide content, so this simplifies and reduces the amount of work for sampling and measurement. In addition, it is possible to establish a correlation between easily measured radionuclides (Cs-137 and Co-60) that a particular waste stream contains, and other radionuclides, which are much harder to measure. Accordingly, having the activity value of these easily detectable nuclides allows the concentration of other radionuclides present in the waste to be estimated fairly accurately.

Conducting characterisation and sorting later, after a joint temporary storage of mixed waste of different origin, is very complex and expensive, and can lead to significant uncertainty. Some difficult-to-measure radionuclides need comprehensive sampling and measurement techniques including destructive radiochemical analyses. Measurement of the characteristics necessary for identifying a suitable disposal option becomes almost impossible after waste conditioning. For example, the specific content of α -emitting radionuclides is a very important parameter in determining the admissibility of surface disposal. Thus, this information must be obtained by sampling and measurements from the “raw” waste prior to processing. In case it is not possible or economically impractical, the mixed waste should be disposed of as one intermediate-level waste (ILW) unit in a geologic repository at intermediate depths.

Current regulatory documents of Ukraine do not have “general waste acceptance criteria” for the disposal of radioactive waste in disposal facilities of different types,

expressed in terms of specific activities of individual radionuclides in the waste. Requirements, established by NRSU-97/D-2000 and BSRU-2005 for classification of waste as “short-lived” and “long-lived”, based on criteria of admissibility/inadmissibility for disposal of waste in surface or geological type storage facilities, respectively, are based on dose criteria of radiation exposure for people. For assignment of certain radioactive waste to a certain type of facility, estimates of radiation exposure for the population 300 years after disposal should be carried out to compare them with the relevant regulations (reference probability of critical events, criteria limit of radiation dose). To fulfil these estimates for each individual case there should be data not only on the actual radionuclide composition, amount and specific activity of radioactive waste, but also on geographic and geological conditions of the site of the future disposal facility. This is a difficult task for the “waste producer”.

Thus, the introduction of appropriate waste characterisation and sorting will be more helpful in reaching the complete waste “disposal” than the introduction of a new waste classification system. Applying the new waste classification system requirements while characterising and sorting waste, the main existing problem regarding the future of waste disposal can be successfully solved using the significant economic benefit of the new waste classification system.

A prerequisite for applying appropriate measures for waste characterisation and sorting is to develop “general waste acceptance criteria” for each waste class. Based on these “general waste acceptance criteria”, the requirements for waste sorting and characterisation can be designed to provide relevant recycling and to receive data and information necessary for verifying compliance with the waste acceptance criteria.

Lessons learnt from Chernobyl experiences

- the present classification of radioactive waste (RW) by type of disposal facility (“near-surface” or “geological”) is not practically used by the RW Producers in Ukraine;
- the existing practice of sorting and characterisation of RW applied by the “RW producers” does not comply with the requirements for the final disposal of RW and should be changed;
- it is necessary to improve as soon as possible the waste characterisation and sorting at the sites of waste;
- waste characterisation and sorting shall be based on the updated RW classification system in Ukraine;
- for every class of RW the “general WAC” should be developed and approved in terms of the limits of specific activities of individual radionuclides that are acceptable for disposal in a certain type of disposal facility;
- it is necessary to introduce methods for determining characteristics of RW generated (including their nuclide composition and specific activity of individual radionuclides) and separating waste streams to meet the requirements for different types of disposal;
- to some extent, it may be necessary to improve the current practice of sampling and measurement for α -emitting radionuclides in particular, since not enough attention was paid to this in the past on the waste generating sites.

First experience of waste classification, characterisation and sorting taking into account disposability of RW in near-surface disposal facility

A facility has been developed for characterising and sorting solid radioactive waste (“SSR-cell”) as part of Plant for Sorting SRW of all categories and treatment of low- and

intermediate-level short-lived solid waste (solid waste processing facility – SWPF). The Industrial Complex for Solid RW Management (ICSRM) includes such facilities as:

- Retrieval facility for solid waste (RFSW) of all categories from the existing solid RW storage facility (SWSF) and their loading into skips, which are subsequently put into transport containers for sending to the sorting facility.
- Plant for sorting solid radioactive waste (SRW) of all categories and treatment of low- and intermediate-level short-lived solid waste (SWPF) that includes:
 - a facility for characterisation and sorting of SRW;
 - facilities for processing low- and intermediate-level short-lived waste (LILW-SL) – size reduction, incineration (combustible SRW and liquid radioactive waste [LRW]), compacting and conditioning for further disposal.
- ENSDF for low- and intermediate-level (short-lived) solid radioactive waste.

The RFSW and SWPF are situated on the ChNPP site territory, while the ENSDF is in the 30-km radius exclusion zone on the vector research and industrial complex territory.

Waste classification used at the SWPF

- Waste classification by the acceptance criteria for their disposal in a near-surface repository

In accordance with Ukrainian radiation safety requirements, the waste accepted for disposal in near-surface repositories for which the conditions established for total or restricted exemption from the institutional control are met in 300 years after repository closure. According to waste classification by the acceptance criteria for their disposal in a near-surface repository, they are subdivided into short-lived (SL) and long-lived (LL) waste. This classification is used at the SWPF while sorting waste to ensure conformance of the final product with the acceptance criteria used at the ENSDF.

- Waste classification by specific activity criterion

Waste classification based on specific activity criterion defines the following three waste categories: LLW, ILW and high-level waste (HLW). This classification is used while sorting waste for subsequent treatment at the RFSW installations and ensuring radiation safety during treatment.

Taking into consideration the waste classifications described above, the waste to be treated at the RFSW is subdivided into the following:

- LILW-SL – those attributed, by specific activity criterion, to low/intermediate-level ones, and according to the permissibility criterion are approved for final disposal in a near-surface repository (ENSDF) constructed especially for conditioned RW from the ChNPP site.
- Low- and intermediate-level long-lived waste (LILW-LL) and HLW – those that are attributed to high level ones by specific activity criterion, or are not allowed for disposal in a near-surface repository and should be stored in the interim storage for LIL-LLW and HLW located on the ChNPP site.

Facility for characterisation and sorting of SRW – SSR-cell

The SSR-cell (sorting and size reduction facility) is designed for sorting and preparation of incoming waste for subsequent processing at the ICSR facilities (i.e. the room with stainless steel walls). The cell contains measurement equipment, and devices for RW fragmentation are installed.

Figure 5.1. Emptying of skip into the receipt bay inside the SSR cell



Source: Tokarevskyi, 2016.

According to the design, solid waste is supplied from the receipt bay (Figure 5.1) into the shielded SSR-cell where they undergo radiological characterisation with the use of gamma camera and passive neutron detectors. This allows the waste to be subdivided into the following categories:

- LILW-SL;
- LILW-LL;
- HLW.

On completion of the necessary measurement, HLW are first removed from the whole mixture. Upon detection of waste where surface gamma dose rate exceeds 10 mSv/h (the lower limit for HLW), such waste is classified as HLW. HLW detected while sorting is placed into a 165-litre drum docked to the SSR-cell sorting position. Drum filling is controlled visually while dose rate control in the loading position area is envisaged to exclude exceeding a permissible HLW activity in the drum.

After retrieval of HLW, LILW-LL is separated. Detected LILW-LL is placed into a 165-litre drum docked to the next sorting position.

The rest of waste (supposedly, LILW-SL) is visually subdivided into the following categories:

- combustible waste;
- non-combustible compactable waste;
- non-combustible non-compactable waste.

Bulky waste which cannot be fragmented is placed in the disposal container installed into the sorting cell docking position. The rest of the waste is put into the relevant 165-litre drums.

Waste in 165-litre drums and disposal containers are delivered to the relevant process areas by the on-site transport system.

For radiological monitoring of the waste, there are installed in the SSR-cell:

- dose rate meter;
- gamma camera;
- detectors for neutron flux measurement;
- HLW dose meter.

The gamma camera is suspended from a crane hook, while detectors are located at the platform below the table for sorting. The dose rate meter is located in front of the container of non-compactable waste while the HLW dose meter is situated near the position of waste loading into drums. HLW dose meter is intended for filling drums by dose rate.

Data processing computers with displays are located in Room II/128. The crane and remotely operated vehicle operators control visually the readings of the gamma camera and the neutron flux measuring detectors for entering into the tracking system database.

The gamma camera NUKEM system includes a measuring head, a dose rate meter, a gamma-spectrometer, scanning unit, on-board computer, main computer and connecting cables.

The system for formation of a measuring head picture is based on the pinhole camera principle. Two conical collimators produce an inverted roentgen or gamma-image of an item on the scintillating plate. An image amplifier is used to amplify the light stream and for decreasing the image size. The matrix of the charge-coupled device is designed to read the optical image. The lead shield minimises the background gamma radiation that reaches the scintillator through the shielding, i.e. gamma rays which do not come from the viewed objects. The device is supplied with a control system.

The video camera enables recording images of an observed item. The video images may be used to identify the gamma-source; this is performed by superimposing the gamma and video images.

The equipment, including the computer, is inside the cell. The computer is used to control the device, both for the preliminary processing and for the compression of gamma images. The main computer, connected to the computer in SSR-cell, is used for reading and saving, displaying it on the monitor and for a detailed processing of measurement results. This image is used by the operator to identify the items with maximum activity.

For the conventional visual representation on the display, the compressed images are interpolated and displayed in pseudo colours. The superimposition of the video and gamma images is also produced in pseudo colour and the relative intensities of the optical and gamma images are adjustable. These opportunities permit complete identification of the location of items with a higher activity. The image compression does not impair the angular resolution, but minimises the time of image transmission to the main computer and may be used for storage of images on a hard disk. Compression of the image increases the signal-to-noise ratio, as the fluctuation noise of the charge-coupled device of the camera rises due to the square law, while the level of the useful signal rises proportionally to the total number of pixels. Therefore, the sensitivity of the device rises, as well.

The supply set includes appropriate software for all the system components that permits it to transmit, fix, process, save and represent optical and gamma images.

The radiation rate is measured by Geiger-Muller counters. The optimal range for measuring the radiation rate will be selected automatically. The measurements range of the radiation rate counters is (100 nSv/h to 100 mSv/h) $\pm 5\%$.

The gamma-spectrum is measured with a detector equipped with a collimator system. To measure the gamma-spectrum the main gamma emitters may be differentiated.

- Detectors for measuring the neutron flux

When measuring the neutron flux, the intensity of the neutron emission is used to calculate the content of actinide elements. Neutrons are detected by means of a He-3-detector. Due to a high gamma-background special devices are used to suppress

gamma radiation. Based on the results of these measurements, the α -activity level is determined. The waste isotope composition is used to count the neutrons according to the α -activity.

▪ Dose meter for non-compactable waste

The exposure dose rate is measured by means of Geiger-Muller counters. The optimal range for measuring the dose rate is selected automatically. The measurements range of the recommended detectors of dose rate is (100 nSv/h to 100 mSv/h) $\pm 5\%$.

Lessons learnt from stages I and II of commissioning of the SSR-cell

- The system should provide characterisation, categorisation and fragmentation of solid RW remotely, with no manual operations.
- There are problems with accuracy and adequacy of measurements provided by systems of radiological control in SSR-cell: how to be sure that all systems of radiological control are working properly with non-characterised waste?
- The ChNPP approach: to divide the commissioning phase of the Solid Waste Treatment Plant (SWTP) into three stages, which involve step-by-step checking of the quality of measurements using different kinds of RW:
 - I stage – using RW with known characteristics in packages.
 - II stage – using RW with known characteristics without packages (RW is measured previously using the In Situ Object Counting System ([ISOCS]).
 - III stage – work with “real” RW to be retrieved from solid RW storage facility (an existing waste storage facility).
- Commissioning of the SSR-cell was longer and more complicated than expected because of difficulties with some equipment (problem with NMS-passive neutron detectors).
- Updating of software is needed.
- There is no drying system for waste in SSR-cell: this may cause problems related to RW that will be retrieved from solid RW storage facility:
 - dealing with wet RW (at the same time there is the requirement established in approved design: “no visible water in RW”);
 - absence of system for drying of RW inside the SSR-cell;
 - practical difficulties with segregation, sorting and characterisation of RW conglomerates and RW present as a “rotting mass” – time delays are possible compared with characterising “normal” RW;
 - some additional technical decision(s) will be necessary depending on the situation with RW.

Principles of SRW sorting as regards radiation characteristics

In the course of SRW sorting at the SWPF the following parameters are measured: γ -Dose rate, intrinsic γ -spectrum of radionuclides, neutron flux, waste mass.

The superposition of video and γ -pictures of sorting RW is performed in the SSR-cell in order to simplify the operator’s actions during sorting, and to accelerate the process itself. The following SRW features required for characterisation and sorting according to the above-mentioned categories are defined by calculations based on the measurements:

- dose rate (mSv/h at 10 cm distance to the SRW surface);
- specific beta activity (Bq/kg);

- specific alpha activity (Bq/kg);
- SRW radionuclide inventory.

In the SRW characterisation, the dose rate measurement is preferred since it allows a high throughput and it is easy to handle.

For determination of specific α -activity the given relation between alpha-emitting nuclides and Cs-137 activities is used where one uses Cs-137 as a key nuclide (marker). At the design stage, the ratio between Cs-137 and α -activity was assumed to be 25. However, the maximum case of mixture depletion when this ratio may be equal to 1 is also addressed for the sorting. When calculated activities of alpha-emitting radionuclides evaluated from Cs-137 activities are close to threshold levels (i.e. to limit values used for activity differentiation on groups), the SRW neutron flux is measured to determine actinide concentrations independently. This is also done in the situation when Cs-137 spectrometric assessment is either not possible or performed with low accuracy because of a high Co-60 contribution.

The neutrons in SRW are generated under spontaneous fission of actinide elements. The probability of spontaneous fission is less than alpha decay probability by several orders of magnitude. The assessments performed for the radionuclide inventory of ChNPP unit 4 fuel demonstrate that neutron flux of 1 kg of RW with α -activity 370 Bq/kg (the boundary between long lived waste and short lived waste) is 0.04 neutron/s. The quantity of neutrons generated as a result of (α ,n) reaction mainly on O-17, O-18 isotopes nuclei do not exceed 10% of the quantity of spontaneous neutrons.

The “nuclide vector” or radioactive fingerprint of SRW is required if the key nuclide method is to be used for radionuclide inventory calculations. At the design stage, a radionuclide fingerprint based on the literature data was developed for the following radionuclides: H-3, C-14, Kr-85, Sr-89, Sr-90, Zr-95, Nb-95, Mo-99, Ru-103, Ru-106, Ag-110, Sb-125, Te-132, I-129, I-131, I-132, Xe-133, Cs-134, Cs-136, Cs-137, Cs-138, Ba-140, La-140, Ce-141, Ce-144, Eu-154, Np-239, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Cm-242, Cm-244.

It was expected that the radionuclide fingerprint should be defined more exactly during the SWPF pre-commissioning stage. In future, during the operational phase it should be periodically verified and corrected as necessary. For reliable sorting, the set of nuclide vectors for different waste types is required. The information related to nuclide vectors for different waste types can be obtained by radiochemical analysis of SRW samples in the ChNPP laboratory and it will decrease the level of conservatism during nuclide inventory determination of SRW.

Conclusions and lessons learnt

- Chernobyl NPP’s approach to characterise waste stored in the old storage facility (SWSF) is to use Cs-137 as a key nuclide (marker) to establish concentrations of other radionuclides. The radionuclide fingerprint established during the design of facility should be defined more accurately during the facility pre-commissioning stage; in future, during the commissioning and operational phase it should be periodically verified and corrected as necessary.
- The real characteristics of RW in the old storage facility (SWSF) are unknown. ChNPP specialists are considering additional characterisation for every portion of SRW retrieved from the SWSF:
 - to provide sampling and laboratory measurement;
 - to correct/clarify the radionuclide fingerprint for every portion of retrieved RW;
 - to provide the final measurement to allow sorting of retrieved RW in the SSR-cell.

Lessons learnt from the ChNPP accident (summary)

Lessons learnt from the Chernobyl accident fall into two broad categories:

- lessons learnt when trying to use waste characterisation data to sentence waste for disposal in near-surface disposal facilities;
- lessons learnt when commissioning waste sorting facilities.

In the first area, it is found that the existing practices of sorting and characterising waste do not comply with the regulatory requirements for the final disposal of radioactive waste, and should therefore be changed in order to be compliant with the legislation. Improved waste characterisation and sorting at producer sites and waste storage sites is required. It is preferable to characterise and separate waste immediately, at the site of waste generation. In particular, better characterisation of alpha-emitting radionuclides is required, since not enough attention was paid to this in the past on the waste generating sites. In parallel, general waste acceptance criteria should be developed and approved for each type of radioactive waste; this will provide certainty on the limits of specific activities of individual radionuclides that are acceptable for disposal in a certain type of disposal facility.

In the second area, the experience has been that commissioning of the automated radiological measurement and sorting systems has been longer and more complicated than expected. There have been problems with ensuring appropriate calibration of the measurement systems in the SSR-cell. There have been difficulties with the operation of some equipment (NMS-passive neutron detectors).

The approach of Chernobyl NPP to the characterisation of non-characterised waste stored in the old storage facility (SWSF) is to use Cs-137 as a key nuclide (marker) for determination of other radionuclide activities, including alpha-emitting nuclides. The radionuclide fingerprint established during the design phase of the facility will be defined more accurately during the facility pre-commissioning stage; in future, during the commissioning and operational phases the fingerprint should be periodically verified and corrected as necessary.

Fukushima Daiichi accident

Introduction

In order to present a safe processing and disposal concept for 1F waste, it is necessary to develop a strategy for waste management. The strategy refers to the procedure and/or methodology for waste management and it could be developed under the investigation interrelated with the following important waste management items:

- waste characterisation;
- waste classification/categorisation (described in Chapter 6);
- waste conditioning and waste volume reduction (described in Chapter 7);
- waste destination (storage/disposal) (described in Chapter 8).

In Japan, this kind of procedure and/or methodology for radioactive waste generated from a severe accident (such as 1F) has not yet been developed yet. However, a project for the development of a strategy for the waste management of 1F has been just launched.

This project is called “Examination of Waste Stream”. The project aims at developing the procedure and/or methodology for handling of the waste, by which safe and rational processing and disposal can be ensured in each process, such as pre-treatment, conditioning, storage and disposal from waste generation through to the waste disposal.

Outcomes will therefore be provided for this project along with the progress of the waste management items described in Chapter 6, 7, 8 and 9.

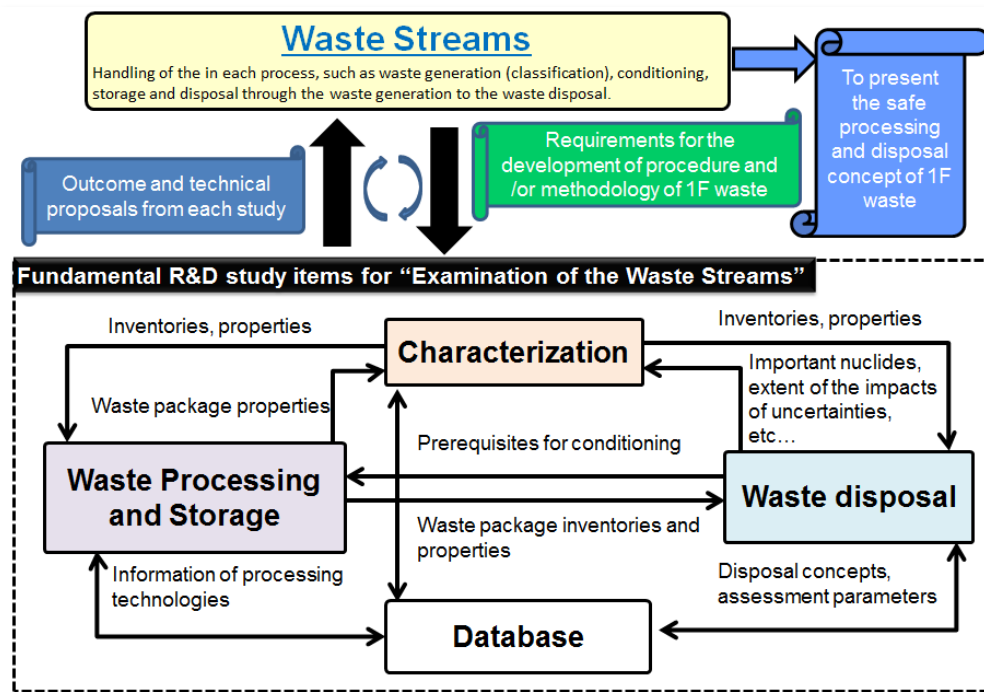
Status of Examination of Waste Stream

In “Examination of Waste Stream”, the following investigation and discussion is conducted:

- Studies on each process, such as characterisation, waste processing and storage, as well as waste disposal are being conducted as fundamental R&D study items.
- For these fundamental R&D items, outcomes will likely improve through repeated investigation using the feedback from additional information obtained about the items.
- The outcomes such as data, information and knowledge obtained from these studies are aggregated.
- Based on these outcomes, a rational procedure and/or methodology for the realisation of safe processing and disposal of 1F waste will be comprehensively examined and discussed. Furthermore, if needed, the issues to be resolved for the development of rational procedure and/or methodology are thrown into characterisation, waste processing and storage and waste disposal studies.
- New outcomes and technical proposals from each study will be recovered and comprehensively re-examined to develop the rational procedure and/or methodology for waste management.
- This information exchange will be repeated, if needed.

The flow of “Examination of Waste stream” is shown in Figure 5.2.

Figure 5.2. Flow of “Examination of Waste Stream”



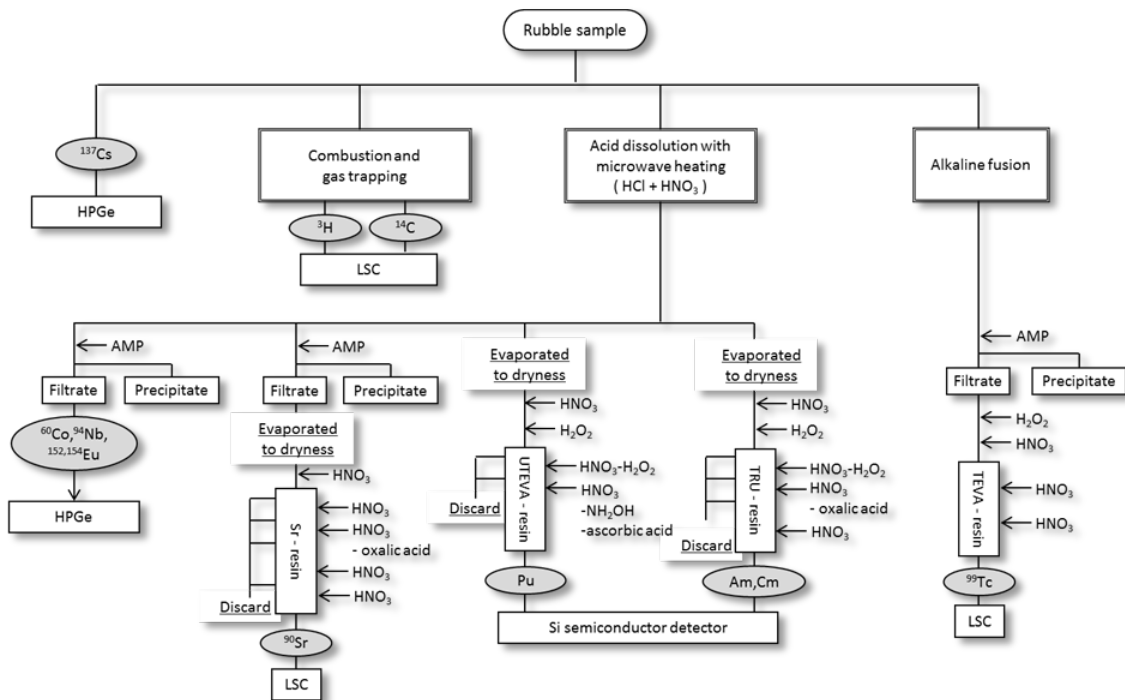
Source: Miyamoto, 2016.

Radiological analysis

In order to obtain the nuclide composition which is characteristic of the contaminated materials in the power station site, soil, vegetation, contaminated water and secondary waste from the treatment of the contaminated water were sampled and analysed. Twenty-eight nuclides among the important 38 nuclides were intended for analysis for safety assessment of waste disposal in Japan since 2012, and some of them are “difficult-to-measure” nuclides. The analysis was conducted based on the integrated method developed for laboratory waste which contains various nuclides (Kameo et al., 2009). The samples were transported from the site to the laboratories.

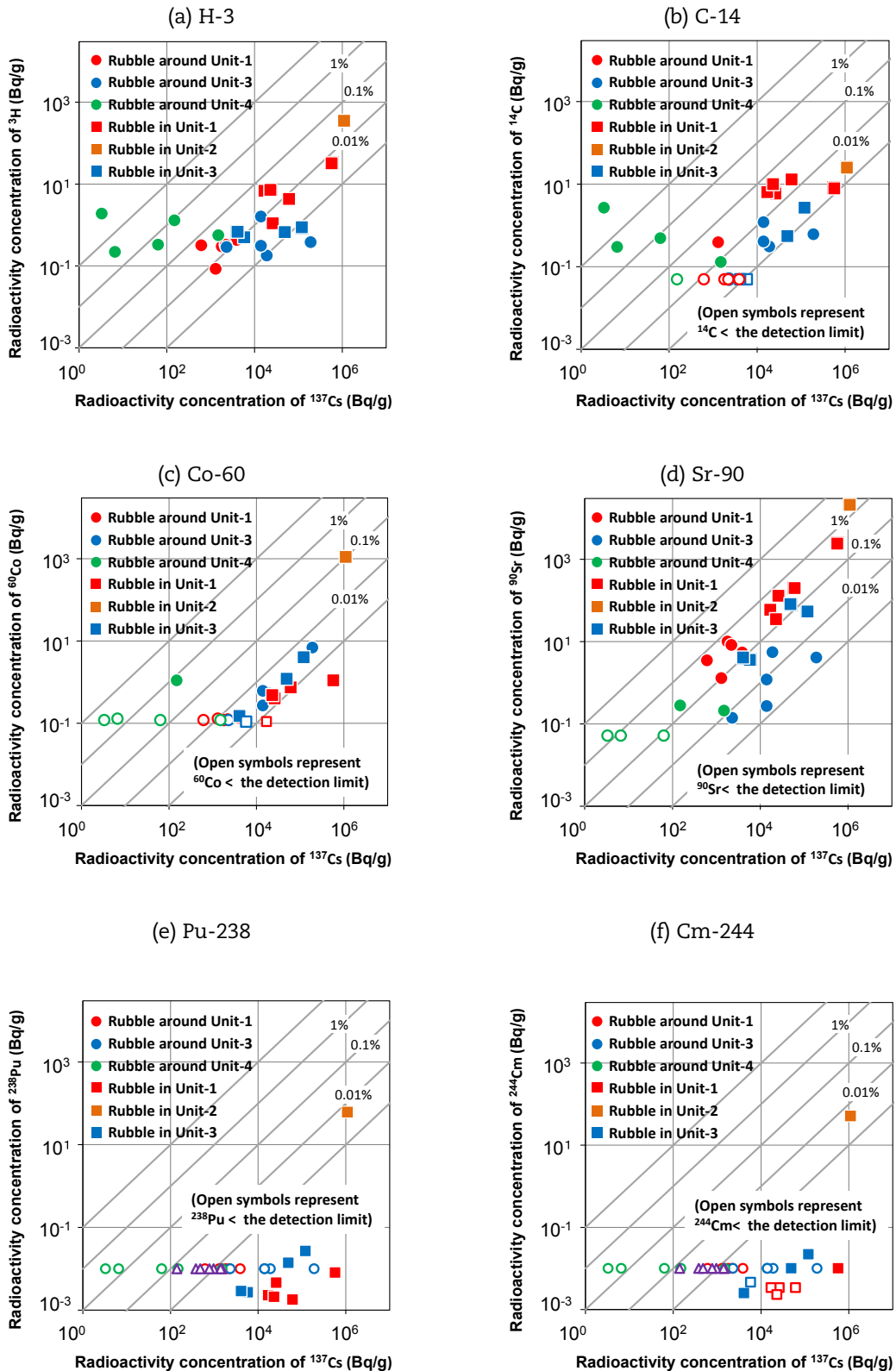
Rubble – Because of the hydrogen explosion of reactors, the amount of radioactive waste is greater than for an ordinary nuclear power plant. Large amounts of rubble, for example, containing various radionuclides have been generated. Analysing rubble is important, and the rubble scattered outside and inside the reactor buildings, as well as the coating paint of the floor were sampled and analysed. These analysed samples were summarised in Table 5.1. (Tanaka et al., 2014). The analytical procedure is shown in Figure 5.3. The concentrations of some detected nuclides plotted against Cs-137 are shown in Figure 5.4. H-3 was detected in all the samples and seems to correlate with [Cs-137] for units 1, 2 and 3; this is similar to ^{14}C detected in most of the samples. The concentration ratio $[\text{Co-60}]/[\text{Cs-137}]$ was high for units 2 and 4. Sr-90 correlated with Cs-137 and the order of $[\text{Sr-90}]/[\text{Cs-137}]$ was unit 2 > 1 > 3~4. For unit 2, the ratio of non-volatile nuclides of Co-60, Sr-90, Pu-238 and ^{244}Cm were greater compared with units 1 and 3. Contamination of unit 4 was less significant since the reactor had no fuel in the reactor core. As a result, the nuclide composition was different from the other reactor units.

Figure 5.3. Analytical flowsheet for rubble



Source: Tanaka et al., 2014.

Figure 5.4. Concentration of detected radionuclides (a) H-3, (b) C-14, (c) Co-60, (d) Sr-90, (e) Pu-238 and (f) Cm-244 as a function of those of Cs-137



Source: Tanaka et al., 2014.

Table 5.1. Samples of rubble (concrete, otherwise noted)

No.	Location of sampling	ID	Dose rate ($\mu\text{Sv/h}$)	Mass (g)	Remark
1	Unit 1 outside	1U-06	63.4	165.4	Including coating of light blue and beige
2		1U-07	2.4	131.2	Including coating of beige
3		1U-08	15.4	155.7	Including coating of grey
4		1U-09	16.4	92.6	Including coating of light blue
5	Unit 3 outside	3U-02	95.4	85.1	Including coating of light blue
6		3U-07	22.4	122.3	Including coating of beige
7		3U-09	1 000	115.6	Including coating of light blue
8		3U-10	113	142.6	Including coating of green and beige
9	Unit 4 outside	4U-01	2.4	40.0	Including coating of beige
10		4U-02	B.G.	152.9	Including coating of beige
11		4U-05	B.G.	177.4	
12		4U-08	B.G.	116.0	
13	Unit 1, 1 st floor	1RB-AS-R1	100	50.9	
14		1RB-AS-R3	74.5	50.0	
15		1RB-AS-R4	87	51.0	
16		1RB-AS-R6	93	26.0	
17		1RB-AS-R10	970	26.0	Insulating material
18	Unit 2, 5 th floor	2RB-DE-C2	73	5.0	Floor coating
19	Unit 3, 1 st floor	3RB-AS-R3	340	26.0	
20		3RB-AS-R4	17	26.0	
21		3RB-AS-R6	13	26.0	
22		3RB-AS-R8	91	26.0	

Source: Tanaka et al., 2014.

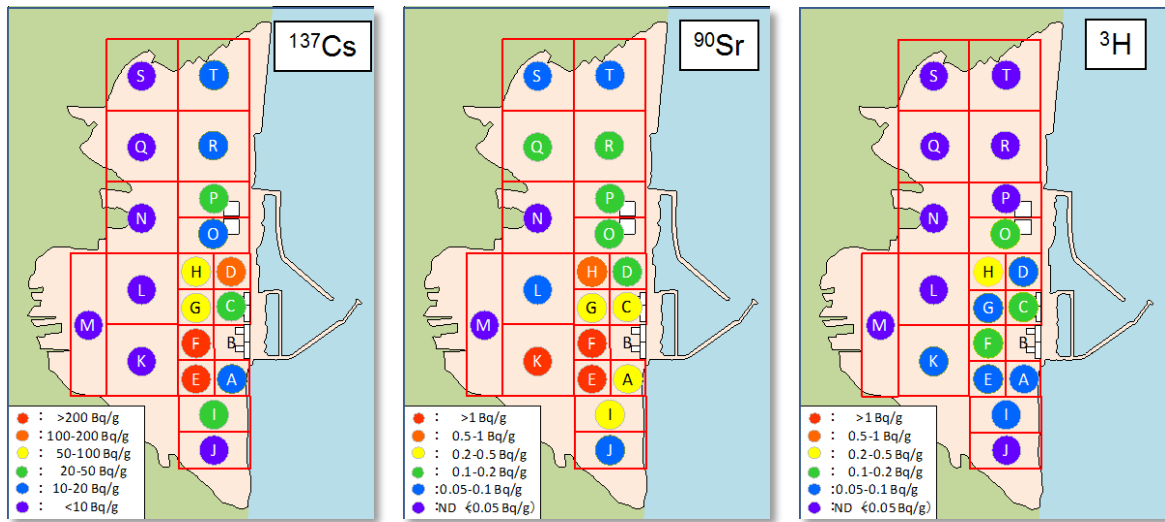
Soil – The soil at the three locations at the site were sampled and analysed over ten months from March 2011. Short-lived nuclides including Mo-99/Tc-99m, I-131, Te-132, Cs-136 and Ba-140/La-140 were detected. The concentration differed by location; it suggests dependency on direction from the reactor buildings. Pu resulting from the accident was identified from its isotope composition. The detected uranium was dominated by natural occurrence.

Vegetation – Concerning the contaminated vegetation, felled trees and living trees were analysed. The living trees were sampled for their leaf-branch, fallen leaves and topsoil (humus) in 2013. The concentration of some nuclides in leaf-branches was high for the locations near the reactor buildings as shown in Figure 5.5; H-3 and C-14 were only detected around them. It seems that Cs-137 and Sr-90 correlated for both felled and living trees. The concentration ratio of Cs-137 of leaf-branches to fallen leaves and topsoil is considerable indicating that Cs was transferred with time into the fallen leaves and soil.

Contaminated water – The nuclide composition of the contaminated water is useful for evaluating the inventory of secondary waste produced from water treatment operations and from contamination of material that has contacted the water. The contaminated water and the chemically treated water were found to contain H-3, Co-60,

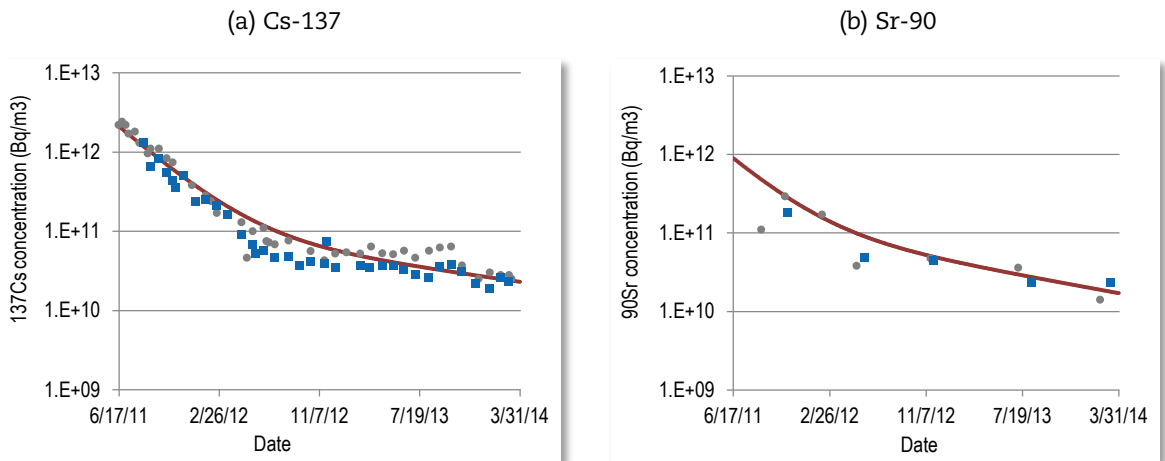
Ni-63, ⁷⁵Se-79, Sr-90, I-129, Cs-137 and isotopes of Pu, Am and Cm. The radioactivity concentrations were decreased by operation of the water treatment equipment. As shown in Figure 5.6, the rate of decrease of Cs-137 and Sr-90 was somewhat slower since the middle of 2012, and the tendency is similar for the other nuclides. Extrapolating the future trend of the concentration change is important in estimating inventory. Therefore, a transport model was developed to consider the effect of dilution of the contaminated water, which contains radionuclides released from damaged fuel and the continuous release from the damaged fuel into the cooling water (Shibata et al., 2016). The model is illustrated in Figure 5.7, and the fitted curves are shown in Figure 5.6 for Cs-137 and Sr-90.

Figure 5.5. Concentration distribution of some nuclides for leaf-branches sampled from living trees in 2013



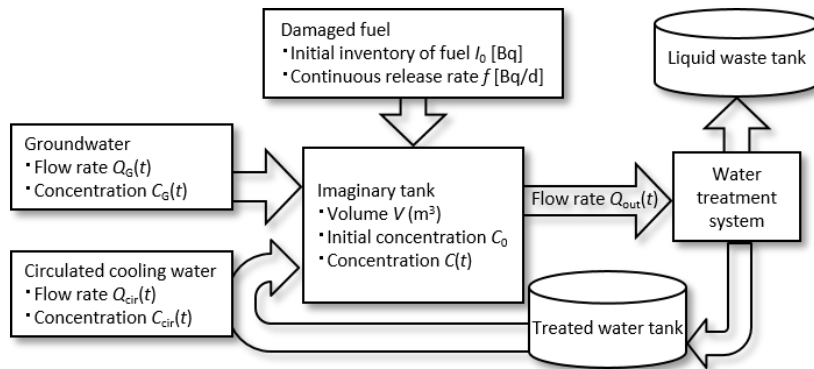
Source: IRID/JAEA, 2015.

Figure 5.6. Decrease of Cs-137 and Sr-90 concentrations in contaminated water



The circle and square denote the process main building and the high temperature incinerator building, respectively, as the sampling place of the water.

Source: Shibata et al., 2016.

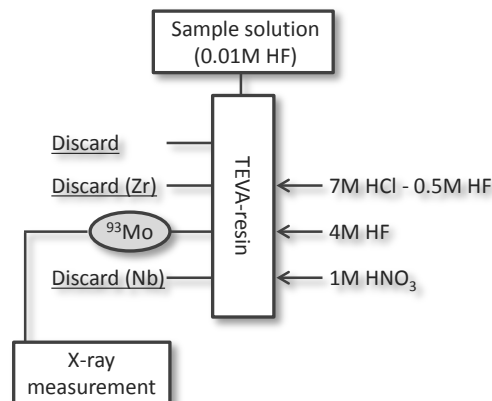
Figure 5.7. Calculation model for concentration change for contaminated water

Source: Shibata et al., 2016.

Secondary waste – The secondary waste generated from purification of the contaminated water contains radionuclides that are dependent on the chemical treatment process; thus, analysis is required. The slurries generated in the multi-nuclide removal equipment were sampled from the waste concentration and analysed. It was confirmed that the slurry of carbonate and iron hydroxide contains mainly Sr-90 as well as α emitters, other fission products and activation products. Sampling the waste for adsorbent and sludge is often difficult due to the structure of container and/or high dose rate. In that case, a practical sampling method will be investigated for analysis.

In the analysis described above, Cl-36, Ca-41, Ni-59, Nb-94, Eu-152, U isotopes, Np-237, Pu-241, Pu-242, Am-242m, Am-243, Cm-245 and Cm-246 were not detected. The target nuclides for analysis should be determined by considering the analytical results, the practical detection limit and the expected concentration in the waste.

Analytical methods for “difficult-to-measure” nuclides were developed along with analysing some waste. Generally, the analytical method is a combination of chemical separation and determination, and the methods for Mo-93, Zr-93, Pd-107 and Sn-126 were investigated. For example, an analysis method of Mo-93 was developed as shown in Figure 5.8; Mo is separated from Zr, Nb and other contaminants with solid extractant and is determined from its X-ray (Shimada et al., 2014). An analytical method will be developed according to requirements.

Figure 5.8. Analytical procedure for determination of Mo-93

Source: Shmada et al., 2016.

Characterisation of waste

Analysis to date has been limited to a small number of samples owing to the detailed analysis required, difficulty in sampling and to certain detection limits for “difficult-to-measure” nuclides. In order to get a good picture of contamination for the site waste, it is important to estimate nuclide concentrations for undetected nuclides and for “difficult-to-sample” waste. For these purposes, transport behaviour of nuclides in contaminated material was investigated by normalising the analytical concentrations with the content of the damaged fuel; the quantity is defined as the following equation and denoted as the transport ratio:

$$T_X = \frac{N_{X,\text{sample}}/N_{X,\text{fuel}}}{N_{\text{std},\text{sample}}/N_{\text{std},\text{fuel}}} = \frac{A_{X,\text{sample}}/A_{X,\text{fuel}}}{A_{\text{std},\text{sample}}/A_{\text{std},\text{fuel}}},$$

where N is number of atom, A is radioactivity, subscript X is the nuclide of interest, std is the standard (key) nuclide, sample is the material analysed, fuel is the damaged fuel. The transport ratio for isotopes should result in an identical value as far as the isotope effect is negligible. Cs-137 was selected for the standard. The ratio for Cs should be unity for the isotopes.

The contaminated water is generated in the course of cooling the damaged fuel (fuel debris) and the water is contaminated in the buildings of the reactor/turbine. The variation in the transport ratio to the contaminated water is shown in Figure 5.9 for the period of 2011-2012 (Koma et al., 2016). The order of transport ratio is $\text{Se} > \text{I} > \text{H} > \text{Sr} > \text{Ni} > \text{Pu}$, and it is considered that the elements which form volatile or water-soluble chemical species should preferentially be transported to the contaminated water. Sr transport ratio was rather small just after the occurrence of the accident, and increased to the extent of Cs as shown in Figure 5.9. This suggests that even a component which makes a solid solution in the fuel would be transported to the water to the same extent as Cs if it forms a water-soluble compound. Figure 5.10 also shows the difference of transport behaviour between the reactor units (Koma et al., 2016).

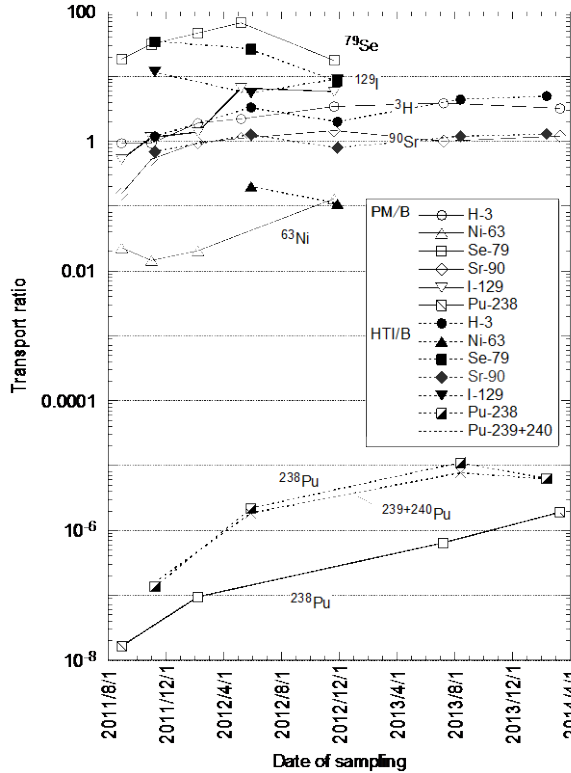
Transport to the rubble which was sampled inside the reactor building is shown in Figure 5.11 and the order is $\text{C} > \text{Co} > \text{Cs} > \text{Tc} > \text{H} > \text{Sr} > \text{Eu-Pu-Am-Cm}$ (Koma et al., 2016). The elements which showed larger values than Cs were different from the case of the contaminated water. As Co is non-volatile, the large value suggests that a source other than the fuel element contributed. Differences between the reactor units were observed and, in unit 2, transport of Sr and actinides was significant. It is implied that the process of fuel damage was so different that the extent of non-volatile elements transported varied for the reactors.

On the other hand, the origins of sample (inside/outside reactor building, material) did not affect the transport of radionuclides. The transport ratios for H-3, C-14, Co-60 and Sr-90 were similar regardless of unit, as shown in Table 5.2 (Koma et al., 2016). Transport ratios of H-3 and C-14 correlate, as shown in Figure 5.12, although neither correlate with Cs-137 (Koma et al., 2016). This finding suggests that the process of transport differs from Cs-Sr and H-C, which come from the damaged fuel cladding. Since reactor 4 did not have fuel for inspection, contamination was influenced by the fuel cladding, fuel crud and cooling water of the spent fuel pool rather than by the fuel elements in reactors 1 through 3.

The order of transport ratio to the soil was $\text{I} > \text{Te} > \text{Cs} > \text{Ag} > \text{Sb} > \text{Mo} > \text{Ru} > \text{Ba} > \text{Sr} > \text{Nb} > \text{Pu-Am-Cm}$ as shown in Figure 5.13 (Koma, 2014). The values for Sr and actinides are close to those of the rubble, and depend on the direction from the reactor units. Sr transport ratio to the wider area around the site scattered around 10^{-3} as shown in Figure 5.14 (Koma et al., 2016) and an apparent change was not observed (the further

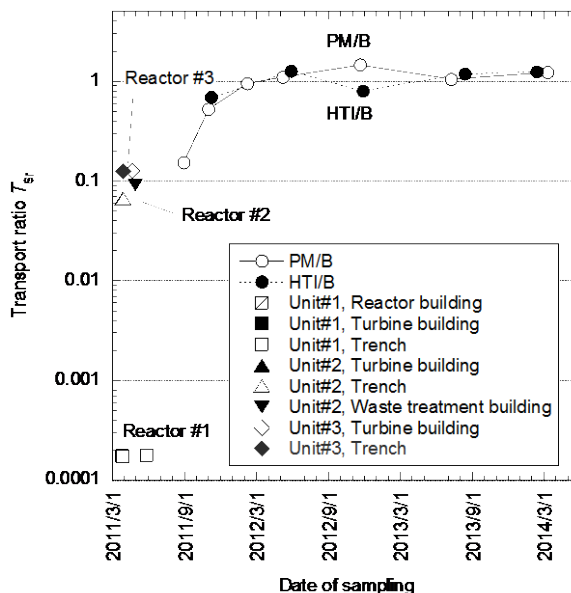
from the site, the lower the Sr concentration, which results in “not detected”). It was considered that the transport ratio of non-volatile elements did not change from inside the reactor building to places several tens of kilometres away.

Figure 5.9. The transport ratio for some nuclides into contaminated water



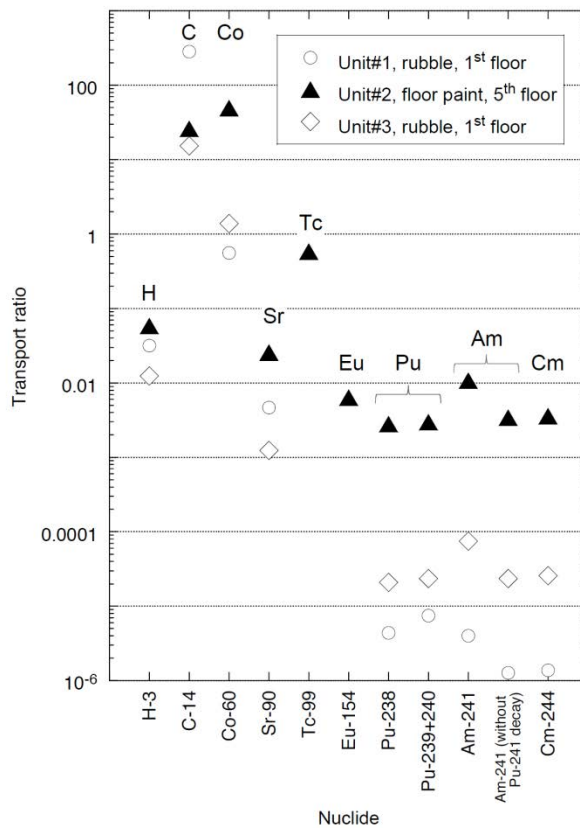
Source: Koma et al., 2016.

Figure 5.10. Increase of Sr transport ratio and difference at the occurrence of the accident



Source: Koma et al., 2016.

Figure 5.11. Transport of some nuclides to the rubble sampled inside the reactor buildings of units 1 through 3



Source: Koma et al., 2016.

Table 5.2. Transport ratio of H-3, C-14, Co-60 and Sr-90 to the rubble

Sampling location		Sample	Number of sample	H-3		C-14		Co-60		Sr-90	
Unit 1	Surroundings	Rubble	5	0.042	(5)*	<290	(1)	3.5	(2)	0.0046	(5)
	1 st floor	Rubble	5	0.036	(5)	260	(5)	<0.92	(4)	0.0046	(5)
	1 st floor	Paint decontaminated	2	<0.4	(0)**	<2×10 ³	(0)	<300	(0)	0.0030	(1)
Unit 2	5 th floor	Paint	1	0.066	(1)	23	(1)	74	(1)	0.026	(1)
	1 st floor	Paint decontaminated	1	0.050	(1)	<100	(0)	<20	(0)	0.014	(1)
Unit 3	Surroundings	Rubble	5	0.011	(5)	<33	(4)	2.3	(4)	0.00012	(5)
	1 st floor	Rubble	4	0.014	(4)	<17	(2)	2.3	(3)	0.0012	(4)
Unit 4	Surroundings	Rubble	5	27	(5)	<2.2×10 ⁵	(4)***	<550	(1)	<0.0013	(2)
	Fuel pool	Gravel, pebble	2					5.2×10 ⁷	(2)		

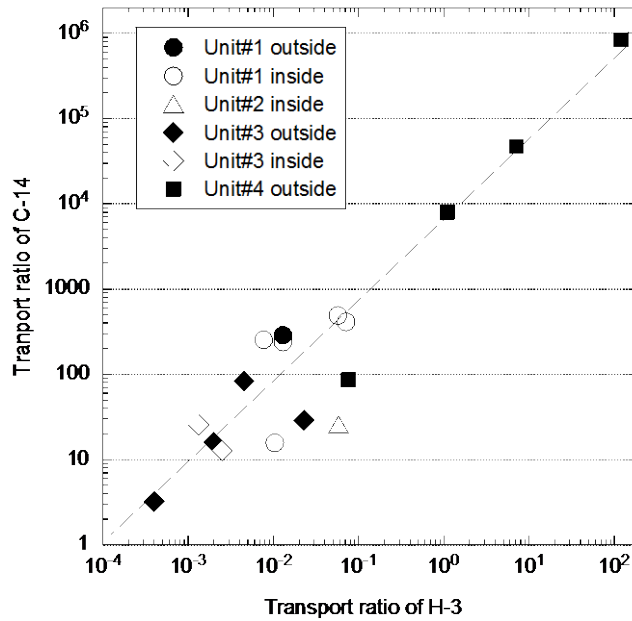
* Number of detection, when the number is smaller than the number of samples, the nuclide was not detected for the rest of the samples.

** The smallest value among the detection limits.

*** The sample(s) of “not detected” had lower detection limit(s) than the concentration obtained.

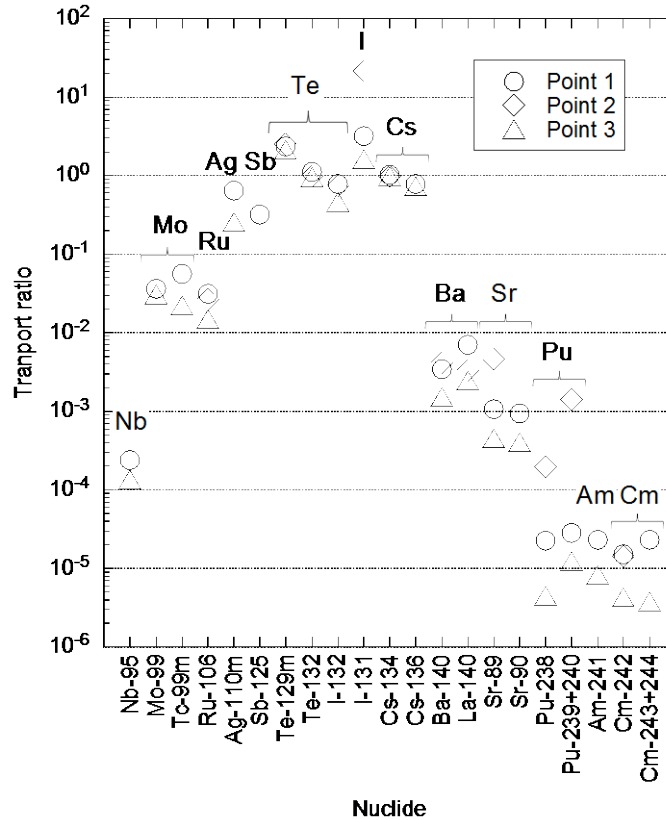
Source: Koma et al., 2016.

Figure 5.12. Correlation of H-3 and C-14 transport ratio to rubble



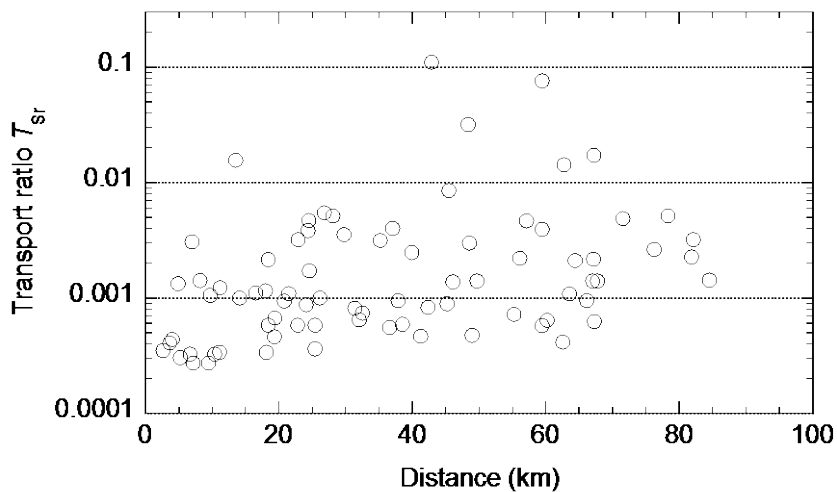
Source: Koma et al., 2016.

Figure 5.13. Transport ratio of some nuclides to the soil at three fixed points on-site



The points were located at the direction from the stack of units 1 and 2 of WNW, W and SSW and referred to as the ground (1), the forest of birds (2) and the neighbourhood of controlled landfill site (3), respectively.
Source: Koma, 2014.

Figure 5.14. Dependency of Sr transport ratio to soil on the distance from the NPP

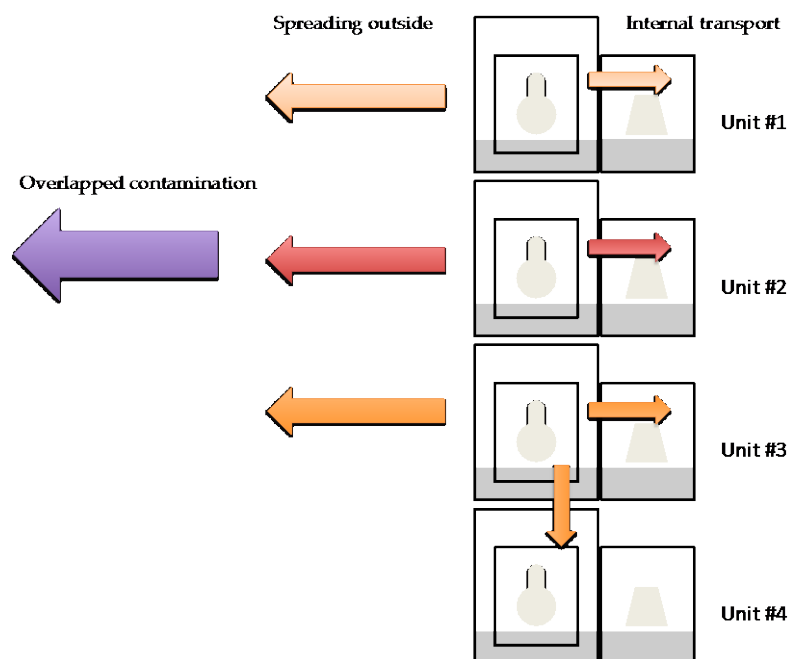


Source: Koma et al., 2016.

The transport ratio of radionuclides to vegetation is dependent on the part of the botanical plant and also time, namely, the life cycle of plant. Generally, radionuclides move from the leaf-branch to fallen leaves, humus or topsoil. According to the analytical data, radionuclides relatively tend to stay in the leaf-branch in the order of H-3 > Sr-90 > Cs-137.

As a consequence, contamination via air from the damaged fuel was modelled as shown in Figure 5.15. Radionuclides are assumed to spread from each of reactor units 1 through 3 with constant transport ratios. Further from the reactor buildings, contaminant plumes from the different reactors overlapped. This assumption should be validated with analysis.

Figure 5.15. A model of contamination with transport from the damaged fuel via air



Source: Koma et al., 2016.

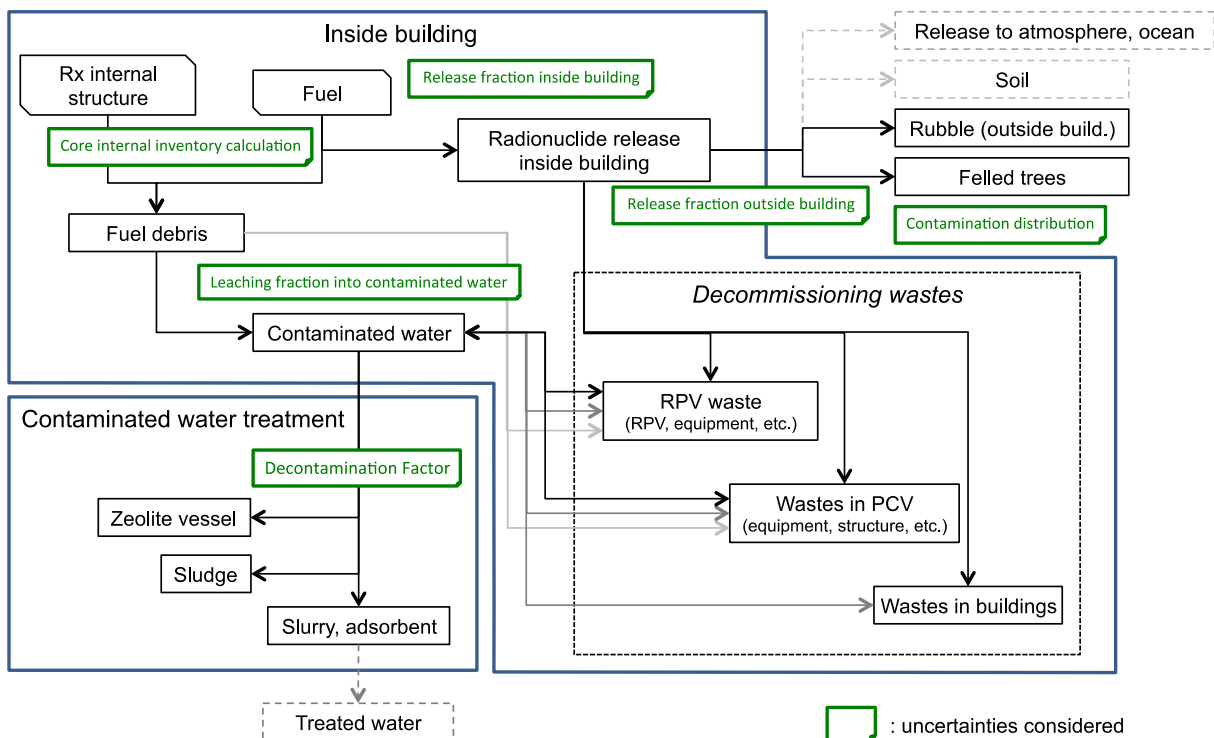
Inventory evaluation

For inventory evaluation, an integrated estimation model has been under development. The current model targets decommissioning waste including the reactor pressure vessel, primary containment vessel and reactor building, secondary waste of water treatment, rubble generated by the explosion and vegetation contaminated with released radionuclides. This model can be improved to reduce uncertainty in estimation by calibration using the analytical data as shown in Figure 5.16.

Inventory of the secondary waste that is difficult to sample is evaluated by using the analytical concentration of the water at the inlet and outlet for the treatment equipment. The estimated Cs-137 inventory for hundreds of zeolite vessels is shown in Figure 5.17 as an example (Kato et al., 2014). It was confirmed that caesium initially released to the contaminated water was already recovered in the zeolite. For the nuclides “not detected” in the water, basic data such as adsorption coefficients and decontamination factors that describe chemical process or processing performance will be employed for estimation.

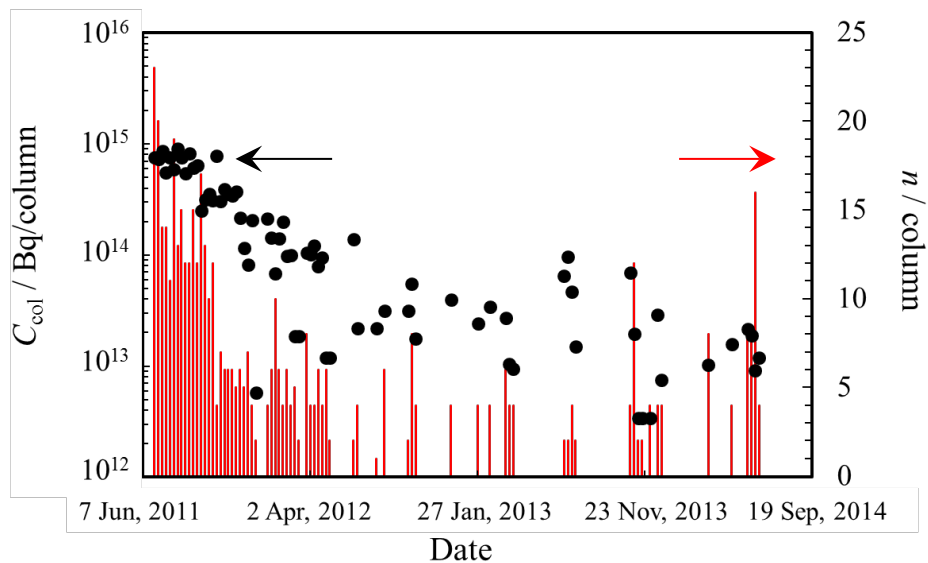
As described above, the series of transport ratios for certain waste differs; therefore, it could be used to categorise various types of waste. In other words, the set of transport ratios will be used as “fingerprints” of this waste. Furthermore, the set of transport ratios is originally connected with a key nuclide, which is Cs-137 in this case, and it will be beneficial to estimate the nuclide composition with the dose rate of waste samples. Practical procedures for categorising waste and inventory evaluation methodology are being further investigated.

Figure 5.16. Integrated model of inventory estimation for 1F waste



Source: Koma et al., 2016.

Figure 5.17. Estimated inventory of Cs-137 for caesium adsorption vessels containing zeolite (black circle) and the number of columns generated per week (red bar)



Source: Kato et al., 2014.

Future plan

Analysing some samples reveals composition and transport behaviour of radionuclides. However, large numbers of samples from various materials must be analysed in the way of retrieving the fuel debris and D&D in the future. Based on the obtained data and findings, a mid- and long-term plan for analysis is being discussed with consideration given to the following:

- The purpose of the analytical programme is to provide information on all of the existing and expected waste for inventory evaluation and R&D on waste management including processing and disposal.
- Radionuclide fingerprints will be developed for all important materials and waste, which will allow future samples to be characterised based on the analysis of a small number of easy-to-measure radionuclides.
- A new facility will be constructed near the site at the end of F.Y.2017 of Japan, and will have the capability to analyse large numbers of samples.
- For inventory evaluation, at present, a cautious approach is taken that assumes the concentration of a radionuclide measured to be below the detection limit is equal to the detection limit. This could lead to waste being assigned to a category that is higher than necessary. Therefore, in addition to increasing analytical capacity, it will be necessary to introduce higher sensitivity of some analysis systems, to reduce detection limits of some radionuclides, and to incorporate the calculation of activity for “difficult-to-measure” nuclides contained in the damaged fuel and structural materials.
- Standardising the analytical methods for efficient operation, quality management of data and waste conditioning.
- Training of technical experts.

The analytical data for waste will be accumulated in a database and be used in the R&D study on their processing, storage and disposal in order to present technical proposals for each waste stream.

Lessons learnt from the Fukushima Daiichi accident

As part of the programme to ensure the safe management of waste from the Fukushima accident, the “Examination of Waste Stream” project has been launched. One aspect of this project is the characterisation of waste. The following lessons have been learnt from the work undertaken to date:

- The need to increase the capacity of the analytical laboratories. Increased analytical capacity is required in order to provide the required information on all existing and expected waste from the Fukushima accident. The requirement to increase the number of samples being analysed has been recognised, and a new analytical facility will be constructed near the Fukushima site at the end of 2017. This also requires training of additional technical experts.
- The need to develop and optimise analytical methodologies. At the time of the accident, analytical methodologies were not available to analyse all of the required radionuclides (including those identified as being safety-relevant for the disposal of radioactive waste) in all of the relevant materials (e.g. concrete, soil, vegetation). New sample preparation and radiochemical techniques to analyse these radionuclides in a range of liquid and solid matrices have been developed and work is continuing. Examples are the development of rapid Sr-90 analysis using inductively-coupled plasma mass spectrometry (ICP-MS) and the application of beta spectroscopy for non-destructive measurement. For some radionuclides, the requirement to improve analytical detection limits has been recognised. Work is also ongoing to standardise analytical methods for efficient operation and quality management.
- The need to improve the validity and reliability of radionuclide inventory estimation in the waste. At the present time, analytical effort is focused on improving the validity and reliability for the estimation of important nuclide inventory in the waste. In order to improve the validity of the inventory data obtained from an inventory analysis model, it is necessary to develop the method of the model calibration based on comparing the analytical results with the inventory data and to analyse the proper radionuclides identified for calibration. Such information facilitates improvement of the validity of the estimation of the radionuclide concentration in the waste. In addition, such improvement facilitates the determination of the fingerprints.
- Approaches to radiological characterisation. At present, waste stored on the Fukushima site has been largely sentenced on the basis of surface dose rate. It is recognised that more widespread use of on-site radiochemical detection systems (such as gamma spectrometry systems) would enable a better understanding of the composition of such waste and improve understanding of its future behaviour.

5.4. Recommendations for post-accident radiological characterisation

Member countries should develop plans to be implemented in the event of a future accident. These plans should include consideration of the amounts and types of radiological characterisation data to be collected, and the best approaches for obtaining these data.

Based on the accident case studies presented in this chapter and on the current “state of the art” of radiological characterisation for “normal” decommissioning

programmes, we make the following recommendations for radiological characterisation after an accident.

Analytical testing capacity and methods

- Sampling plans, which specify the numbers, types and locations of samples to be analysed should be developed and justified. Such plans will be iterative, and will develop as understanding of radionuclide concentrations and their spatial distribution improves. At the beginning of the project, such plans are likely to involve combinations of expert judgement and statistical considerations (for example, based on the DQO approach). As characterisation data are obtained, it may be appropriate to also include geostatistical approaches to optimise future data collection.
- The ability to radiologically characterise materials should not be the limiting factor that controls the rate of decommissioning activities. It is the view of the expert group that acquiring sufficient radiological characterisation data is likely to be the most important short-term challenge following an accident. Sufficient characterisation equipment should be made available, both in off-site analytical testing laboratories and on the accident site to meet the needs of the programme.
- In addition, suitable sample preparation and analysis methods should be identified for all of the significant radionuclides (for example, hard-to-detect beta nuclides) and material types that will require characterisation (concrete, metals, soil, vegetation, etc.).
- The proposed radiological characterisation should enable waste to be assigned to existing waste categories, but it will also be important to ensure that sufficient characterisation data are collected to enable waste to be sentenced against alternative categories if these were to be applied at a later date.

Approach to radiological characterisation

- In the short term after an accident, it will be necessary to sort waste (for example, to consign them to appropriate storage facilities) on the basis of quick, simple, easily measurable parameters such as surface dose rate. As soon as possible after the accident, routine on-site analysis for easy-to-measure radionuclides such as Cs-137 should be started. This would allow the activities of a wide range of radionuclides in the measured materials to be calculated once radionuclide fingerprints had been established.
- As soon as practicable, radionuclide fingerprints should be established for all the materials and waste streams that will be produced. This will allow materials to be characterised and waste consigned on the basis of analysing a small number of “easy-to-measure” radionuclides. The approach to determining radionuclide fingerprint should be documented and justified. Calculations based on an understanding of material chemistry, the irradiation history of the material and fuel burn-up will be required, in addition to analytical measurements to determine fingerprints for higher-activity waste. In lower-activity waste, where contamination (for example, by airborne deposition) is a major contributor to total activity, most emphasis is likely to be placed on direct measurement.
- Radiological characterisation should be suitable to categorise and sentence waste for both storage and disposal.
- Approaches for radiological characterisation have been developed for the various stages in conventional “non-accident” waste management and decommissioning programmes, and these approaches should be reviewed and adopted where appropriate. Various OECD reports on this subject have already been produced.

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6. Waste classification and categorisation

6.1. General description

International recommendations and guidance, general waste classification

Requirements for decommissioning of nuclear facilities (EC, 1999) include the need for:

- clearly defined regulatory requirements for decommissioning of nuclear facilities;
- clearly defined waste management and disposal routes.

Appropriate classification and categorisation of waste consistent with those requirements and disposal routes are important tools in the development and implementation of a decommissioning strategy.

International guidance on the classification of radioactive waste has been provided by the International Atomic Energy Agency (IAEA), most recently in IAEA 2009a. The objective was to set out a general scheme for classifying radioactive waste that is based primarily on considerations of long-term safety and disposal of the waste. The range of solid waste considered in the guidance is very broad, but focuses on solid radioactive waste. However, the fundamental approach to classification could also be applicable to the management of liquid and gaseous waste, with appropriate consideration given to aspects including the processing of such waste to produce a solid waste form that is suitable for disposal.

The IAEA (2009a) emphasises that, apart from waste containing only short-lived radionuclides, all other types of radioactive waste need to be eventually disposed of in a manner consistent with the Fundamental Safety Principles (IAEA, 2006a) and with safety requirements for the pre-disposal management of radioactive waste (IAEA, 2009b) and for the disposal of radioactive waste (IAEA, 2011; ICRP, 2013). In particular, a report by the IAEA (see IAEA, 2006a) says that radioactive waste shall be characterised and classified in accordance with requirements established or approved by the regulatory body. The characterisation process provides information relevant to process control and assurance that the waste or waste packages will meet the acceptance criteria for processing, storage, transport and disposal of the waste.

The relevant characteristics of the waste have to be recorded to facilitate its further management. Such characteristics include the radiological, chemical, physical and biological properties of the waste and data on the particular radionuclides it contains (IAEA, 2009a). Radioactive waste may present non-radiological hazards as well as radiological hazards, which may also require consideration from a safety and regulatory perspective. Examples include contaminants such as PCB and heavy metals (NEA, 2014). Also particularly relevant in decommissioning is asbestos waste (LLW Repository Ltd, 2011).

The IAEA's classification guide was also intended to facilitate communication. Such a commonly applied classification scheme facilitates communication of waste management practices internationally, particularly in the context of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (IAEA, 2006b). However, it is quite technical in nature and does not indicate the scale of hazard associated with the waste in a manner which supports ready

understanding by general stakeholders who are not specialists in radiological protection or radioactive waste management.

While noting the above important high-level international recommendations and guidance documents, it is also appropriate to recognise that there is a need for further, more detailed consideration at the national level, to take account of local circumstances. For a comprehensive example of waste characterisation and categorisation, see the descriptions and data for the UK radioactive waste inventory at www.nda.gov.uk/ukinventory.

Given the significance of damaged fuel and fuel debris (NDF, 2015), an important example in the Fukushima Daiichi nuclear power plant (NPP) decommissioning waste (1F) inventory concerns the definition of high-level waste (HLW). The IAEA (2009a) defines it as “waste with levels of activity concentration high enough to generate significant quantities of heat by the radioactive decay process or waste with large amounts of long-lived radionuclides that need to be considered in the design of a disposal facility for such waste.” The quantitative definition of significant quantities of heat is not provided by the IAEA, since significance will depend on the local circumstances.

UK waste package specifications (Nirex, 2007a) require that, “the heat output from all sources within the waste package (including radiogenic, chemical and biological sources) shall be limited to a value that will prevent excessive temperature rise within the waste package during all the phases of [management].” This is then quantitatively specified as, “the heat output from a 500-litre drum waste package should not exceed 50 W for transport and 25 W at the time of vault backfilling.” Note that this specification allows for all heat generated within the package (unlike the IAEA definition of HLW), and accounts for transport as well as disposal safety. The technical basis by which these numbers are justified is provided in (Nirex, 2007b). Such analyses may be helpful in deciding whether small amounts of damaged spent fuel need to be considered as HLW or as heat generating.

Also important for the categorisation of damaged fuel and fuel debris is its status as a waste. For the case of damaged spent fuel stored at Andreeva Bay (see Chapter 1) it was necessary to develop specific guidance setting out the arrangements for safe management of objects containing nuclear materials, while transferring them to the category of radioactive waste (FMBA, 2011).

An approximate comparison of the international IAEA waste classification scheme and that in the United States is shown in Table 6.1:

- Very low-level waste (VLLW) would be equivalent to the lower end of the class A in the scheme of the US Nuclear Regulatory Commission (NRC).
- Low-level waste (LLW) spans the remainder of class A waste, includes all class B waste, and extends into the lower end of class C waste in the NRC scheme.
- Intermediate-level waste (ILW) includes the remainder of, and extends beyond, class C to include the US designation of greater than class C (GTCC).

Many national schemes show variation from the IAEA scheme.

Table 6.1. Comparison of the IAEA (IAEA, 2009a) and NRC waste classification schemes (NRC, 2015)

NRC part 61.55	Class A	Class B	Class C	Exceeds class C or GTCC
IAEA GSG-1	VLLW	LLW	ILW	HLW

VLLW: Very low-level waste; LLW: Low-level waste; ILW: Intermediate-level waste; HLW: High-level waste.

International recommendations and guidance, waste classification for nuclear facility decommissioning

Radiological characterisation of waste arising in the decommissioning of nuclear facilities in normal (non-accident) circumstances has been considered in detail by the NEA Working Party on Decommissioning and Dismantling (NEA, 2013). Here, radiological characterisation with respect to decommissioning shall, among other things:

- determine waste classifications for packaging, shipping and disposal;
- determine which remedial actions will be needed, including the extent of decontamination that will be required.

In particular, the NEA (2013) notes that the most comprehensive characterisation campaigns are usually carried out during the transition phase in preparation for implementation of dismantling activities, or during the dismantling phase where systems, structures, components and buildings have to be characterised for decisions regarding the extent of decontamination, application of appropriate dismantling techniques, identification, classification, treatment of radioactive materials, etc. The final status survey on the site has quite distinctive features as it also has to take into account the possibility of subsurface contamination, which may lead to radionuclide transfer into ground water and surface water bodies.

Radiological characterisation efforts are needed during all stages of a nuclear facility's life cycle, in order to plan and perform decommissioning in a safe and efficient manner.

The above considerations are considered relevant to decommissioning after an accident. It is notable that, even in a planned situation, significant uncertainties arise, e.g. concerning the nature and extent of subsurface and groundwater contamination, which need to be accommodated in any waste classification scheme.

Decommissioning after an accident was explicitly considered in an IAEA report (see IAEA, 2013). Here it was suggested from experiences at other sites that the existing legislation may not be adequate to deal with the waste arising from an accident. In turn, this may hamper the development of solutions for managing large quantities of lower-activity waste. For example, waste classifications existing prior to the Chernobyl accident did not have sufficient scope to encompass the diversity of radioactive waste that arose after the accident. The same conclusion can be drawn from the development of regulatory guidance on VLLW management specifically needed in relation to waste arising at the site for temporary storage at the Andreeva Bay (see Chapter 1). It may be noted that in this case, where the radiological risks from the VLLW were very low, the guidance was designed to fit within the framework of hazardous waste management (FMBA, 2008).

An NEA task group reported on regulatory challenges in decommissioning in an NEA report from 2003. It was concluded that it is important that requirements and responsibilities be defined clearly, particularly in the cases where interim storage is built to store waste until a final disposal site is available. A particularly difficult challenge recognised for the regulator was to establish a clear set of site release criteria for terminating the licence. At that time of writing the NEA report in 2003, there was "no consensus within OECD countries on a preferred set of site release criteria or even the form of such criteria". This is still an area of continuing work; for example by the NEA WPDD Task Group on Nuclear Site Restoration. The publication of this task group is expected in 2016. Whatever conditions or criteria are chosen, it is important for openness and transparency, and ultimate public acceptance of the decommissioning process, for the operator to have public discussions of the site release criteria.

6.2. Case studies

Three Mile Island 2 (TMI-2)

Abnormal waste

Much of the radioactive waste that resulted from the TMI-2 accident clean-up could not be disposed of as low-level waste and there was no disposal facility for this higher level or “abnormal waste”. In addition much of the waste were not comparable to those produced at an operating power plant.

The waste contained a high concentration of fission products or small quantities of fuel materials. The waste processing systems had not always been configured to produce waste in the form and concentrations allowed for shallow land burial.

As part of the solution, the US Department of Energy (DOE) and the NRC signed a Memorandum of Understanding (MOU) in July 1981 to ensure the TMI site did not become a long-term waste disposal facility. The agreement also took advantage of the chance to learn from the accident. The DOE agreed to evaluate each form of waste to determine the research and development (R&D) value and if of value, to accept the waste for research and later disposal. If the waste was not of research value or could not be made acceptable for commercial disposal, the DOE would temporarily accept and store the waste on a cost-reimbursable basis. This agreement was crucial for disposal of all the TMI-2 radioactive waste.

The MOU identified several types of radioactive waste and potential means of disposal, and these included:

- EPICOR-II waste – For the highly loaded prefilters, the DOE proposed to develop a high-integrity container that might allow commercial land burial at Richland. Characterisation work would also be performed on one or more vessels.
- Submerged demineraliser system (SDS) waste – For the 19 highly loaded SDS vessels, the DOE would conduct a waste immobilisation R&D and testing programme, including monitored retrievable burial.
- Reactor fuel – Initially, the DOE planned to take samples for analysis, characterisation and research while the balance of the fuel debris remained on-site in the spent fuel pool. Final disposition would await resolution of the national spent fuel issue. As the issue was going to take a long time to resolve (and is still not resolved as of the publication of this report), the DOE and NRC modified the MOU in March 1982 so that the DOE accepted the entire reactor fuel core. Part would be used for R&D; and the remainder would be stored until ultimately disposed of. The TMI-2 damaged fuel is currently in dry cask storage at the Idaho National Laboratory.

Lessons learnt from the TMI-2 accident

The criteria for siting of a nuclear power facility is not the same as the criteria for siting a waste storage/disposal facility and thus plans need to be developed to remove waste resulting from an accident from the reactor site.

A significant quantity of waste arising from clean-up after an accident will not meet existing waste acceptance criteria and thus the owner of the facility will need to work with government and research institutions to develop acceptance criteria for this waste.

Chernobyl radioactive waste management

Waste classification in Ukraine: Current situation and problems

Currently, Ukrainian legislation defines several radioactive waste classification systems depending on the tasks for which they were developed:

- Classification of radioactive waste on the state of aggregation:
 - solid radioactive waste (SRW);
 - liquid radioactive waste (LRW).
- Classification of solid radioactive waste based on the criteria of “exemption level”, set for a given group of radionuclides contained in radioactive waste into four groups.
- Classification of solid and liquid radioactive waste by specific radioactivity in the category of low-, medium- and high-level radioactive waste. The category of high-level radioactive waste is divided into two sub-categories: “low temperature” and “heat generating”.
- Classification of waste with unknown radionuclide composition and unknown specific activity by the criteria of absorbed dose rate in air at a distance of 0.1 m from the surface of the object (container) for “low”, “medium” and “high level”.
- Classification of radioactive waste based on the half-life time of radionuclides in the waste:
 - short-lived (“day-long”, “month-long”, “year-long”);
 - medium-lived;
 - long-lived.
- Classification by type of production and sources of waste.
- Classification of radioactive waste in terms of exemption from regulatory control with respect to disposal of radioactive waste into two types: short-lived (achieved in less than 300 years after the disposal, near-surface disposal facilities), and long-lived (they cannot be exempt from control in 300 years, disposed of in geological formations).

The last “classification” is the most important and based on the definition of short-lived and long-lived radioactive waste that is done in the Law on radioactive waste (RW) management and “Basic Sanitary Rules of Radiation Safety in Ukraine – 2005”. It means that in Ukraine only two classes of RW are possible and only disposal of those classes is possible – near-surface and geological. In fact the classification of RW by type of disposal is not practically used by “RW producers” in Ukraine.

To improve this situation and to find approaches on how to update the RW classification in Ukraine, the international project “Support of Introduction of the New Classification System to the Regulatory Framework of Ukraine” was implemented. This project was supported by the European Commission and performed by the experts from ANDRA (France), COVRA (Netherlands), DBE Technology (Germany), ENRESA (Spain) and SKB (Sweden). The team of European experts reviewed Ukrainian legislation and after analysis of existing requirements established for RW classification, treatment, conditioning and disposal proposed some approaches for establishing a new classification based on the disposal methods.

Based on the best international practices in the area of waste management, it is proposed to allocate the following classes:

- exempted waste (EW);
- naturally occurring radioactive materials (NORM) waste;
- VLLW;
- LLW;

- ILW;
- HLW;
- used sealed sources (dust suppression system – DSS).

Radioactive waste assignment to a particular class, definition of the values of maximum permissible content of specific radionuclides in the waste is carried out in accordance with the type of disposal; the boundaries between the classes are defined based on general acceptance criteria for disposal of waste in the disposal facility of the appropriate type. It was proposed to delete the following terms and their definitions: “long-lived radioactive waste”, “short-lived radioactive waste” and to introduce the new terms and definitions for RW classes, and update the definition of “disposal of RW” and “facility for RW disposal”.

As was mentioned above, the basic idea of the new waste classification system is that the waste should be classified not in accordance with the waste production, conditioning and packaging, as well as their physical and chemical characteristics, but according to the future method of waste disposal. In order to achieve this goal, the general waste acceptance criteria should be developed for all prospective methods of waste disposal in:

- surface storage (like solid waste disposal);
- the near-surface storage with engineered barriers system;
- underground storage at intermediate depths (shallow depositories);
- deep geological repositories.

An important conclusion was reached regarding cost issues related to implementation of the new RW classification system. Rather than increasing the waste management costs, a new waste classification system can provide significant cost savings over time, making possible the waste distribution by optimised disposal methods and storage types. In particular, this applies to the VLLW which would have been classified as LLW if there was no VLLW category, which, in turn, would have required a more expensive disposal in a constructed near-surface repository.

On the other hand, in order to take full advantage of all the benefits of a new waste classification system, some of the sorting and characterisation must be undertaken immediately after the waste generation. This will involve some costs straight away that may vary over time. But experience suggests that it will be much easier and cheaper to sort and characterise waste immediately after its generation than sorting and characterising mixed waste from mixed storage later. Moreover, with the full use of the above correlation methods for determining radionuclides in non-mixed waste streams, the radiation exposure received by employees performing preliminary waste sorting and characterisation will be much lower than during complete sorting of mixed waste, and later on during complete characterisation, with the latter being a costly and a time-consuming operation.

Following the newly developed methodological approach, special attention will be given to the large quantities of radioactive waste generated by the accident at the Chernobyl nuclear power plant (ChNPP), so-called Chernobyl waste, which will be disposed of in the Chernobyl exclusion zone. As a result of the Chernobyl disaster, a large amount of waste with a low level of activity was generated, with the main contribution to the total activity by Cs-137 and Sr-90. However, the waste also contains fission products with long half-lives (alpha nuclides). Applying activity limits for waste acceptance criteria used in Europe for disposal in near-surface repositories, it would be impossible to dispose large amounts of Chernobyl waste in this type of storage. It would have been disposed of at an intermediate depth in storage for ILW with severe consequences for the economy.

However, the waste acceptance criteria have been developed considering the basic radiological protection criteria (such as the annual amount of risk, or the maximum annual limit of radiation dose from the repository), and taking into account all the possible scenarios of the disposal evolution. Thus, obtained waste acceptance criteria will be less stringent for the near-surface repository located within the exclusion zone. When analysing the safety of storages in the exclusion zone, it makes sense to consider the repository's evolution and other relevant scenarios for radiation assessment. For example, distance to wells used for drinking water will be much greater, area for food crops will be more remote, etc. Thus, without mitigating basic radiological protection criteria, less stringent acceptance criteria for LLW and VLLW of Chernobyl origin located in the near-surface storages in the exclusion zone can be developed and justified.

This approach is in full compliance with the safety assessment methodology developed in the DBE Technology GmbH for the radioactive waste disposal site "Buryakovka" within the framework of project "Improving Infrastructure for Radioactive Waste Management in the Chernobyl Exclusion Zone – Phase I: Safety Assessment" and is completely approved by the Ukrainian regulatory body based on this methodology, where less stringent waste acceptance criteria for disposing of Chernobyl waste in LLW and VLLW repositories inside the exclusion zone is justified. On the one hand, this is caused by limiting human access and land use in a zone where waste will be stored for a long period, and, on the other hand, by the fact that such disposal activities are considered as "interference" case.

Lessons learnt and conclusions

- Existing RW classification for short- and long-lived waste is not implemented practically because of lack of clear criteria for the RW producer on how to separate these classes.
- "Chernobyl RW" is not defined as a separate class; it means that management of such waste should be based on existing rules and regulations. Such an approach is not effective taking into account the large amounts of "Chernobyl RW" located in the exclusion zone. Moreover, taking into account strict safety requirements for near-surface disposal of RW, it could be practically impossible to organise geological disposal for all RW, that cannot be disposed of in near-surface facilities.
- RW classification in Ukraine needs to be updated and clarified for RW producers. A number of changes should be done in terms and definitions related to RW classifications, treatment and disposal.
- RW classification should be based on disposal methods for every RW class.
- The general waste acceptance criteria should be developed for every RW Class taking into account disposal methods.
- The new RW classification proposed in the frame of EC supported project can save money (no need to provide additional sorting and characterisation of RW).
- For the management of large quantities of radioactive waste generated by the accident at the ChNPP, a special approach should be developed taking into account the peculiarity of Chernobyl exclusion zone (no people living in the zone).
- The plan to include the new classification in Ukrainian legislation has been developed but progress is very slow.

Existing waste classification and categorisation in Japan

Existing waste classification for disposal

The burial of radioactive waste is classified as either category 1 waste disposal or category 2 waste disposal, in accordance with the Act on the Regulation of Nuclear

Source Material, Nuclear Fuel Material and Reactors (Reactor Regulation Act) (Table 6.2, Figure 6.1).

Category 1 waste disposal requires greater radiological safety care. Such waste contains high radioactivity concentrations with long half-life nuclides, and because of this the waste must be isolated from living environments for a long period. Therefore, in addition to regulation governing category 2 waste disposal described below, the approval of the closure plan and procedures to check conformity with the approved closure plan are also stipulated to ensure the proper closure of underground facilities.

Category 2 waste disposal regulations apply to radioactive materials in solid waste radioactive material, which has a radioactivity concentration below the upper limit specified in the Order for Enforcement of the Reactor Regulation Act. It is buried by intermediate depth disposal, near-surface pit disposal, or near-surface trench disposal, depending on the nuclides present in their individual concentrations. Consideration is given to radioactive decay in the management of such waste.

General maximum waste concentrations for each type of disposal are assigned in the Reactor Regulation Act and the Ordinance on Standards for the Location, Structure, and Equipment of Category 2 Waste Disposal Facilities (Ordinance of Category 2 Waste) (Figure 6.1). Specific maximum waste concentrations and their total radioactivities for each specific disposal assigned by implementer should be selected so that their effects for the public are less than the dose criteria. Dose criteria during the operational and after closure period (approximately 300 y) is 50 $\mu\text{Sv/y}$ for basic scenario and 5 mSv/y for accident scenario. Dose criteria for after the termination of the licence (300 to 400 years after closure of the pit type of disposal) are 10 $\mu\text{Sv/y}$ for a likely scenario with most probable condition, 300 $\mu\text{Sv/y}$ for a less likely scenario with range of uncertainty, and 1 mSv/y for other natural event scenario and human intrusion scenario.

Waste for category 2 waste disposal should meet the following technical standards:

- i. in the case of intermediate depth disposal:
 - a) radioactive waste to be disposed of shall be generated at the factory or place of business where a fuel facility (limited to a facility solely conducting the fabrication and enrichment of fuel assemblies that contain mixed uranium and plutonium oxide), research and test reactor facility, power reactor facility or reprocessing facility is installed;
 - b) radioactive waste to be disposed of shall be a waste package;
 - c) the waste package shall be as specified in the following paragraph.
- ii. in the case of pit disposal:
 - a) radioactive waste to be disposed of shall be generated at the factory or place of business where a research and test reactor facility or power reactor facility is installed;
 - b) radioactive waste to be disposed of shall be solidified concrete waste or a waste package;
 - c) waste package or solidified concrete waste shall be as specified in the following paragraph or paragraph (3).
- iii. in the case of trench disposal:
 - a) radioactive waste to be disposed of shall be generated at the factory or place of business where a research and test reactor facility or power reactor facility is installed;
 - b) radioactive waste to be disposed of shall be solidified concrete waste;

c) solidified concrete waste shall be as specified in paragraph (3).

Technical standards for a waste package are as follows:

- i. for the prevention of radiation hazards, radioactive waste shall be encapsulated in a vessel or solidified with a vessel by the method specified by the Nuclear Regulation Authority;
- ii. the radioactivity concentration shall not exceed the maximum radioactivity concentration stated in the application for;
- iii. the surface density of radioactive material shall not exceed 10% of the surface density limit as set forth in Article 14, item (i), (c);
- iv. any material that may damage the integrity of the waste package shall not be included;
- v. waste package shall have enough strength to bear the potential load that may be applied when landfilled;
- vi. there shall be no significant damage;
- vii. a radioactive waste sign shall be attached in a prominent place on the surface of the waste package so that it cannot be easily dislodged, and a serial number for cross checking said waste package with the details stated in the application form set forth in the preceding article shall also be indicated.

Technical standards for solidified concrete waste are as follows:

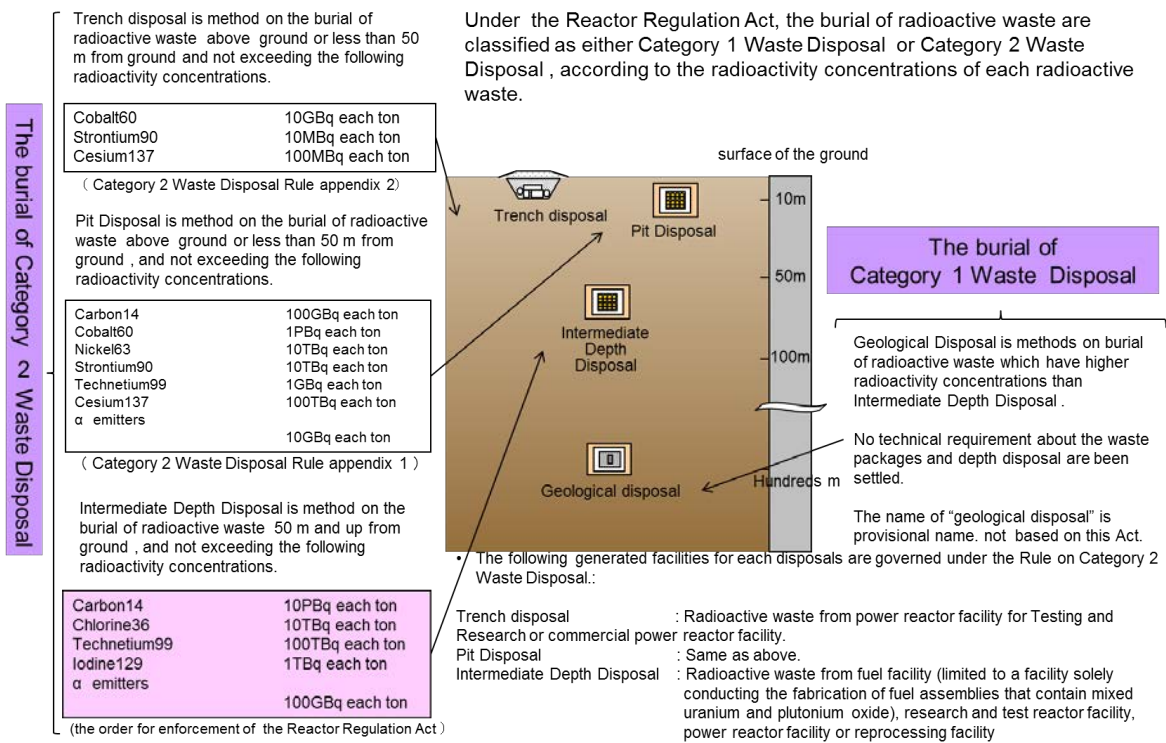
- i. explosive materials shall not be included;
- ii. measures for cross checking said solidified concrete waste with the matters stated in the application form set forth in the preceding article shall be taken.

Table 6.2. Radioactive waste management prescribed in the Reactor Regulation Act

Category	The burial of category 1 waste disposal	The burial of category 2 waste disposal		
		Intermediate depth disposal	Pit disposal	Trench disposal
Name	N/A ¹	Intermediate depth disposal	Pit disposal	Trench disposal
Contents	Final disposal by a method for the burial of radioactive waste in the excess of criteria defined by order ² as they have potential significant risks to human health.	Final disposal by a method for the burial of radioactive waste ⁴ at a depth of 50 m and up from ground, and not exceeding criteria defined by order. ²	Final disposal by a method for the burial of radioactive waste ⁵ above ground or less than 50 m from ground, and not exceeding criteria defined by the rule ³ (limited to methods either to fix radioactive waste at waste disposal sites with the engineered barrier structure or fix integrally radioactive waste at waste disposal sites without the engineered barrier structure).	Final disposal by a method for the burial of radioactive waste ⁵ above ground or less than 50 m from ground, and not exceeding criteria defined by the rule ³ (excluding for methods either to fix radioactive waste at waste disposal sites with the engineered barrier structure or fix integrally radioactive waste at waste disposal sites without the engineered barrier structure).

1. The name "geological disposal" is not based on the Reactor Regulation Act, but often used in order to distinguish it from other forms of waste disposal.
2. The Order for Enforcement of the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors.
3. The Rule on Category 2 Waste Disposal of Nuclear Fuel Material and Materials Contaminated with Nuclear Fuel Material.
4. Radioactive waste from a fuel facility (limited to a facility solely conducting the fabrication of fuel assemblies that contain mixed uranium and plutonium oxide), research and test reactor facility, power reactor facility or reprocessing facility.
5. Radioactive waste from research and test reactor facility or power reactor facility.

Figure 6.1. Methods for the burial of radioactive waste for final disposal



Source: NRA, 2014.

Waste in Fukushima Daiichi NPP

Rubble and other waste from the accident at TEPCO’s Fukushima Daiichi NPP are stored at the power plant site. They are classified by their characteristics and surface dose rates.

As mentioned above, Fukushima Daiichi NPP is designated as the “specified nuclear power facilities”, and with this system, waste management methods to minimise the total risk were applied with the approval of the NRA.

Lessons learnt and challenges for waste classification waste in Fukushima Daiichi NPP

Not enough information is available about Fukushima Daiichi NPP waste to categorise it according to the existing classification scheme for waste arising from normally operated NPPs. Adopting those classification systems to the waste in Fukushima Daiichi NPP presents difficulties because of the large amounts, high activity or unsettled conditions. Nevertheless, minimum characteristics should be known for the safety of disposal.

Moreover, keeping in mind that the storage of waste could be carried out over long periods, more consideration should be given to stable storage methods or appropriate classification and characterisation of the waste.

6.3. Lessons learnt

The NEA has reported on 30 years of experience in nuclear decommissioning (NEA, 2011), and specific trends have been observed regarding technical challenges over the years. Large contaminated components, such as heat exchangers, steam generators or large tanks, that have previously been cut in situ into smaller pieces, are increasingly removed “in one piece” and transported outside the contained area into separated facilities for

further processing. Regarding the use of robotics, it was observed that industrial robots may have limited practical applicability in decommissioning, contrary to earlier expectations that robotic methods would be extensively used in the decontamination and dismantling of radioactive structures and components. However, they will remain necessary for some applications especially in high radiation areas. The clean-up and verification for the release or declassification of alpha contaminated concrete structures, where seepage of contamination has occurred in cracks and along pipe penetrations, has proven to be very challenging and in fact in some cases has prompted authorities to impose much more stringent release criteria. The experience gained in relation to classification and categorisation in nuclear decommissioning projects may be relevant after accidents as well as in normal conditions.

Careful planning and implementation of radiological characterisation campaigns are expected to result in significant reductions in time, costs and effort. On a strategic and managerial level, there are ways to maximise the efficiency of measurement techniques (e.g. by combining several types of measurement and sampling approaches) to increase efficiency of characterisation (e.g. by integrating characterisation into other tasks), or to choose an optimum form of organisation by allocating staff and resources in a timely and adequate manner to achieve the required characterisation results when needed, thus avoiding delays in the normal decommissioning workflow or radioactive waste management.

Preliminary assessment at an early stage of levels of risk or dose associated with sources of contamination and initial estimates of derived concentration levels which meet regulatory criteria can help to identify the areas of greatest radiological concern and/or greatest uncertainty. This relies on an early understanding of safety criteria and a preliminary view of waste classification and categorisation. In this context, dialogue with regulatory authorities and other stakeholders as early as possible is especially important. Lack of clearance regulations or of clear definitions of the clearance process leads to uncertainties regarding the detection limits that have to be achieved during radiological characterisation.

The NEA has also previously evaluated technology innovation with respect to nuclear decommissioning (NEA, 2014). Here, it was noted that all materials at the facility must to some extent, be characterised and sentenced. The sentencing process involves determination of the most cost-effective ultimate disposition of the material. This may be to leave the material on-site and verify that it meets site clearance criteria. It may involve targeting it for asset recovery to be sold and used at another facility or to be cleared and sold for recycling. It can also mean determining the most likely suitable waste classification: e.g. very low-level radioactive waste (VLLRW), low-level radioactive waste (LLRW), intermediate-level radioactive waste (ILRW) or high-level radioactive waste (HLRW) and methods to optimise disposal options for the materials. Many materials, such as activated graphite, reactor internals and high-activity sludges or organic materials such as resins, pose significant challenges to handle, stabilise and package in ways that are suitable for interim storage and long-term disposal, and therefore require significant investments in the development and study of final waste disposal facilities, including analysis of safety.

Also relevant in the current context, an NEA report on R&D and innovation needs for decommissioning (2014) highlights the role of targeted characterisation activities. These efforts are used to provide more detailed knowledge of the contaminants to support the planning of decommissioning activities for industrial safety and environmental considerations, as well as to plan the removal, treatment (e.g. decontamination and stabilisation), packaging, transport and ultimate disposition of the materials. They are also used to plan clearance or the sentencing survey, assay or sampling protocols. High-priority or high-risk components are targeted for more detailed survey sampling and assay efforts. This can involve more rigorous and detailed surveys, accessing system and component interiors for sampling and survey, or coring structures to determine

contaminant profiles prior to disassembly. It can also involve detailed analysis using computer models and material properties for activated reactor components. More detailed evaluations of physical characteristics may also be targeted in order to refine dismantling plans and sentencing options. A determination of the nature of the contaminants' distribution may be required (such as levels being uniform and homogenous or intermittent and localised) in order to plan sentencing and segregation of materials and ensure that proper monitoring and characterisation meets required statistical confidence levels.

In addition, confirmatory characterisation surveys, sampling and assays can be conducted during and after the dismantling/remediation process to ensure that workers, the public and the environment are adequately protected and to verify that final sentencing, storage and transportation planning are correct and were properly conducted. These surveys and sampling can be performed on the material removed to confirm fingerprints and monitoring and assay plans for material sentencing and waste classification. They can also include confirmatory surveys to verify that the relevant materials have been removed and that further remediation is not required to meet licence termination or clearance criteria. These are also critical surveys since it is costly to demobilise dismantling and remediation resources only to find that further remediation is required after the materials are packaged or upon performance of the final status survey.

Some challenges and possible technical innovations that are potentially relevant to waste classification and categorisation were identified, including:

- statistical and calculation methods for modelling, including validation of methods (e.g. in relation to representativeness; grid density; number of samples; where; point samples/heterogeneity within the grid; defining an acceptable level of uncertainty) and the efficiency and accuracy of non-destructive testing (NDT) methods (for example, see the discussion on the Kola experience in Chapter 1);
- correlation between contamination measurements from sampling and calculated values from dose rate measurements (gamma emitters) and scaling factors (beta and gamma emitters), including piping; concrete and depth of intrusion of contamination into the concrete, graphite (including alpha contamination);
- modelling the movement of highly mobile nuclides (e.g. tritium);
- measuring the activity of hard-to-detect pure beta and alpha emitters;
- correlation of key radionuclide ratios and scaling factors (between easy-to-measure and hard-to-measure nuclides), different solubility of scaling radionuclides such as Cs-137, Co-60, Am-241, and more difficult-to-measure radionuclides such as H-3, C-14, Cl-36, Ni-59, Ni-63, Se-79, Sr-90, Np-237, Pu-239/240 and Cm-242/243;
- estimating levels of impurities in metals, concrete, etc. for recycling and reuse;
- development of remote and non-destructive techniques for rapid characterisation of contaminants to allow segregation and/or changes in the classification of waste; in situ (rather than off-site) measurements, e.g. use of mobile laboratories;
- characterisation in and around difficult to access structures (e.g. drains).

Experience from a variety of case descriptions shows that clearly defined waste management and disposal routes should be supported by an appropriately defined waste classification and categorisation scheme. Such a tool is a very important aid to planning remediation and waste management activities in a manner that does not create future problems. Conformity with international approaches, such as the IAEA safety principles (2006a), may also engender confidence at the national level.

Radioactive waste arising following accidents may have characteristics not otherwise expected. It can be abnormal, e.g. waste from TMI-2 contained a high concentration of fission products and/or small quantities of fuel materials. Special consideration may also be needed for damaged fuel, as indicated for the ChNPP, Windscale Pile, TMI-2 and at the site for temporary storage at Andreeva Bay. Fingerprint techniques used in normal situations may not be applicable to accident waste.

Early interaction of operators with regulators is regarded as beneficial. Co-ordination among all relevant regulatory bodies is also important, for example bearing in mind that for VLLW and LLW it is possible that hazards other than radiological hazards may be present and be dominant.

Decommissioning is closely linked to waste disposal. Early decisions on remediation, without due consideration to final disposal, can make final disposal more difficult. Waste packaging arrangements in the UK address this issue, even in the absence of a site-specific design/plan for disposal of waste requiring geological disposal.

Applying normal regulatory requirements and procedures is preferable, as far as possible, so as to most effectively use existing equipment and procedures. So a key question to consider is, "What characteristics of the accident waste are such that they cannot fit into normal requirements and procedures"?

In the case of Fukushima Daiichi accident, the NRA only began to develop the mid- to long-term safety management policy on radioactive waste generated from Fukushima Daiichi accident and its recovery activities on-site from the end of 2015. NRA and TEPCO have been discussing waste management at Fukushima Daiichi, e.g. waste storage, volume reduction, inventorying of waste, and characterisation of waste. Regarding the waste classification and categorisation for accidental waste, existing requirements and procedure will be a starting point for discussion.

As usual, the classification and characterisation of radioactive waste have been defined not only by the radiological characterisation of waste but also by the disposal concepts and its safety assessments. In Japan, a project to identify the waste classifications for accident waste has just started based on the safety assessment of the disposal options which were normally considered before the accident. It is not certain whether existing Japanese requirements and procedures on waste classification and categorisation may be applied to the accidental waste. To find a solution to this issue, research organisations have been implementing basic analyses of radiological characterisations of accident waste.

Experience shows that it can be useful to recognise that some degree of iteration needs to be expected, but that a convenient starting point is the existing safety objectives and the existing waste package specifications coupled with preliminary waste characterisation data. Thereafter, a data quality objectives (DQO) process could be used as a strategic planning approach to plan further characterisation activities, which are in turn to be used to support the next phase of iteration. The process provides a systematic procedure to define the objectives of a characterisation programme. The characterisation programme objectives developed via this process are qualitative and quantitative statements that:

- clarify the characterisation objective;
- define the most appropriate type of characterisation measurements to make, the number of measurements to make, and the most appropriate measurement methods to use;
- determine the most appropriate locations and times to make characterisation measurements;

- specify the level of measurement uncertainty that is allowable to support the decisions that are made based on the characterisation measurements.

It can be generally understood that non-specialists stakeholders may have an interesting contribution to the technical definition of a waste classification and categorisation scheme. Stakeholder support is important and implies improved engagement with all those involved, as discussed in the context of Fukushima Daiichi NPP remediation in Kohzaki et al (2015). This in turn suggests that apart from the wide range of technical aspects of the design of such a scheme, it could be useful to include a component which helps convey a broad understanding of the level of hazard associated with different classes of waste.

6.4. Recommendations on waste classification and categorisation development for decommissioning and waste management

High-level international guidance on radioactive waste classification generally, and in the context of decommissioning of nuclear facilities, is available. This guidance is recommended as a suitable starting point for any national specific classification scheme.

Experience in other countries shows that national schemes need to allow for nationally relevant factors. Furthermore, more detailed specific features are needed to address the complex circumstances of abnormal conditions arising, for example, after a major accident. It is recommended that a classification scheme be developed specifically to address the abnormal features of accident waste.

Any scheme developed specifically to address decommissioning and waste management for accident waste should follow as far as appropriate international practice and take account of lessons learnt at other abnormal sites. The scheme should incorporate all waste arising and include classification of the waste which does not need to be managed or regulated as radioactive waste. The scheme should also accommodate or account for other hazardous features, so as to avoid planning, regulatory and safety management contradictions.

Ideally, the classification scheme adopted should support all aspects of management in a holistic manner, while leading to and not foreclosing on options for final disposal. This means allowing for:

- protection of workers involved in the most radiation-hazardous operations including application of optimisation (e.g. materials requiring remote handling distinguished from those which can be handled directly);
- protection of the public and the environment (e.g. effluents which can be discharged directly as distinguished from those which require treatment before release to the environment);
- emergency preparedness and response during dismantling and remediation operations (materials whose management requires special safety analysis, e.g. criticality assessment, as distinguished from those which do not);
- materials which can be managed without the need to consider radiological protection issues, and implications for possible end-states for the contaminated areas;
- waste types which meet waste acceptance criteria for packages and materials due to be stored, taking account of possible transport and options for subsequent disposal;
- a basis for designating materials containing fuel fragments or debris as radioactive waste rather than spent fuel;

- non-radiological hazards associated with materials (including the basis for analysis of the major hazard, radiological or other, which should determine the management method).

A single scheme which works effectively to address all these would be complicated. A possible solution could be to include sub-categories for each class of waste, e.g. remote handling only LLW.

Devising an effective classification and categorisation scheme should be considered as an iterative process, which takes account of regulatory developments and new information about the waste, as it arises. The iterations could be timed in line with key steps in the overall strategy (NDF, 2015), but typically could involve the following steps:

- a) collation of currently available preliminary waste characterisation data;
- b) assumption of normal waste management arrangements, including safety requirements, disposal routes and waste classification schemes;
- c) identification of potential technically feasible management options;
- d) safety analysis of each option to identify preferred and provisionally safe options, and to identify priority further information needs, particularly as regards the most hazardous waste and/or the most poorly characterised waste;
- e) further iteration of characterisation to provide more waste data, particularly for the priorities identified at d);
- f) further iteration, including more detailed description of safety requirements, disposal routes and waste classification schemes.

Engagement with regulators is recommended to be included at each step. Note that at step b) it is suggested that normal arrangements are used initially, so as to take advantage of existing techniques and procedures, and local experience in their application. As discussed above, it is likely that variations will be needed. The variants should be fully discussed in an open and transparent manner with relevant stakeholders.

The scheme for waste generated from on-site work is recommended to be consistent with any scheme used for managing waste generated in off-site remediation work.

Based on the currently available preliminary waste characterisation data, it is recommended that the sampling programme be expanded significantly. Priority should be given to the identified most hazardous waste, as identified in step d) above (see also Table 3.2 of NDF [2015]). The use of modern software tools is recommended to support efficient identification of priority waste sampling points.

In some cases gross beta/gamma, and or gross alpha measurements may be adequate, and/or dose rate measurements. In other cases, radionuclide-specific measurements may be necessary. It is noted that the radionuclides which typically dominate short-term safety and pre-disposal operations are relatively easy to measure, such as Co-60 and Cs-137. In contrast, radionuclides which dominate safety demonstration in the context of waste disposal are long-lived and low-energy emitters which are not easy to detect, such as C-14 and I-129 identified in sub-section 5.4. Other examples have been identified in wide ranging research (NEA, 2009; Keesmann et al., 2011).

It may be possible to develop fingerprint approaches to facilitate waste characterisation, but the fingerprints appropriate for characterising waste from Fukushima Daiichi NPP accident are likely to be specific to this case, such as the dose rate to Cs-137 in rubble concentration (see Figure 5.6).

Waste characterisation activities supporting the development of a waste classification scheme and other related activities are recommended to be carried out within an integrated and structured programme similar to the DQO system mentioned

above. Such a programme is recommended to be designed to address logical sequences of questions which resolve decisions on the priorities identified at step d) above.

Most waste classification schemes developed in the past were devised to address technical issues and were necessarily expressed in technical terms. They do not readily indicate the scale of hazard associated with the waste and therefore do not help explain to non-specialist stakeholders the significance of the hazards. It is suggested that the classification scheme include a component which relates to the hazard of each class of waste to another readily understandable or commonly encountered hazard. Ideally, this would include consideration of chemical as well as radiological hazards within a single coherent and proportionate approach, as discussed in the NRPA report on this subject (2015).

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7. Waste conditioning, decontamination and reduction

7.1. General description

In the case of nuclear power plants, radioactive waste resulting from an accident differs from waste generated during normal operation, especially waste from on-site decontamination activities, from management of contaminated water or from decommissioning work, including fuel debris.

Waste with high levels of contamination and radiation is generated in large quantities and thus needs adequate treatment (decontamination, reduction or conditioning) before intermediate storage in dedicated areas.

Two major steps can be differentiated in waste management:

- storage (temporary, by definition), which consists in orderly placing waste or spent fuel in a safe place with the intent of retrieving it at a later date to provide it with a more permanent future. Following this step, part of the occurring radionuclides may be transmuted to reduce potential radiotoxicity in ultimate waste.
- disposal (final, although possibly designed to be reversible during a given period of time), which consists in placing ultimate waste in a deep geological formation so as to protect it from the natural and human disturbance until radioactive decay has reduced the hazard to acceptable levels. Waste is acceptable for final disposal only if it consists of a solid, non-dispersible block which does not contain water liable to leak out. Packages should be easy to handle and shock resistant. They should ensure durable confinement of the waste and exhibit satisfactory resistance to leaching by water.

Waste conditioning has to be compatible with these two complementary, non-exclusive major steps, which will take place in succession. Consistently, easy handling of conditioned waste packages is required, in compliance with safety and radiological protection rules. This involves the possibility of retrieving these packages under constantly safe conditions at the end of the storage period. In addition, the conditioning material selected must show a suitable behaviour in the long term, in accordance with the future disposal step. In this very context, the aggressive component is underground water, which will inevitably come into contact with the material in the geological environment after a period of variable duration.

The package is the first of the successive barriers separating radioactive elements from the environment, giving guarantees that radioactive elements will not be disseminated. It also complies with standards for transportation, storage or disposal. Consequently, a long-term management of packages implies evaluating the quality of this barrier as time elapses.

Conditioning thus includes all the successive operations required for manufacturing this package.

Generally, radioactive waste is conditioned at the site where it has been generated. Short-lived low- or intermediate-level waste can be transferred to a dedicated disposal facility. Long-lived waste is generally kept at the production site in specific storage facilities adapted to the package type.

It is difficult to ensure disposability of higher-level waste when the disposal concept is still at a generic stage and future decommissioning plans have not been decided. (This could be achieved through disposability assessments in order to avoid re-conditioning.)

Decontamination

Given the radioactive materials involved (waste contaminated by reactor cooling water, decontamination rubbles, etc.) treating this waste requires a specific, adapted methodology. The strategy adopted in most applications is decontaminating waste prior to considering its future. The decontamination operation is a preliminary step which consists of removing as much radioactivity as possible from the technological waste considered, thereby making the waste “cleaner” and allowing for easier management. In most cases, radioactive contamination is located at the surface of solid waste, making it possible to collect it by various “washing” operations.

The main challenge of radioactive decontamination is to perform such an operation while generating a minimal quantity of secondary waste or effluents. For every process, minimising the quantity of waste generated requires taking into account a number of parameters in the process development: the nature of contamination, accessibility conditions, temperature, moistness, the nature of the material to be processed, etc. For all these reasons, it is difficult to use a universal process. For each decontamination operation, it is necessary to reflect on a process adapted to the operation. In practical terms, the decontamination of solid pieces of equipment can be achieved with imbibed pads or lyes. Such rustic processes, however, produce significant volumes of waste, and thus it may be interesting to consider more recent processes. For example, decontaminating solids with foams allows the amount of secondary effluents to be divided by a factor 10.

Surface decontamination with gels, when feasible, makes it possible to consider dry treatment (Faure et al., 2001) which results in solid by-products, easy to be conditioned. In fine, the decontaminated solid will show low residual contamination, and, from the technico-economical viewpoint, it will be easier to immobilise it.

Concerning contaminated aqueous effluents, the technical challenge of decontamination is optimising radionuclides precipitation using the most selective reagents with the lowest concentrations as possible. As a consequence, the amount of resulting sludge is minimised, which optimises the final volume to be immobilised. New decontamination processes with customised reagents are also developed and implemented through physico-chemical techniques to reduce downtime and improve effluent treatment facilities, e.g. decrease radiological and chemical releases (Barre et al., 2015), in particular for caesium and strontium (Villard et al., 2015).

In terms of contaminated soils, interesting processes are underway to concentrate contamination, mainly caesium and strontium (e.g. flotation for soils with clay).

Concerning organic waste treatment, the challenge is quite different. One possible process, particularly well adapted to organic effluents and waste, lies in drastically reducing their volume by incineration, which concentrates the contamination within a mineral ash easy to condition. A new process for the incineration of radioactive solvents containing chlorine or Fluor by plasma under water is also under way in industry (Lemont et al., 2014). The use of plasma under water should simplify dust treatment and prevent corrosion of the facilities.

Waste volume reduction

Case of high-activity ion-exchange resins

Following the waste characterisation described in Chapter 6, in either characterisation scheme when dealing with high-activity radioactive resins, applying this process

converts the initial resin waste to a lower-activity waste class by moving it to the left. The majority of the radioactive burden from the initial high-activity radioactive resin is shifted to a smaller volume in a higher-activity waste class to the right. During reactor power operation, ion-exchange resins are used on-site for a number of purposes, e.g. reactor water clean-up, fuel pool clean-up and condensate polishing. These operations give rise to the accumulation of spent ion-exchange resin, some of which is class B/C in nature.

Initial work with the remaining cation resin demonstrated it was possible in principle to regenerate the resin chemically with acid, and remove the metals driving the waste class, thus rendering the original cation resin class A waste. The resulting acid waste was neutralised, producing a precipitant, which matched the class C waste category.

■ **Conclusion:**

- A survey of class B spent resin waste shows the majority (over 80%) to be from a mixed bed origin for both pressurised water reactor (PWR) and boiling water reactor (BWR) facilities.
- Comparison with 10 CFR Part 61 regulations shows isotopes of Ni-63 and Cs-137 drive the waste classification in PWR waste resin, and isotopes Sr-90 and Cs-137 in boiling water reactor resins.
- Technologies have been identified to perform resin separation, including chemical regeneration and waste recycling.
- A first review of the economies of the process indicates that a differential of at least USD 1 000 /ft³ (USD 35 000/m³) must exist between class A and B waste disposal rates for the process to be viable.
- Further work has been identified to underpin the technologies suggesting the work cover three specific areas of the development programme for volume reduction methods and waste from changes in high-activity spent resin, including:
 - Treatment of the original radioactive resin with acid regeneration.
 - The production of a solid waste form with a substantially smaller volume containing the majority of the radioactivity from the original resin.
 - The effect on the process of the presence of powdered resins within the original waste resin. The following conclusions can be made with regard to the process development.
 - Chemical regeneration of radioactive resins show that the radioactivity can be removed to below the level required to reclassify the original resin as class A waste.
 - A sequence of treatments following the chemical regeneration allows the secondary radioactive waste to be produced as a substantially smaller volume of solid form class C waste, and allows the water to be recycled for the next chemical regeneration.
 - The production of the solid waste precipitate calcium sulphate (CaSO₄) has been shown to be a robust and reproducible process.
 - The precipitation of the metal ions from the regenerated resin has been shown to coincide with the production of calcium sulphate, co-precipitation, in the correct conditions. This includes the radioactive species with the exception of caesium.

- The mixture of resin types, including cation and anion form bead resins and cation and anion form powdered resins, has been shown to separate using water flow.
- Initial tests with powdered resins show that their physical behaviour is unaffected by low concentrations of surfactants.
- This work demonstrates that the volume reduction methods and waste form changes for high-activity spent resin are viable for full-scale development.
- Further scale up work has been identified for the continuation of a development programme.

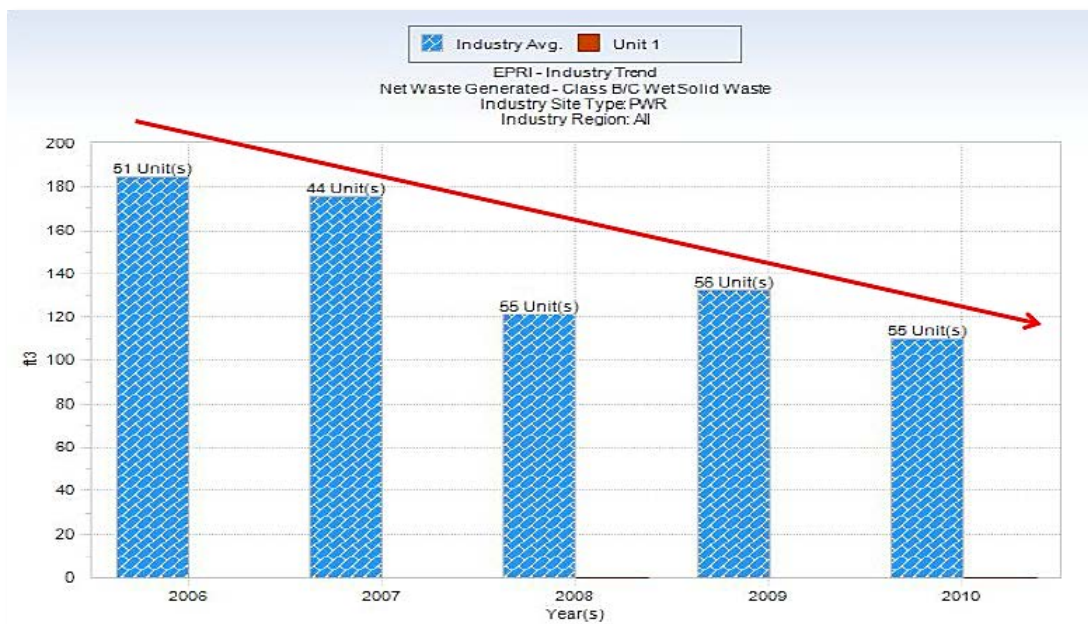
Waste is classified in accordance with US federal regulations (7) into four fundamental categories, A, B, C or greater than class C (GTCC) (Table 1.1).

Three major areas of focus including guidance for on-site storage of waste, increased disposal flexibility through regulation and guidance changes, and the focus on this project, class B/C waste reduction related to processing media including filter elements, resin, and flowable filtration and ion-exchange media (pre-coat).

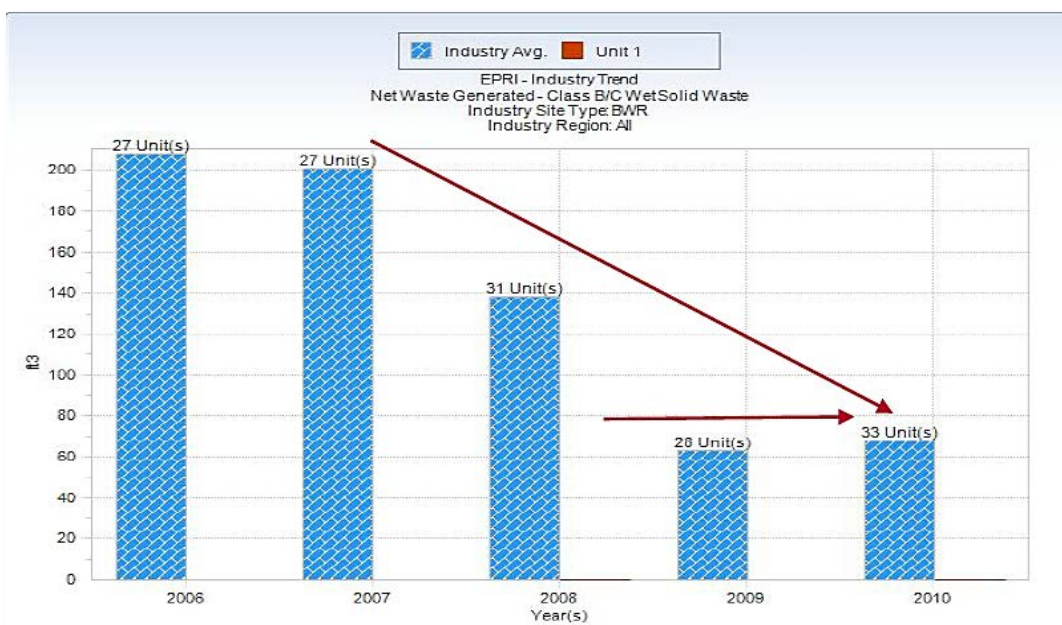
Several proven and potential options for managing processing media result intentionally, or otherwise, in reductions to the generated and disposed volume of B/C waste. The combined effect of current waste disposition costs and on-site storage are significant enough that all options warrant consideration.

Many power plants have, or are considering implementing one or more of those strategies. The significant success of the industry-wide class B/C waste reduction effort is clearly illustrated in Figures 7.1 and 7.2.

Figure 7.1. PWR class B/C generation 2006-2010



Source: EPRI RadBench.

Figure 7.2. Boiling water reactor class B/C waste generation 2006-2010

Source: EPRI RadBench.

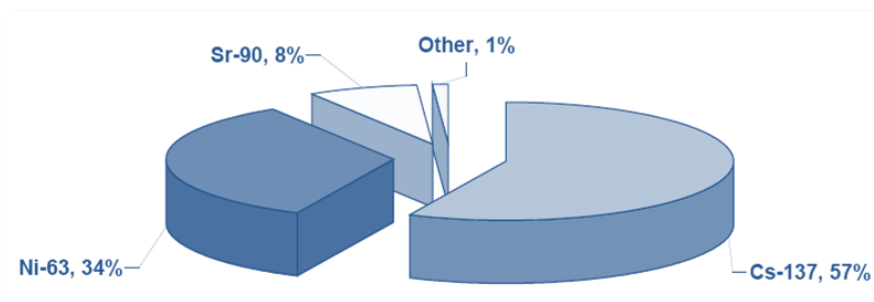
In spite of this success, opportunities remain. Several of the strategies contained herein are related to processes that can directly affect chemical parameters and therefore may require additional evaluation relative to site-specific chemistry or source term programme objectives.

This class B/C waste reduction update specifically targets the inclusion of chemistry programme considerations and experience, and provides chemistry data (when available) for several of the strategies. Implementation of one or more of the alternative media management options requires an in-depth process evaluation and development of concise implementation plans to ensure the targeted waste-driven initiatives do not conflict with chemistry programme performance. The industry operating experience indicates that this approach can produce significant reductions in class B/C waste volumes and equally significant cost savings.

These strategies may be directly applicable to any reactor that is considering reductions to:

- process media generation volumes;
- stored or disposed waste volumes;
- spent media activity levels.

In addition to media based reduction strategies, source term reduction programmes and measures have a direct effect on the volume of class B/C waste that is generated. Fuel integrity, materiel selection, and chemistry and operating regimes will all effect the concentration of activity and radionuclide distribution that is available for removal by purification media (5, 6). The following figure illustrates the principle nuclides that drive waste from class A to class B/C or greater.

Figure 7.3. Class BC nuclides of interest

Source: 2003-2010 industry waste classification data.

Strategies for reducing the generation of class B/C waste

Ten plant processes affect the volume of B/C waste that requires disposition:

1) Primary ion exchanger (chemical and volume control system – CVCS) – Online lithiation

Online lithiation is a lithium management option that is typical of CE- and B&W-type PWRs and is used at only a few Westinghouse plants. This option is implemented by loading two mixed beds in parallel. One bed serves as a de-lithiator for a cycle. The other bed serves as the reactor coolant system purification bed having been lithiated in the previous cycle. In this configuration, the mixed beds' resin can serve for two cycles performing a different function in each. This option is easily implemented at those plants that have the bed volume and piping configuration to support multiple bed media management. Otherwise, system modifications would be required. This option also requires a significant commitment from both the chemistry and operations organisations to ensure the beds are aligned in the proper sequence during the pertinent period. This practice can reduce CVCS cation resin consumption and the associated lithium management costs. Duke Energy, and other utilities, employ this strategy **specifically targeting resin volume reduction**.

2) Reactor water clean-up (RWCU) in service run length

Most boiling water reactors have historically worked to increase the run length for their RWCU filter demineralisers. While this may reduce the total volume of generated media, the increased run length can result in an increase in the spent media activity and therefore increase the volume of generated class B/C waste. This volume reduction process is the inverse of that strategy, shortening run lengths to reduce the generated volume of class B/C waste. The Susquehanna plant performed a plant-specific analysis and determined that the run length reduction strategy was cost effective and reduced the final disposed or stored waste volume.

3) In-service media management – spent fuel pool

The majority of PWRs operate their spent fuel pool purification system continuously per original design considerations. The media selection process for this system typically addresses maintaining chemistry and activity in specifications without regard to waste classification. The option evaluated included implementation of a custom ion-exchange load, and with that in place, using the system only as needed for chemistry or activity control, as opposed to the historical full time service runs.

4) Cation and anion media – point of generation separation and blend ratio media separation

Alternate vessel configurations include segregating resin functional types (anion and cation) by vessel versus using traditional mixed bed strategies, modifying the cation to anion ratio in mixed beds, or modifying the in-service operating sequence. These options can result in improved throughput (gallons processed per media volume), improved effluent quality and/or ultimately reduced waste volumes. This option requires careful evaluation of chemistry influent and effluent characteristics for each vessel as they relate to the manufacturer's expected media performance.

5) Ion-exchange vessel short loading

Short loading involves using media volumes in ion-exchange vessels that are less than those in the original design. This requires evaluation of the vessel design and media load to ensure the media performance will not be affected by system flow rates and pressure and that it will not impact spent media removal options. This strategy is very effective, in some cases resulting in media reductions in excess of 50% without sacrificing performance.

6) Media segregation in spent resin tanks and waste containers

This option involves the use of dual spent resin tanks and/or one or more waste containers to segregate spent resin following generation. This applies to segregating high- and lower-activity beds of all types and segregating cation and anion, if possible. This strategy reduces the potential for increasing class A waste to class B/C waste as a result of commingling the waste with higher-activity waste streams; therefore the total volume of class B/C waste is reduced.

7) Post-generation segregation of cartridge filters

This option involves the use of multiple waste containers and/or filter vaults to segregate spent filters following generation. This applies to segregating high- and lower-activity waste streams and reduces the potential for increasing class A waste to class B/C waste following container loading and classification; therefore the total volume of class B/C waste is reduced. This option is applicable to any reactor that generates cartridge filters that meet and exceed class A waste limits. It requires physical floor space and/or shielded areas/modules to support staging multiple waste containers. It can result in significant cost and storage volume savings and has no impact.

8) Cartridge filter dose rate and activity management

This option considers the use of remote radiation monitoring equipment to provide live dose rate data for filters. That information is used to complete an estimated waste classification calculation and develop dose rate based values for removing filters from service. This is a relatively easy strategy to implement and relies solely on existing manual or remote radiation dose rate monitoring equipment and waste classification software. It typically will increase the total number of filters generated. However, in most instances the generated filters are class A waste that can be packaged more efficiently in larger volume liners that currently can be disposed of versus stored.

9) Cartridge filter reduction using alternate ion-exchange media

This option involves the use of alternative macroporous "filtration" resin in ion exchangers that results in improved removal of insoluble species. The improved removal efficiency reduces the particulate challenge to downstream filters. This in turn reduces the rate of increase for filter activity and/or dP, and reduces the subsequent volume of filter waste. This reduces both class A and BC filters and if adopted in conjunction with the previously discussed segregation and dose rate management strategies would significantly reduce generated waste volumes and eliminate generation and/or storage of class B/C filter elements.

10) Spent resin classification options

The fundamental process for waste classification involves obtaining representative waste stream data/samples, analysing the activity content, and scaling that data to a full waste container volume. Some difficult-to-detect nuclides also require scaling from identified nuclides. The collective activity and volume data are entered in an industry approved software program to generate a final waste package classification. Variations in sampling techniques and technologies can result in creating a waste package with a higher (or lower) waste classification, affecting cost, transport, volume reduction and disposal/storage options. Understanding and carefully evaluating waste management strategies and options helps to ensure that waste is accurately segregated, analysed, and ultimately categorised by waste class.

Advanced volume reduction and waste segregation strategy for low-level waste

■ Conclusions related to conversion reforming

The advanced technology evaluated in this study was “conversion reforming,” a technology which has been developed by Studsvik-USA, Inc. This is a pyrolysis process essentially identical to the steam reforming process commonly used for volume reduction of spent resin. The primary difference for this study was the use of smaller equipment and the application of the technology for filter waste. The study was further supported by OREX Technologies and Framatome ANP, who provided the filter cartridges used to evaluate the conversion reforming technology. The following conclusions apply:

- The study demonstrated that conversion reforming is a viable and highly efficient volume reduction technology for nuclear plant spent filter cartridges. It is limited to non-metal filters and filters which are not made primarily of fibreglass. Although non-metal filters are not widely used in commercial nuclear plants, a wide range of such filters are available to replace existing metal-reinforced filters.
- Conversion reforming offers an exceptionally high volume reduction efficiency for filter waste. In this study, the net disposal volume reduction was 54:1. Even if it was only 10:1, conversion reforming would produce very substantial benefits to the nuclear industry. This exceptionally high volume reduction efficiency translates to a very substantial reduction in disposed waste volumes, as well as reducing stored low-level waste (LLW) volumes for plants which do not have access to a disposal facility. If an existing plant or an advanced light-water reactor were forced into long-term on-site storage, application of this technology would reduce stored reformed filter waste to only one or two containers over the entire life of plant.
- If the nuclear industry broadly embraced non-metal filters and conversion reforming technology, industry-wide cost savings over the next 25 years would reach millions of dollars.

DAW and mixed LLW processing and waste volume reduction

This chapter describes dry active waste (DAW) and mixed waste (MW) treatment technologies commercially available to the commercial nuclear power industry. The chapter also identifies major DAW and MW treatment facilities available. Brief descriptions are provided for each available technology, and a brief overview addresses the capabilities of each waste treatment facility.

■ Steam reforming

For DAW, capital investment in new technologies or for expanded use of existing technologies focuses on high-activity waste, such as resin processing. In 1999, a new steam reforming process went online to compete for volume reduction of spent resin up

to 100 R/hr. As experience is gained with this equipment, it is likely that it will compete for other plant waste streams, such as charcoal beds, oil and other organic media.

- High-activity DAW

Until recently, high-activity DAW (>1 R/hr) was usually shipped directly for disposal. Increased competition has encouraged volume reduction facilities to accept high-activity DAW for super-compaction. This has had a very significant impact in terms of reduced disposal volumes and lower waste management costs.

- Overfill

The use of grit blast media, soil, dirt, and small diameter rubble as overfill for packaged waste has increased significantly since 1997. This increased attention occurred as a side effect of the disposal structure, which scaled waste disposal fees based on varying waste densities. In many situations, using overfill material to fill void spaces resulted in higher waste densities and lower overall disposal costs for the same waste package. This had the effect of disposing of the overfill waste at no cost.

- Glassification

The application of glassification technology as a prime competitor to DAW incineration has encouraged many waste generators to evaluate potential cost savings while realising similar volume reduction efficiencies. New and better glassification equipment currently being installed to handle MW is expected to provide increased competition in that arena as well.

- The Electric Power Research Institute's DFD process

The Electric Power Research Institute's (EPRI) decontamination for decommissioning (DFD) dilute chemical process has been expanded to address large, individual plant components, thereby expanding on its success in decontaminating in-plant reactor coolant and reactor water clean-up systems in both operating reactors and for plant decommissioning applications. The DFD process has been demonstrated successfully on more than a dozen heat exchangers, as well as shroud head bolts and control rod drives. Two commercial waste treatment facilities are now licensed for the process to handle large components, such as steam generators and pressurisers. This fairly recent but proven technology has opened the door to volume reduction and treatment possibilities for many large components formerly considered as not practical to decontaminate.

- Drum super-compaction

Some inefficient waste management technologies continue to be used long after they cease to be cost effective. This has been the case with super-compaction of pre-compacted drums. Over the last few years, this once-dominant volume reduction technology has finally given way to the much more cost efficient bulk super-compaction and combustion technologies.

- MW treatment

The cradle-to-grave cost of managing MW remains 20 to 30 times higher than for DAW. This has created considerable competition among the few MW treatment facilities. MW treatment facilities are evaluating other treatment technologies which have proven to be successful for hazardous waste and which could be applied to MW as well. New treatment facilities are also expected to come online in the next year to provide state-of-the-art treatment technologies with higher throughput capacities. Expanded technologies include glassification, thermal desorption, mercury amalgamation, MW compaction, and liquid solvent extraction.

How to identify the optimum technology

Although cost always plays a significant role in technology selection, it is not the only significant consideration, and it is often not the most important consideration. As part of this technology review, each treatment facility was asked to identify what the waste generator should be asking themselves when they evaluate existing, alternative or emerging technologies.

- What is the typical volume reduction efficiency and range of efficiencies for any given waste type?

For example, bulk super-compaction ranges from a volume reduction low of 5:1 to a high of around 12:1, with 8:1 being the most common for DAW. Variations are more often due to the initial, as-generated waste density than due to variations among vendor equipment. Similarly, incineration of plastic and paper results in a typical volume reduction of 100:1, whereas incineration of resin produces a typical volume reduction of around 8:1.

- What alternative technologies can be applied to the same waste type?

Waste managers tend to look at identical technologies for managing waste based on historical experience. For example, combustible LLW is shipped for incineration, with the expectation that it will result in the greatest volume reduction. Today, glassification and steam reforming offer alternatives for DAW and resin, respectively. Similarly, steam reforming is emerging as a competing technology for charcoal beds and oily waste.

- What determines whether a mixed waste can be sent to a specific treatment, storage, or disposal (TSD) facility?
- What is the available storage capacity at the vendor facility, and what percentage of that storage is under cover?

Storage capacity arose as a significant consideration at most treatment facilities, and it was most significant for MW facilities. At least one DAW treatment facility relied 100% on outside, uncovered storage, and a few others were limited in terms of waste receipt due to storage restrictions. Resin processing stored waste capacity is of particular importance, as it forces utilities to either sit on a waiting list to ship waste or look for alternative waste treatment. Waste stored at MW facilities is currently limited to 364 days. In addition, all MW should be stored under cover. Although most of the MW treatment facilities had covered storage, that storage was not of unlimited capacity.

- How soon will waste be processed and disposed of?

All waste shipped to a commercial waste treatment facility should be processed and shipped for disposal within 180 days. Some utilities contract for earlier disposal, particularly for mixed waste. These times restriction should be incorporated into the waste treatment contracts. All waste generators should review their monthly vendor reports and identify any waste which has been at the treatment facility for longer than 180 days and contact the facility operator to expedite treatment and disposal. It also is a good practice to require MW treatment facilities to send letters certifying that each waste container was treated, the technology used to treat the waste, and that the final waste form and specified waste volume was disposed of at a given disposal facility on a given date. (This practice is already in place for some MW treatment facilities.)

- What secondary waste is generated in what quantities, and who is responsible for that waste?

Most waste treatment technologies result in secondary waste, even if this waste is in the form of protective clothing or housekeeping materials used by facility workers. Other secondary waste from glassification processes, ash from any combustion process or

steam reforming process, bag house waste or high-efficiency particulate air (HEPA) filters generated during treatment, and any other waste that is a by-product of the treatment process. The waste treatment contract should specify whether the cost of managing this waste will be passed on to the waste generator. It should be noted that some treatment facilities pick up the cost for secondary waste, but the disposal volume is credited to the waste generator (a common approach for metal waste).

- What is my long-term liability associated with the disposed final waste form?

This question really is asking “Who takes title to the waste?” For example, all super-compacted waste is credited to the waste generator. By contrast, the slag from a metal melt process is the responsibility of the treatment facility operator. Although the answer to the long-term liability question is reasonably well known, it is far less certain for MW. This is one of the key reasons why it is important to obtain letters certifying that each waste container was treated, the technology used to treat the waste, and that the final waste form and specified waste volume was disposed of at a given disposal facility on a given date.

Conditioning matrix

Cementation

Operational waste is most often cemented. In the case of solid waste, it is placed in a metal or concrete container into which cement is poured: this is the so-called “cement-immobilised waste”. As regards liquid waste, it is used in cement manufacturing as a mixing liquid prior to cement pouring into a metal or concrete container. Several container models are available, adapted to the form and size of the waste to be conditioned.

In relation to waste confinement, cementitious matrices display a number of assets which counterbalance the drawbacks associated with the significant volume of this conditioning type:

- versatility (ability to confine a number of physico-chemical waste forms);
 - low cost, easiness of implementation;
 - good mechanical resistance;
 - insolubilisation of a high number of radionuclides owing to the interstitial solution basicity.
- Cementitious matrices thus rank as reference materials for low- and intermediate-level waste conditioning

Cement-based materials are widely used in radioactive waste conditioning: grouts for waste embedding, mortars for immobilisation operations (immobilisation of bulk waste in a container, immobilisation of a primary container in a secondary container), and concretes for container or structure elements manufacturing on disposal sites. Given the specific nature of the issues raised by cement/waste interactions and the timescales to be considered, a new approach of cementitious materials has emerged in which physico-chemistry has a predominant role and provides the data required for modelling the processes involved.

They result from the setting of a mixture of anhydrous cement, aggregates of various sizes, and water. Several categories may be distinguished depending on whether aggregates are present or not, their size, and the water/cement ratio:

- **pure pastes**, only consisting of cement and water;

- **grouts**, pure pastes or fine mortars with a low sand content and a high quantity of water (water volume > cement volume), which gives them the relevant rheology for the pouring that follows mixing;
- **mortars**, which contain aggregates (sand) under 6.3 mm in size (generally, the sand volume is higher than the cement volume, which is itself higher than the water volume);
- **concretes**, which, in addition to sand, include aggregates of a size between 6.3 and 80 mm. In order to increase their tensile strength, they may be reinforced with bars of short metal fibres (fibre-reinforced concrete).

Grouts are mainly used as waste embedding matrices. Mortars are used for immobilisation operations (immobilisation of bulky waste in a container, immobilisation of a primary container in a secondary container). Last but not least, concretes are used for container manufacturing and for the making of structural components on disposal sites.

Nature-related diversity: Waste is under such forms as aqueous solutions, suspensions (chemical co-precipitation sludge), or bulky or powdered solids. “Homogeneous” waste is intimately mixed with the cementitious binder (embedding) and are potentially reactive (evaporator concentrates, sludges, small-grain-sized powdered solids). “Heterogeneous” waste, more bulky and non-reactive in cementitious media, is subjected to a mere mechanical immobilisation (plastic-material objects, rubble, some metallic waste, etc.).

Composition-related diversity: The composition of the waste to be immobilised in cement depends on the activity it arises from, as well as the treatment and decontamination processes implemented upstream the conditioning stage. The contaminated aqueous waste volume can be thus reduced by evaporation or chemical co-precipitation (in order to insolubilise the radioelements), as well as filtration. Besides, the composition of a given waste may vary significantly. Such a diversity in waste implies diversity in embedding materials, based upon tailored formulations. The latter have to take into account both the constraints of the implementation industrial process and the specifications for further package disposal.

- Specifications inherent to the implementation

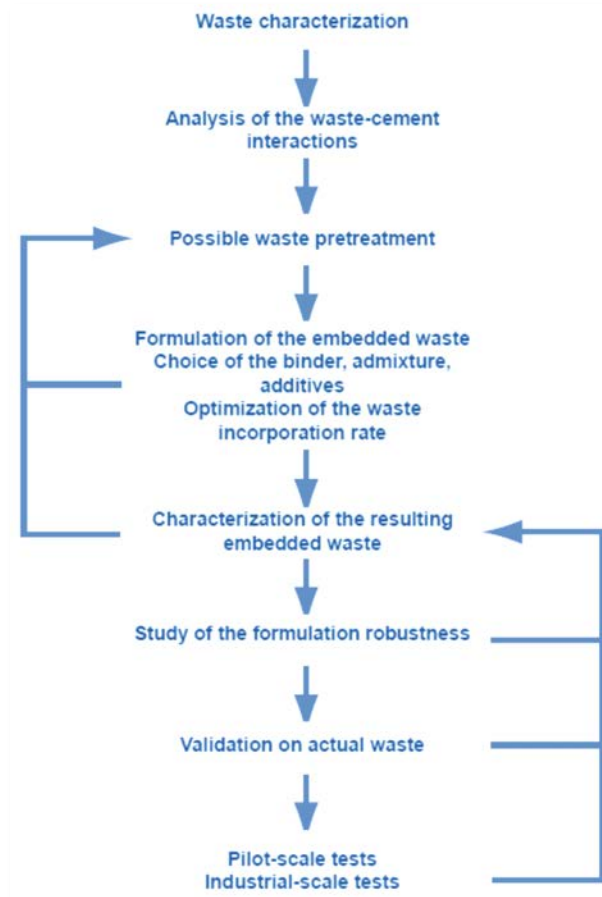
Cementation conditioning is generally performed near the waste-producing sites. The embedded waste preparation is carried out either in the container itself, using lost or retrieved impeller blade stirring, or in a separate mixer, prior to pouring the mixture into the container.

The **rheology** of the embedded waste after mixing is an important criterion for assessing the quality of a formulation, especially when the latter is used in a mixer.

- Defining an embedding material formulation

Figure 7.4 gives an overview of how an embedding formulation is developed, with the various steps. Multiple parameters are involved in developing the embedding material formulation and investigating its robustness. Hence the use of experimentation plans to help define efficient experimental strategies. Only the more informative experiments with respect to the aims fixed are achieved in the research area. The number and cost of the tests are therefore reduced. Operational models are built in order to predict the embedded waste properties as a function of the formulation parameters or the waste composition. They may eventually be used to perform a multi-criteria optimisation of the formulation or to check through calculations that the embedded waste meets the specifications for all of the waste composition range.

Figure 7.4. General process for the formulation of a cementitious material designed for waste embedding



Source: CEA, 2016.

▪ Finding new alternatives to improve conditioning

Using hydraulic binders as immobilisation materials perfectly meets waste producers' needs; this operation is well controlled on the industrial scale and, as a consequence, has not spurred new developments. Yet, research work on embedding matrices is generally conducted in collaboration with waste producers, with a view to increasing waste incorporation rate and improving confinement performance of cementitious materials, bearing in mind two factors: the increase in waste volume after conditioning, and possible interactions between some waste constituents and cement phases, likely to upset cement hydration and influence the durability of the materials obtained.

In sum, the complexity of cementitious matrix formulation results from the following items:

- the large diversity of the waste to be conditioned;
- the cement/waste interactions liable to degrade the quality of the resulting embedded waste;
- the specifications to be met for the final material, which depend upon its implementation process and its disposal conditions.

The improvements achieved originate in:

- a formulation process rationalised;

- a better understanding of the behaviour of some waste constituents in a cementitious medium;
- newly developed binders likely to afford solutions tailored to specific waste conditioning or ensure better compatibility with the environment.

Geopolymers

Geopolymers is one of the alternative technologies of the cementation. Radioactive waste – such as concentrate, sludge, incinerated ash and resin – is solidified by mixing it with a geopolymer material and an alkaline activator. Silicon and aluminium should be included in geopolymer materials as the major components. The hardening reaction of geopolymer is a dehydration reaction whereas that of cementation is a hydration reaction. The geopolymer is known to form inorganic non-crystalline three-dimensional networks and to have high confinement property of metals than cement solidification, enough compressive strength, resistances to heat and acid, and so on. However, geopolymer is not frequently used for solidification of real waste yet from the point that the material cost is higher than cement materials.

Geopolymer has been investigated and developed for secondary waste generated from contaminated water treatment system in Fukushima Daiichi. Caesium adsorption material (zeolite), slurry and adsorbent of Advanced Liquid Processing System (ALPS) were solidified using geopolymer for basic experiments.

Vitrification

As early as the late fifties, the CEA's Directorate became aware of the management problem related to fission product solutions, and started research programmes in order to solve this problem. After being pre-concentrated so as to reduce their volume, fission products solutions are stored in stainless steel tanks which are constantly stirred and cooled. Their activity, related to spent fuel burn-up, may reach 3.75×10^{13} Bq/L and the power released is significant (up to 7 W/L). These nitric solutions (1 to 2 N) feature high physico-chemical complexity. Their chemical composition generally includes inactive elements such as:

- corrosive products (iron, nickel, chromium);
- additive products (aluminium, sodium);
- solvent degradation products (phosphorus);
- elements issued from clad materials (aluminium, magnesium, zirconium).

There is a broad range of radioactive elements, fission products and actinides concerned, since more than 40 different elements can be numbered ranging from germanium (Z=32) to californium (Z=96). Contrary to what is suggested by the word "solution", usually reserved for homogeneous liquids, "fission products solutions" also prove physically complex: for they contain flocculates and precipitates (zirconium phosphates and molybdates) as well as fine metallic particles (undissolved platinoids such as ruthenium, palladium, rhodium, or intermetallics, e.g. with molybdenum), and fines resulting from fuel clad shearing (zirconium for PWR fuels).

The material selected for conditioning these solutions must display very specific properties because of the complexity of the problem. Early research routes were first focused on mica- or feldspath-type crystalline materials prior to being re-oriented to vitreous materials made in the late fifties. During the sixties, glass was selected by France and the world's community as the confinement material for fission products solutions, because of the flexibility of its disordered structure that enables glass to confine many chemical elements. Glass is endowed with satisfactory properties of:

- thermal stability;
- chemical durability;

- resistance to self-irradiation.

Determining a glass composition means making a compromise between the material properties and the technological feasibility of its industrial-scale fabrication. France has thus selected alumina-borosilicate glasses as confining materials for fission product solutions resulting from the treatment of “graphite-gas” and “light water” reactor fuels.

It must be emphasised that the aim is not a mere embedding, but an atomic-scale confinement, since radionuclides are intimately incorporated in glass structure.

In the case of high-activity powdered waste or sludge coming from decommissioning or legacy waste retrieval, a process of vitrification called “in-can melting” is being developed to encapsulate high activity. It would be useful to study the interest of this process for fuel debris encapsulation once it had been removed from the reactor in pieces less than 10 cm x 10 cm x 10 cm and some remaining powdered sludge. An adequate pre-treatment could be sufficient. Secondary water treatment waste and water contaminated waste could follow the same process.

Canisters

Different types of canisters need to be designed for filter, knockout and fuel debris. Standardisation should always be sought. In France, for example, compacted waste is introduced into a container of the same type as that used for vitrified waste, which ensures standardised conditioning for the ultimate waste generated from La Hague plants. Once the canisters reach the intermediate depository, they are supposed to be continuously vented, but during shipment the problem of the potential production of gas has to be taken into account. New systems have been developed (getters) in order to trap these gases.

Figure 7.5. Ultimate waste arising from spent fuel treatment



Source: Nuclear Waste Conditioning, CEA (2009).

Behaviour studies

Behaviour studies have to be taken into account during transportation, during intermediate storage and then during disposal. Given the long periods of time to be considered, especially for geological disposal, it is not sufficient to simply perform time extrapolation of laboratory-scale results obtained over only a few years. As a first step, it is necessary to understand and prioritise the phenomena occurring in package lifetime under storage or geological disposal conditions. This can be done, in particular, performing laboratory experiments, and observing natural or archaeological analogues. Based upon the data collected, the package evolution can be mathematically described with models which simulate the intervening phenomena, ranging from matrix deterioration to near-field radionuclide migration. This first step aims at getting the assurance, through a wide range of consistent data, that matrix alteration mechanisms are well understood and thoroughly reproduced by modelling. Such studies of waste package and confining matrix long-term behaviour stand for the first step in the safety assessment related to a disposal facility. Owing to the works carried out in the past few years, evolution models could be established for any type of package. Glass was selected for confining long-lived high-level waste due to its flexible use and durability. However, the very long periods of confinement required for long-lived waste disposal made it necessary to closely investigate long-term glass behaviour under disposal conditions. Such studies confirmed that glass showed good behaviour. First, although glass is normally metastable, and, thus, likely to recrystallise into a form thermodynamically more stable than the initial amorphous form, this process is outstandingly slow if glass composition is well chosen. Moreover, this already amorphous material undergoes few structural changes under self-irradiation. Last but not least, glass exhibits good resistance to water: it is true that glass oxides are slowly turned into hydroxides, but this transformation is very slow. The phenomena involved, such as inter-diffusion and hydrolysis, are now well understood. Yet, glass alteration in the very long term is still being thoroughly investigated, since it is much dependent on its environment.

- Concerning cemented waste packages, the main risk to be considered in relation to storage is concrete cracking due to its physico-chemical evolution, as well as some waste/cement interactions, and reinforcement corrosion. Such risk can be reduced using tailored concrete formulations and reinforcing materials (fibres or reinforcement). As part of an exploration approach, other studies have been carried out to optimise cementation in two ways: performing waste pre-treatment, and improving cement formulation for better compatibility with the waste to be conditioned. In the case of a geological disposal site, the major phenomenon affecting cementitious material behaviour is chemical degradation, which strongly depends upon the sulphate and carbonate ion content in the site water. Various models have been developed, especially to predict the evolution of radioactive element confinement in cases when a concrete container is externally degraded by water.
- Concerning the compacted waste package which contains metallic pieces, the proposed model is based upon the localisation of radioactive elements in the package. Radioactive elements located at the surface of metallic pieces are directly driven away by water, whereas those included within metallic pieces are released progressively as metal corrodes. For example, according to laboratory-scale corrosion experiments, radioactive elements included in stainless steel pieces are released within a hundred thousand years.
- Spent fuel direct disposal has also been studied at the CEA though it is not part of the French strategy for the backend of the fuel cycle. Studies carried out on the physico-chemical state of out-of-pile spent fuel have demonstrated that, indeed, fuel rod clads can still confine radionuclides over a time frame compatible with a dry or pool storage of about one hundred years. Yet, it was also shown that fuel

rod clads cannot ensure conditioning with suitable confinement over longer periods, which implies the use of other engineered barriers.

Self-irradiation effects

Generation of radiolysis H_2 outside packages is much dependent on several variables. The first and most important is the dose rate, which induces radiolysis: the higher the energy deposited in water, the higher the water amount decomposed and the H_2 amount produced. The second is the system confinement, which determines whether a steady regime or an equilibrium pressure may be possibly reached. Other factors are involved as well, such as the radiation nature. All radiations do not generate the same radicals in the same proportions. Given the diversity of conditionings and the specificity of operating conditions, measuring source term H_2 for all package types cannot be contemplated. Using simulation is a must, assuming that an integrated model is available, likely to manage a minima the radiological inventory evolution, in-solution reactions, homogeneous and heterogeneous equilibria and a gas transport.

The CEA developed the “operational description of water radiolysis in irradiated materials” (DO-RE-MI) model which simulates radiolysis over several hundred years and evaluates the hydrogen amounts generated. Furthermore, simulation provides a tool for better understanding as it enables various configurations to be tested.

Methodologies for studying self-irradiation effects

The aim of these investigations is to determine whether the glass properties will be altered or not by the successive disintegrations occurring within glasses during their geological disposal. Therefore, it is of prime interest to determine how to speed up the timescale so as to simulate the potential consequences liable to occur on very long durations, typically from ten to several hundreds of thousands of years. For this purpose, an approach based upon several complementary axes has been implemented at the CEA. It mainly consists of specific experiments allowing nuclear glass ageing under disposal conditions to be explored on the laboratory scale (over a period of about one year), as well as atomistic simulations which can help understand the origin of the observed phenomena at the atomic scale. This whole set constitutes the basis required to achieve robust long-term behaviour models.

For high-activity waste, radiolysis gas generation is not only a shipping concern but also a concern during on-site storage. As a result, the vessels have to be designed to be vented during storage which aided in the installation of a sampling system to characterise the gas generation. Consequently, work began on development and testing of a catalytic trapping system. Successful tests were performed which demonstrated that catalyst inserted in the existing vent port screen provided satisfactory trapping performance using a palladium catalyst. This eliminated the gas generation hazard.

Figure 7.6. CEA sites for temporary storage of high-level radioactive or long-lived waste



Source: CEA, 2016.

7.2. Case studies

TMI-2 Waste conditioning

Low-level waste management

As discussed in Chapter 1, due to the issues with radioactive waste disposal, the cost of disposal and limited on-site storage, practices of radioactive waste minimisation were adopted early in the TMI-2 clean-up process. These practices included:

- reducing waste at the source;
- only taking into contaminated areas the tools needed to do the job;
- storage of contaminated tools;
- performing maintenance on tooling and equipment inside contaminated areas;
- recycling water to the extent practical;
- on-site waste reduction.

To support on-site waste reduction, a waste handling and packaging facility and a respirator cleaning and laundry facility were constructed.

The Waste Handling and Packaging Facility went into operation in February 1987. This 2 500 ft² facility was designed and built to provide an environment for the decontamination of materials for unconditional release, volume reduction, sorting of materials and the compaction of materials in drums. The Waste Handling and Packaging Facility (WHPF) was justified by cost savings resulting from the commercial release of decontaminated material, improved packaging efficiency for non-compacted material in boxes and the improved packaging efficiency for compacted material in drums. In terms of volume reduction, the WHPF improved packaging efficiency by 25-30% and significant quantities of metal and other items were released for commercial scrap or reuse on-site.

After the accident, a temporary contaminated laundry and respirator cleaning complex was set up to launder contaminated protective clothing and to decontaminate, clean and sanitise respirators. The complex operated from shortly after the accident until early 1985, when the permanent laundry and respirator cleaning facility was completed and became operational.

Abnormal waste management

▪ EPICOR-II

EPICOR-II resin waste was stored in the solid waste staging facility as described in Chapter 1; it was an engineered storage facility constructed as a long-term solution to spent resin liner storage. The facility consisted of 2 modules containing 60 cells each. Each rectangular concrete module was approximately 50 feet (15 m) wide by 90 feet (27 m) long by 19 feet (6 m) high. The module base and walls were 3 feet (1 m) thick to ensure the surface radiation levels remained below 50 Sv/hr. The 6-foot (2 m) diameter by 12-foot (4 m) high cells consisted of concrete-shielded, galvanised, corrugated-steel cylinders with welded steel base plates. A drain line from each cell led to a common sump. A 3-foot (1m) thick concrete lid covered each cell.

In mid-1981, a task was initiated to develop a sampling/purging to prepare EPICOR-II prefilters stored in the solid waste staging facility for transport from TMI. In late 1981, a plan was developed to transport, store, examine and dispose of the 50 EPICOR-II prefilters. The plan reflected agreements outlined in the US Nuclear Regulatory Commission (NRC)/Department of Energy (DOE) memorandum of understanding, as described in Chapter 8, regarding acceptance of abnormal waste by the DOE for research. Subsequently the prototype gas sampler was designed to remotely remove/reinstall vent

plugs and sample, vent and purge the liners, thus removing potentially combustible gases. It was delivered to TMI in early 1982, for testing/training operations. GPU Nuclear built a portable, concrete “blockhouse” to sit over and enclose the device and shield operators during opening and venting of liners. An operations trailer housed the control panel, related equipment and operating personnel. Integrated functional testing of the prototype gas sampler, blockhouse, and operations trailer was completed in 1982. As described in Chapter 8, beginning in April 1982, the EPICOR-II prefilters were retrieved from storage, vented and purged using the prototype gas sampler, and shipped to the Idaho National Laboratory. The last prefilter was received in July 1983.

■ Submerged demineraliser system

Submerged demineraliser system (SDS) vessels were stored underwater in a fuel storage pool. Because of the much higher radioactivity loadings for the SDS vessels, the gas generation issue was more formidable. At these activity levels radiolytic gas generation was not only a shipping concern but also a concern during on-site storage. As a result, the vessels were designed to be vented during storage which aided in the installation of a sampling system to characterise the gas generation in the SDS vessels. The following observations were drawn as a result of this programme.

- The gas generation rate was proportional to the curie loading and was approximately 0.001 cc/Ci-h.
- The gas generation rate per curie was approximately proportional to the amount of remaining water in the vessel for the range from 2.8 to 5.2 ft³ of water.
- The gas generation rate showed no sign of decreasing with increasing gas pressure. No approach towards equilibrium was observed.
- Stoichiometric gas mixtures did not immediately evolve in the vessels. The hydrogen/oxygen ratio of the resulting gas mixture was higher than stoichiometric but approached it with time.
- The gas generation rates in the SDS vessels loaded with more than 15 000 curies were sufficient to result in a flammable gas mixture by the end of the 14-day testing period.

The results of this testing showed that at curie levels above 15 000, vacuum dewatering, venting and inerting prior to shipment was not enough to ensure compliance with US Department of Transportation requirements.

Consequently, work began on development and testing of a catalytic recombiner system. Successful tests were performed which demonstrated that catalyst inserted in the existing vent port screen provided satisfactory recombiner performance using a palladium on alumina catalyst. This eliminated the gas generation hazard.

■ Damaged fuel

As a result of the TMI-2 accident, the TMI-2 core was severely damaged, the extent of the damage was not known for several years but defueling commenced in 1985 based on the known conditions. To commence defueling three types of canisters, filter, knockout and fuel were designed and licensed by the NRC in accordance with then existing NRC regulations for storage of spent fuel.

The filter canisters were used with water clean-up system to capture fine material on sintered metal filters to maintain water clarity in reactor vessel.

The knockout canisters were used in conjunction with the vacuum and air lift systems. Water and smaller pieces of debris were pumped into the canister. As the velocity of the water decreased in the large diameter of the canister, the pieces of debris settled out of the water.

The fuel canisters were the basic canister for containing core debris. It had a removable lid and could be loaded with larger pieces or most of a fuel assembly. Since few fuel assemblies were full length (none were full cross section) the length limitation was not a problem.

- Accident-generated water

At the time of the accident, the installed auxiliary building storage capacity could only accommodate about 190 m³ of excess water.

Work on a liquid waste storage facility known as the “tank farm” was started in early April 1979. This facility consisted of four (4) 55 m³ tanks and two (2) 95 m³ tanks with associated piping. These tanks were placed in the “A” spent fuel pool, which was lined with stainless steel and had been empty at the time of the accident. This location was selected because it was in a safety-related building that had surfaces amenable to decontamination and was out of the way of ongoing recovery tasks. The tank farm began receiving radioactively contaminated water in July 1979. It eventually held about 264 m³ of auxiliary building water waiting processing.

In addition to new tanks, existing tanks were converted to new purposes. There were two 850 m³ condensate storage tanks on the south side of the plant, adjacent to the turbine building. These tanks normally contained non-radioactive makeup water for the secondary system. In 1980, one of these tanks (COT-1A) was converted to store processed water containing low levels of radioactivity. This water was used principally for decontamination flushes.

Since the company could not release accident-generated water after processing, storage tanks were needed to store the processed water for reuse and recycling. Two (2) 1 900 m³ tanks were constructed along with a small building containing the valves, piping system and two pumps. Construction of these processed water storage tanks on the east side of the auxiliary building began in March 1980. The tanks and associated processed water storage and recycle system were placed in service in July 1981. These tanks have since been turned over to TMI-1.

Lessons learnt from TMI-2 waste conditioning

- Low-level waste management

TMI-2 was not dismantled, however the quantity of lower-activity radioactive waste was difficult to control because the clean-up had to proceed as quickly as possible. Consequently, the project team focused on controlling the final volume to be shipped. This was done by decontaminating and reusing equipment or material whenever possible, solidifying waste when necessary, and boxing or compacting the rest.

- EPICOR-II and submerged demineraliser system waste

Based on experience gained during the shipment of SDS and EPICOR-II vessels, radiolysis of water in the canisters was expected. Once the canisters reached the Idaho National Laboratory, they were continuously vented, but during shipment a problem could exist. Drying the contents of each loaded canister would have been difficult, expensive, time-consuming and unnecessary. Consequently, each canister was dewatered before shipment and the debris transported damp. However, catalytic recombiners were built into each canister to control the accumulation of hydrogen.

This eliminated the gas generation hazard in accordance with US Department of Transportation requirements.

- Core accountability

Another significant issue that affected TMI-2 was accounting for the special nuclear material (SNM) inventory at TMI-2. Due to the accident, fuel accountability by the normal method (i.e. accounting for individual fuel assemblies) was not possible.

Since the canisters were filled with a mixture of SNM, other materials and water, there was no feasible method to determine the exact SNM content in each canister and a different accountability methodology was needed as described in Chapter 1.

Management of liquid and solid radioactive waste (SRW) located and stored on the Chernobyl NPP site

The plant for sorting SRW of all categories and treatment of low- and intermediate-level short-lived solid waste (solid waste processing facility – SWPF)

SWPF is part of the Industrial Complex for Solid RW Management (ICSRM) (Figure 7.7).

Figure 7.7. Solid waste processing facility



Source: Tokarevskyi, 2016.

The following process operations are performed:

- receiving SRW of all the kinds intended for subsequent treatment from the Chernobyl nuclear power plant (ChNPP) entities (retrieval facility for solid waste [RFSW], Liquid Radioactive Waste Treatment Plant [LRTP], ISF-2, shelter, units 1-3);
- sorting SRW of all the categories based upon radiological criteria (separation between low- and intermediate-level short-lived waste [LILW-SL], low- and intermediate-level long-lived waste [LILW-LL] and high-level waste [HLW]);
- segregation of HLW, and LILW-LL, from the rest of SRW, and placing them into the special packaging;
- subsequent sorting LILW-SL depending upon the kind of subsequent processing (compaction, incineration);
- transportation of packed HLW and LILW-LL for interim storage inside the specially equipped solid liquid waste storage building;
- receipt of liquid radioactive waste (LRW) for burning in the SWPF facility;
- incineration of combustible liquid and solid LLW-SL in the relevant facility;

- compaction of LILW-SL in the compaction facility;
- transfer of waste packages between different facilities (working areas) included into the RFSW and SWPF, and provision of transfer control;
- sealing and immobilisation of LILW-SL as well as products of their treatment in the grouting facility, inside the disposal containers;
- monitoring activity of waste loaded into packages to ensure the compliance of the waste package with the requirements of the regulations;
- export (delivery) of processed and packed (immobilised) waste to the Engineered Near-surface Disposal Facility (ENSDF) for disposal;
- process control and monitoring of all areas and operations.

Conditions of safety and radiological protection of the personnel are observed at all areas while conducting all the process operations mentioned above.

The main process systems and assemblies of the SWPF are as follows:

- area for import/export including receipt bay, buffer store for incoming waste, and export area, including a buffer store for waste sent to the ENSDF;
- system and equipment for sorting, size reduction and packaging waste;
- compaction facility;
- incineration facility;
- grouting facility;
- transport system;
- waste package control and tracking system;
- heat, ventilation and condition system (HVAC);
- system of waste transportation to the ICSRM repository.

Also, the SWPF includes an interim store for LILW-LL and HLW for which rearranged compartments of the solid liquid waste storage building are used.

■ Incineration facility

The incineration facility (Figure 7.8 to Figure 7.11) is intended for burning solid combustible waste coming from the sorting facility as well as liquid waste generated during the ChNPP operation and subsequent decommissioning.

The incineration facility includes the seven process systems designed for:

- loading solid combustible waste;
- incineration (primary and secondary);
- two-staged cleaning flue gases;
- fine cleaning off-gases;
- exhaust ventilation to maintain underpressure;
- unloading ash;
- delivery of liquid waste for incineration.

The facility is designed for continuous operation mode with minimum personnel involved.

Underpressure is permanently maintained inside the facility even if it is idle. This prevents the release of residual activity into the atmosphere of rooms where the personnel stays permanently. There is also the special additional fan envisaged in the exhaust ventilation system.

Efficient system of liquid cleaning gas, and filtration of flue gases reduces releases into the ventilation stack down to the limits established by the regulations and the ChNPP requirements.

Residual ash generated while burning is collected into 165-l drums and, after pre-estimation of activity and radionuclide composition, undergoes compacting in the compaction facility.

Figure 7.8. Facility for radioactive waste sorting before incineration



Source: Tokarevskiy, 2016.

Figure 7.9. Incineration facility



The incineration facility is designed for incineration of solid waste coming from the sorting facility as well as organic liquid waste of the ChNPP (oils). The incineration facility envisages the required level of industrial safety (protection against hot surfaces) and radiological safety (radiological protection).

The incineration facility consists of seven process elements:

- solid combustible waste feeding system;
- incineration (primary and secondary combustion);
- two-step flue gas scrubbing;
- off-gas fine filtration;
- exhaust ventilation to maintain negative pressure;
- ash discharge system;
- liquid waste feeding system.

The incineration system consists of three subsystems:

- Incineration subsystem, including delivery of solid waste to the incinerator, incineration, ash discharge, reception and charge of liquid combustible waste.
- Off-gas filtration subsystem, including off-gas cooling, gas scrubbing, filtration and pressure control.
- Waste packaging subsystem, including packaging of solid waste into the plastic bags and then into drums and storage of solid combustible waste.

Figure 7.10. LRW: The warmed tank with mixer for contaminated oil



Source: Tokarevskiy, 2016.

Figure 7.11. Camera for unloading of ash from the furnace of burning and camera of reburning



After unloading drums filled with ash should be treated as compactable waste.

Ventilation system is a safety-related system and very important for safe operation of incineration facility. Two groups of high-efficiency particulate air filters will be used.

Figure 7.12. One of the two groups of high-efficiency particulate air filters



Source: Tokarevskiy, 2016.

■ Compaction facility

The compaction facility (Figure 7.13) is designed for compacting LILW-SL in 165-l drums coming from the sorting and incineration facilities after passing the monitoring and control system.

The use of compaction method in SRW treatment allows for the reduction of the volume of waste by some 2-5 times, and, hence, the number of containers sent for disposal, and it provides for maximum cost savings while filling the ENSDF.

The waste drum is loaded into the press where the regulated force of up to 20 000 kN is applied to it. The resulting pucks are placed in the disposal container and then, the filled container is delivered to the grouting facility for immobilisation of loaded SRW.

After sorting and fragmentation in the SSR-cell, the waste is placed in 165-l drums in which the waste is compacted in the high-pressure press (superpress). Additionally, the ash from the incineration facility is compacted.

Operations of the compaction facility include:

- delivery of drums with waste for compaction;
- loading of drums with waste for compaction;
- pressing;
- storage of compacted waste (pucks);
- filling of concrete containers with pucks.

The compaction facility is intended for treatment of waste arriving from the sorting and incineration facilities after passing through the monitoring system.

Through compaction, the waste volume is reduced by 2-5 times; thus, the number of containers sent for disposal is reduced respectively and the filling of the ENSDF is achieved with maximum efficiency.

The drum loading system is a part of the compaction facility and includes a supporting frame, loading trolley, drum centring unit and drum perforation unit (the perforation unit punches holes to release the air during pressing process). Radiological risks related to compaction of drums with ash are:

- spreading of the radioactivity (ash) from perforated holes;
- contamination of room and equipment by radioactive ash and increasing of concentration of radioactive aerosols in the air;
- decontamination of surfaces and equipment;
- additional doses for personal.

Because of these risks, it was decided to postpone the operations related to the ash compacting before an appropriate technical solution is found (for example, creation of additional facility for mixing ash with hot paraffin and only after that compaction of drums with solidified paraffin).

Figure 7.13. Compaction facility



Source: Tokarevskiy, 2016.

■ Grouting facility

The grouting facility (Figure 7.14) is designed for immobilisation of SRW inside the disposal containers.

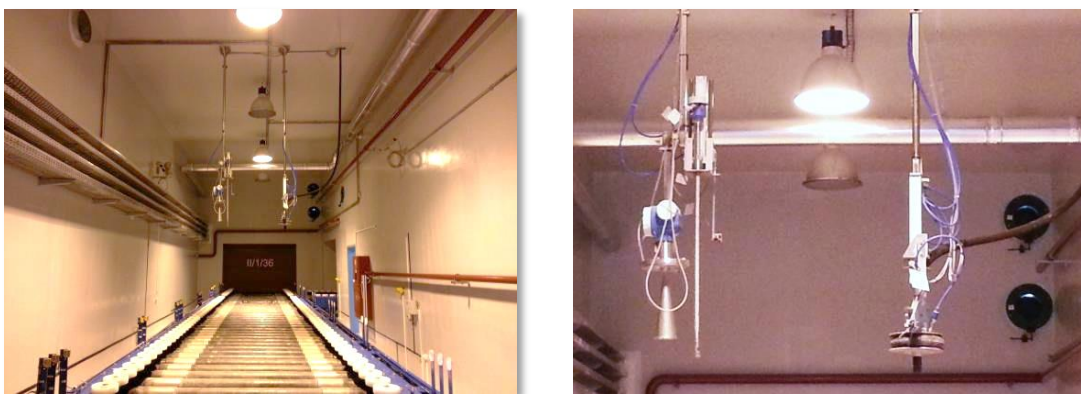
Before grouting, the activity is calculated along with radionuclide inventory of waste located inside the container. These data are introduced into the waste monitoring and

tracking system, and then used while compiling the passport for the container sent to the ENSDF for disposal.

Immobilisation of the waste to be disposed of in the ENSDF is achieved through grouting, i.e. filling the free space in the waste container with dough-like cement mass. Because the waste volume in the container is calculated before grouting, and the level of cement in the container is monitored by an instrument during the cementation process, it is ensured that every batch of the cementation solution is prepared strictly in the quantity that is required, with virtually no remaining residues. The volumes of the first batches are defined based on the calculated waste volume in the container; the volumes of the final batches are defined based on the actual level of the container being filled, as monitored by an instrument.

There are two types of solid waste delivered for grouting. The pucks produced by the compaction facility represent the first type. The pucks are stacked up inside the container. The second type is non-compactable waste placed in the SSR-cell directly into the disposal containers.

Figure 7.14. Grouting facility



Source: Tokarevskiy, 2016.

The grouting facility consists of two main subsystems:

- solution preparation subsystem;
- solution delivery subsystem.

These two subsystems operate together to perform the filling of the containers. If during 30 minutes after filling one container the next containers are not ready, which may happen after cementation of all containers or at the end of the working day, then the small grout residue still present in the concrete mixer and supply hoses is unloaded and the plant is flushed with water (except the loading nozzle). All used water is collected and recycled. The loading nozzle BD-AF1001 is dry-cleaned by a wire brush.

Control of the compound supply process is made from the local control panel BD-GS9051. The system of cementation monitoring, making compound and supply, is located at control panel BD-GS9001. Cabinet BD-GS9002 – is the control system that fills the bunker with cement and cleans its filter. The control system doses the feed of water and super plasticiser into the mixer tank, feed of cement from the silo into the mixer tank, monitors the mixing process and cyclicity of operation of the mixer during grout supply.

The control system can work in automatic and manual modes. Working in the manual mode is limited to checking after maintenance or removal of the grout residues from the body before cleaning.

The control system also manages the automatic flushing sequence and post-flushing water cleaning, the supply of prepared grout into the grouting facility by the feed pump, and monitors the dosing level. After filling of cement solution and curing the packages should be removed to the buffer storage (Figure 7.14).

Buffer storage (see Figure 7.15): The volume is defined to include 21 disposal containers at their arrangement in one layer. Such a volume of the store permits continuous operation of the SWTP for seven days in case of any delays in the ENSDF operation. The RFSW and SWTP operation has to be stopped in case of the suspension of the receipt of waste to the ENSDF for a term above seven days.

Figure 7.15. Buffer storage



Source: Tokarevskyi, 2016.

- Liquid RW treatment plant

Figure 7.16. Liquid Radioactive Waste Treatment Plant



Source: Tokarevskyi, 2016.

The technological process of liquid RW processing includes:

- extraction and supply of LRW in LRTP receiving tanks;
- LRW analysis;
- preliminary treatment of waste in order to meet requirements of the next stages of a technological process;
- decrease of the preliminary treated waste volume (evaporation and centrifugation);

- solidification of preliminary treated waste by cementation and loading cement compound in 200 l drums;
- curing RW packages in LRTP curing room;
- RW packages radiological monitoring and loading them into special transport packing set;
- overpack container radiological monitoring;
- RW packages transfer to ENSDF.

LRW to be treated are:

- evaporated concentrate;
- ion-exchange resins;
- filter perlite pulps.

LRW should be retrieved from the nine 1 000 m³ tanks and five 5 000 m³ tanks located in exiting storages of liquid waste. Transportation of retrieved LRW will be performed by connection pipelines. Using suggested formulas for final product preparation during mutual treatment of LRW, there will be 303 116 RW packages capacity that will take 29 years to produce, where:

- mixture of concentrate (vat residue) and ion-exchange resin – 129 786 RW packages (12.4 years will be required);
- spent filter perlite – 30 705 RW packages (2.9 years will be required);
- remaining concentrate (vat residue) – 142 625 RW packages (13.5 years will be required).

Decreasing of the preliminary treated waste volume should be achieved due to processes of:

- Centrifugation (before cementation, sludge containing perlite pulps or resins are transferred to separator for a liquid/solid separation. The pulps are centrifuged until reaching a constant quantity of residual water).
- Evaporation of concentrates (concentrates are evaporated in the LRTP evaporator to increase the salt content up to 690 g/cub.dm. The evaporator is a conical vertical tank without overflow heated by steam circulation).

Figure 7.17. LRW separator



Source: Tokarevskiy, 2016.

Figure 7.18. Mixer



A mixer is used for incorporating de-watered resins, perlite pulp and over-concentrate in the final product. Waste and water metering are performed directly in the mixer according to formulas of the final product. Batches of initial components-additives for solidification – cement and kaolin – are stored in silos.

L RTP design foresees use of two types of RW packages:

- 200-l drum;
- overpack container (KT3-3.0).

The final product of the L RTP is a compound composed of incoming waste and binding substance. After being mixed, the final product is put into 200-l metal drums. The mass of filled drums ranges between 370-417.6 kilograms depending on the type of incoming waste that they contain. The design capacity of L RTP is 42 drums/day. For transportation RW packages to the disposal site, TPS type K3-3 (3) (KT3-3.0) is used. The drums with the final product are kept in the room for drum storage (curing area) and decontamination during seven days. The room is designed for location of 251 RW packages. The dose rate of surface contamination is monitored. Four drums are placed in a reinforced concrete transport container K3-3 (KT3-3.0). Then the dose rate of γ -radiation is measured on the outer surface of the transport container. Before delivery to ENSDF, preliminary monitoring of RW acceptance criteria for disposal must be met.

Figure 7.19. LRW curing area



Source: Tokarevskiy, 2016.

Figure 7.20. Transport container



Lessons learnt from Chernobyl NPP case study

Positive conclusions related to the designs of all radioactive waste (RW) treatment and conditioning facilities:

- implementation of the best international experience/practice for RW treatment and conditioning (“commonly used technologies”);
- all operations related to RW treatment and conditioning are managed automatically – no manual operations with RW;
- exposure doses of workers are minimised due to distant operation of facilities.

Some lessons learnt after construction and commissioning of facilities for treatment and conditioning of liquid and solid waste (SWPF and L RTP):

- All operations are managed automatically and distantly. At the same time, fall out of “one link in the chain” can stop the working process.

- Organisation of quick reparation/replacement of broken detail or equipment is needed to avoid delays in working process (especially equipment of radiological control system); staff or contractor should be available.
- Human intrusion on occasion into the technological process to correct/change/update some operation.
- High operational expenses.
- IT specialists are needed to deal with software and it is updating.
- Technological process does not always provide compliance with waste acceptance criteria (WAC). Problem after cementation: how to deal with RW packages that are not in compliance with WAC? (Additional procedures should be established by the ChNPP and discussed with operator of disposal facility and regulatory body.)
- Technology and equipment to be used for LRW treatment have to ensure practical implementation of recipes and further acceptance of conditioned RW for disposal in ENSDF: Development and approval of technical specifications for RW packages shall be required.
- Treatment technology should be checked very carefully with “RW imitators” before dealing with real RW. After solidification, some drums may be cut or damaged to check the compliance of the final product with physical and chemical criterion established in WAC for disposal in ENSDF.

Some lessons learnt after commissioning of the compaction facility of SWPF:

- allows for a significant decrease (minimisation) of the volume of RW;
- possible problem: how to ensure safety during the compaction of 165-l drums with ash (Product of incineration facility that should be compacted before cementation.)

Some lessons learnt after construction and commissioning of LRTP:

- need to “redesign” some safety-related systems and components in the end of LRTP construction;
- system for extraction of LRW approved in the initial design of LRTP was totally replaced (the ChNPP decided to update and modernise their own “LRW extraction system”);
- radiation control system was changed, taking into account changes in “zoning” inside the LRTP, updated technical specifications were developed;
- delays with commissioning of LRTP because of needs to review and approve of updated design and Safety Analysis Report LRTP before commissioning.

Some lessons learnt related to some LRTP equipment and technology:

- System (“separator”) for perlite treatment was only “theoretically” suitable for operations with perlite. In practice, components of installed separator were broken in a very short time after starting of work.
- The problem “how and what to do with perlite” is not yet solved. The decision was postponed by the ChNPP.
- Problem with use of “LRW recipes” approved in the initial LRTP design were based on the mixing of all kinds of LRW with cement. Practical impossibility of using the existing perlite treatment technology created new challenges:
 - need to change the recipes;
 - need to change the sequence of different LRW type treatment

(It was decided by the ChNPP [before the LRTP commissioning] to develop a special document related to the changes in recipes and sequence of different LRW types of treatment and to approve it with the regulatory body).

Waste volume reduction – Chernobyl NPP site

As was mentioned above, there is a number of facilities on Chernobyl NPP site to be used for radioactive waste volume reduction:

- the compaction and incineration facility for solid RW volume reduction (including the possibility of radioactive oil incineration);
- evaporator for reduction of liquid RW volume.

It should be mentioned that the purpose of waste volume reduction will be reached if we plan to reduce a volume of so-called “raw” (non-treated) waste allocated on the site. In the case of the Chernobyl NPP, there is a complex of facilities for treatment and conditioning of radioactive waste. As a final product, the drum or container with solidified (cemented) waste will be produced. This final product should comply with waste acceptance criteria developed for near-surface disposal facility. It means that using facilities for radioactive waste volume reduction cannot guarantee that the volume of conditioned waste will be less than the volume of non-treated waste. The total amount of conditioned waste should be calculated taking into account the waste acceptance criteria of the disposal facility, physical and chemical characteristics of conditioned waste and other factors.

The potential application of the clearance concept to materials (mainly “metal”) accumulated at the ChNPP site became urgent with the increased dismantling of equipment that is not considered important to safety. ChNPP units 1, 2, 3 are under decommissioning and the amount of equipment to be dismantled increases each year. In 2010, the regulatory authority developed and approved a regulatory document to govern activities related to the release of materials from regulatory control (the “clearance procedure”). Implementation of this regulatory document by the licensee envisages development of a “clearance methodology” for specific materials (at present, only metal) and further efforts of the licensee in compliance with the “clearance methodology” agreed by the regulatory authority. At the same time, documents complying with the established criteria for release from regulatory control for each batch of materials have to be submitted to the regulatory authority. The regulatory authority has the right to inspect this activity. It should be noted that only the “unconditional clearance” option is currently applied and such an activity is allowed within the licence for ChNPP units 1, 2, 3 decommissioning. Concerning the possibility of clearance for shelter materials (the completely destroyed ChNPP unit), this issue has not been considered.

Today, it should be added that the ChNPP involved subcontractors for activities related to dismantling, decontamination and release from regulatory control, but in the near future, the ChNPP plans to improve and extend the scope of dismantling, decontamination and release of materials from regulatory control. At present, under the support of the European Commission, a project to create a “free release facility” is implemented on the site, and an additional site is being arranged for decontamination of dismantled equipment.

Lessons learnt:

- the purpose of waste volume reduction will be reached if the plan is to reduce a volume of non-treated waste allocated on the site;
- use of the facilities for radioactive waste volume reduction cannot guarantee that the volume of conditioned waste (ready for disposal) will be less than the volume of non-treated waste;

- establishment and implementation of clearance procedures is an effective instrument to reduce the volume of waste during the dismantling of equipment not related to safety (mainly metal components);
- dismantling, decontamination and release of materials from regulatory control are performed on units 1, 2 and 3 of the Chernobyl NPP and not on the shelter object (totally destroyed unit 4);
- only the “unconditional clearance” option is currently applied at the ChNPP site.

Waste conditioning studies for Fukushima Daiichi RW

Waste conditioning studies for Fukushima Daiichi RW

During FY 2017, the “basic concept of conditioning and disposal for solid radioactive waste” should be complied with, while conducting such measures as characterisation of solid waste, studying the applicability of conditioning/disposal technologies widely selected, and developing an analysis method for difficult-to-measure nuclides and an inventory evaluation technology, as well as referring to comments of the Nuclear Regulation Authority.

Based on these efforts, actively using radioactive-material analysis/research facilities now under design and accelerating research and development (R&D) through the characterisation of solid radioactive waste, prospects of a conditioning/disposal method and a technology related to its safety should be made clear by around 2021.

In parallel with confirmation of the prospects concerning safety, TEPCO should present, at an early stage, measured data or a coping policy related to safety securement during storage and management, and should take other measures to properly address this issue.

In accord with these efforts, specifications and production methods of the waste packages should be determined in phase 3.¹ A conditioning system should be installed in the power plant. After establishing the prospects of disposal, production of waste packages should then be started, and then they should be carried out.

Waste volume reduction in Fukushima Daiichi

The amount of generation of solid radioactive waste should be reduced on an ongoing basis by:

- preventing materials that can turn to waste from being brought onto the site as much as possible;
- minimising generation of solid radioactive waste;
- reuse;
- recycling.

With these efforts, aiming at more properly storing various types of generated solid radioactive waste, continuous work should be carried out to manage and arrange storage/volume reduction facilities (cutting and crushing system) of solid radioactive waste, specifically installing an incinerator for the volume reduction process.

1. Implementation phase from fuel debris retrieval to the completion of decommissioning.

Lessons learnt from waste conditioning studies for Fukushima Daiichi RW

Concerning volume reduction of solid radioactive waste:

- In the future, TEPCO will install both an additional incinerator and a volume reduction facility for metal and concrete in the solid radioactive waste.

Concerning contaminated water treatment:

- Contaminated water has been treated with purification systems² and secondary waste generated from contaminated water treatment systems, as a process of contaminated material volume reduction, and is stored in the temporary storage area.

7.3. Recommendations

The lessons learnt pointed out in each case study of Section 7.2 offer input into the integrated waste management strategy for the damaged site after the accident at the Fukushima site: advice in terms of requirements for treatment and conditioning to make waste suitable for disposal facilities (physical stability, release of radionuclides).

When evaluating existing, alternative or volume technologies for decontamination or waste reduction, the main questions to be asked are:

- What is the typical volume reduction efficiency and range of efficiencies for any given waste type? Who is responsible for that waste?
- Does it change the waste category?
- What determines whether a mixed waste can be sent to a specific treatment, storage, or disposal facility?
- What is the available storage capacity on-site, and what percentage of that storage is under cover?
- How soon will waste be processed and disposed of?
- What is the long-term liability associated with the disposed final waste form?

Then, the technology and equipment to be used have to ensure practical implementation and further acceptance of conditioned waste for disposal.

The purpose of waste volume reduction will be reached if it truly allows for a reduction of the volume of non-treated waste allocated on the site.

Establishment and implementation of clearance procedures can also be an effective instrument to reduce the volume of waste during the dismantling of equipment not related to safety (mainly metal components).

If conditioning with a confinement matrix is required, the choice of process has to be made with safety authorities depending on storage and disposal requirements, taking into consideration cost optimisation: cement, geopolymers, vitrification in-can, etc.

2. Referring to a multi-nuclide removal equipment, additional multi-nuclide removal equipment and high-performance multi-nuclide removal equipment (hereinafter referred to as “multi-nuclide removal equipment”), as well as a mobile strontium removal system, RO-concentrated water treatment equipment, cesium adsorption system and second cesium adsorption system.

The formulation process must in any case be rationalised with a better understanding of the behaviour of some waste constituents in the matrix. For cements, newly developed binders are likely to afford solutions tailored to specific waste conditioning or ensure better compatibility with the environment.

For new types of canisters that need to be designed, development standardisation and approval of technical specifications for waste packages shall be required.

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8. Destination (storage/disposal)

8.1. General description

Introduction

All countries with nuclear power programmes or research programmes involving radioactive materials have developed approaches for the safe storage and disposal of radioactive waste produced from normal activities. These approaches are developed within the legislative frameworks of the individual countries and in compliance with safety standards developed by the International Atomic Energy Agency (IAEA). National legislative frameworks define aspects such as the categorisation of radioactive waste to be stored or disposed of (see Chapter 6) and the requirements that a developer of a proposed storage or disposal facility must fulfil in order for authorisation to develop and operate such a facility to be granted. In the case of disposal facilities, a key requirement is to demonstrate that any releases of radioactivity from the facility during the operational and post-closure periods will not result in regulatory limits or targets being exceeded.

Interim storage of radioactive waste packages produced as part of normal reactor operation and decommissioning activities is an essential part of all radioactive waste management programmes. Interim storage facilities are designed to meet the requirements of national programmes (numbers, locations, etc.), and have design lifetimes commensurate with the expected timescale on which disposal solutions can be implemented. Given the challenges with obtaining approvals for developing new disposal facilities, interim storage facilities sometimes consider storage periods up to about 100 years.

The international consensus for the disposal of higher-activity radioactive waste is focused on deep geological disposal. Deep geological disposal isolates the waste from the human environment and provides containment of radionuclides within the waste by preventing or delaying and attenuating any releases of radionuclides from the repository to the biosphere. Deep geological disposal generally utilises a multi-barrier concept, whereby the waste package, the engineered barrier system and the surrounding geosphere all contribute to safety. Deep geological disposal has been implemented for defence-related intermediate-level waste (ILW) in the United States; preferred sites and geological disposal concepts for deep geological disposal have been identified in countries such as Sweden and Finland. Other countries are currently in the process of developing disposal concepts and selecting suitable sites for deep geological disposal. A range of engineered solutions and potentially suitable geological environments are being considered; these are termed “geological disposal concepts”.

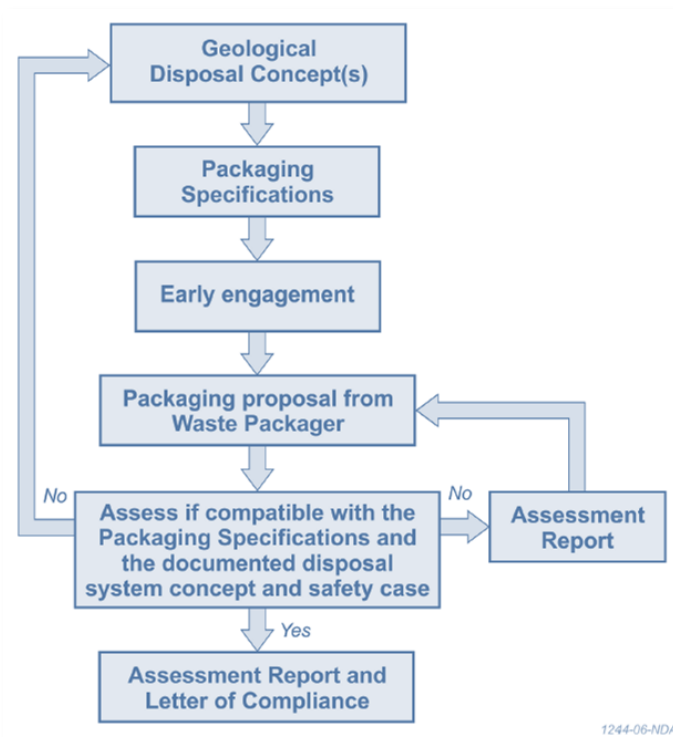
Most lower-activity waste is disposed of in near-surface facilities. The IAEA defines (SSG-29) near-surface disposal facilities as either “ground level”, constructed within a few metres of the ground surface, or “caverns”, constructed tens of metres below the ground surface. Near-surface disposal facilities for waste produced from normal reactor operation and decommissioning activities are operational in many countries.

Waste acceptance

Safety cases for existing disposal facilities will have been developed based on assumptions about the inventory and nature of the waste to be disposed of and on the expected performance of the various barriers that prevent radionuclides returning to the human environment in harmful quantities. For operational facilities, the operator specifies waste acceptance criteria (WAC), which define the radiological and physical/chemical characteristics of waste that is acceptable for disposal at the facility. WAC and associated control arrangements are an essential part of ensuring the safety of the facility, both during and after the operational period. An example of the approach to developing waste acceptance criteria for a near-surface disposal facility is provided by the Low Level Waste Repository in the United Kingdom (LLW Repository Ltd., 2011).

It is not possible to specify WAC for a disposal facility where the location is not yet known or the design is not yet fully developed. An example of this situation is provided by the UK geological disposal programme for higher-activity waste, where the siting process for the geological disposal facility (GDF) is still at an early stage and a range of geological disposal concepts are still being considered. The UK programme is also of interest to this report because the history of nuclear power development in the United Kingdom has meant that a wide range of operational and decommissioning waste requires disposal. As a precursor to WAC, Radioactive Waste Management Ltd (RWM) has developed a disposability assessment process (NDA, 2014) whereby waste producers wishing to consign new waste streams or to use new packaging solutions are required to undertake disposability assessments to demonstrate the performance and safety of waste packages during their transport to the GDF, during handling and emplacement at that facility, and in the longer-term post-closure period. The disposability assessment process is illustrated in Figure 8.1.

Figure 8.1. Disposability assessment process



Source: NDA, 2014.

The aim of the process is to minimise the possibility that the conditioning and packaging of radioactive waste results in packages incompatible with the selected geological disposal concept.

In some countries, there is a more prescriptive approach to waste destination, with specific disposal concepts being acceptable for specific waste categories. In other countries, the approach is less prescriptive. For example, in the United Kingdom, Section 3.4 of Statutory Guidance (Environment Agency, 2009) states that “Types of solid waste that might be suitable for disposal in near-surface facilities include very low-level waste (VLLW), low-level waste (LLW), and shorter-lived or less radiotoxic ILW.” Clearly, the onus is on the developer to demonstrate the safety of the facility to the appropriate regulators. A consequence of this approach is that RWM is undertaking work to identify opportunities for improved management of higher-activity waste in the United Kingdom, including those waste at the LLW/ILW boundary where a safety case could be made for either near-surface or deep geological disposal (NDA, 2015). Potential benefits from diverting such waste to near-surface disposal include reduced packaging and disposal costs and optimising the use of a GDF.

Subject to disposal capacity constraints, it is desirable that waste produced as part of an accident are disposed of using existing or planned disposal facilities for waste from “normal operations”. In these circumstances, it will be necessary for waste packages to meet either existing WAC or precursors, or to demonstrate using a process such as “disposability assessments” that the waste is likely to be acceptable for a planned facility. Guidance on these processes is widely available (e.g. NDA, 2014).

Clearly, where larger volumes of waste exist as a result of an accident, use of existing or planned disposal facilities is not always possible. The following sections describe the final waste destinations for each of the case studies. Note that in this report, the concern is only with waste that arises from the nuclear site itself (i.e. within the site perimeter), not with waste that is produced from the surrounding area. Recommendations are then made in Section 8.4.

8.2. Case studies

Three Mile Island 2 (TMI-2)

Abnormal waste

Much of the radioactive waste that resulted from the TMI-2 accident clean-up could not be disposed of as low-level waste and there was no disposal facility for this higher level or “abnormal waste”. In addition, much of the waste was not comparable to those produced at an operating power plant. The waste contained a high concentration of fission products or small quantities of fuel materials. The waste processing systems had not always been configured to produce waste in the form and concentrations allowed for shallow land burial.

As part of the solution, the US Department of Energy (DOE) and the Nuclear Regulatory Commission (NRC) signed a memorandum of understanding (MoU) in July 1981 to ensure the TMI site did not become a long-term waste disposal facility. The agreement also took advantage of the chance to learn from the accident. The DOE agreed to evaluate each waste form to determine the research and development (R&D) value and if of value, to accept the waste for research and later disposal. If the waste was not of research value or could not be made acceptable for commercial disposal the DOE would temporarily accept and store the waste on a cost-reimbursable basis. This agreement was crucial for disposal of all the TMI-2 radioactive waste.

The MOU identified several types of radioactive waste and potential means of disposal. These included:

- EPICOR-II waste – For the highly loaded prefilters, the DOE proposed to develop a high-integrity container that might allow commercial land burial at Richland. Characterisation work would also be performed on one or more vessels.
- Submerged demineraliser system (SDS) waste – For the 19 highly loaded SDS vessels, the DOE would conduct a waste immobilisation R&D and testing programme, including monitored retrievable burial.
- Reactor fuel – Initially, the DOE planned to take samples for analysis, characterisation and research while the balance of the fuel debris remained on-site in the spent fuel pool. Final disposition would await resolution of the national spent fuel issue. As the issue was going to take a long time to resolve, and was still not resolved in 2016 the DOE and NRC modified the MOU in March 1982 so that the DOE accepted the entire reactor fuel core. Part would be used for R&D; the remainder would be stored until ultimately disposed of. The TMI-2 damaged fuel is currently in dry cask storage at the Idaho National Laboratory.

At Three Mile Island, neither the owner, General Public Utilities (GPU), nor the NRC considered the site a suitable location for the long-term storage of waste. Because of this concern, negotiations were conducted between the three parties to develop an agreement that would lead to the removal of all waste from TMI. As all three parties shared the same goal, removal of waste from TMI, and the DOE has an R&D mission, this agreement provided benefits to all parties.

▪ EPICOR-II

Beginning in April 1982, the EPICOR-II prefilters were retrieved from storage, vented and purged using the prototype gas sampler, and shipped to the Idaho National Laboratory. The last prefilter was received in July 1983. Meanwhile, since 1980, alternative means of disposing of the EPICOR-II prefilters were investigated. The high-integrity container (HIC) was selected for possible use as an overpack in disposing of prefilters commercially as class C waste. The HIC was designed to retain liquid and solid waste of a prefilter while buried at intermediate depths for 300 years (approximately ten half-lives of the predominant radioisotopes).

In April 1984, the first prefilter was transported to Richland and disposed of in a trench. The remaining prefilters designated for disposal were shipped to Richland with disposal operations being completed in February 1985.

Various high-integrity containers were used for disposal of TMI waste. Although we do not have access to the specific radiation embrittlement testing that was performed, each container would have needed to satisfy the requirements in the NRC Final Waste Classification and Waste Form Technical Position Papers, dated 11 May 1983.

These requirements are as follows:

- The high-integrity container design should consider the radiation stability of the proposed container materials as well as the radiation degradation effects of the waste.
- Radiation degradation testing should be performed on proposed container materials using a gamma irradiator or equivalent. No significant changes in material design properties should result following exposure to a total accumulated dose of 10^8 Rads. If it is proposed to design the high-integrity container to greater accumulated doses, testing should be performed to confirm the adequacy of the proposed materials. Test specimens should be prepared using the proposed fabrication techniques.

- Polymeric high-integrity container designs should also consider the effects of ultra-violet radiation. Testing should be performed on proposed materials to show that no significant changes in material design properties occur following expected ultra-violet radiation exposure.

- Submerged demineraliser system

Of the 19 SDS vessels that the DOE agreed to accept, three were shipped to Pacific Northwest Laboratory in 1983 for use in vitrification experiments. The contaminated zeolites were removed from the vessels, glass formers were added, and the mixture was placed in special steel canisters. A full-scale, in-canister melting process was then used to vitrify the material. In this process the canister served as the container for the solidified (glass) final waste product.

The other 16 vessels were sent to Rockwell Hanford for experiments demonstrating remote dry handling techniques and monitored burial in special concrete overpacks. The overpacks were buried at least 3 m underground in a trench. One of these SDS vessels and its overpack was instrumented for monitoring during long-term burial.

- Damaged fuel

A shipping cask was designed to ship the fuel to the DOE Idaho National Laboratory. It was licensed by the NRC in accordance with then existing NRC regulations to ship up to seven canisters from TMI to Idaho by rail. Initially two shipping casks were built by the DOE, with a third shipping cask being built by GPU Nuclear. A total of 342 canisters were eventually shipped in 49 cask loads in 22 separate rail shipments. As part of the shipping programme the fuel transferred ownership from GPU Nuclear to the DOE as it crossed the TMI site boundary such that the DOE was the shipper and not GPU Nuclear.

- Accident-generated water

The final water processing challenge was determining a method for disposing of the processed water. The 8 700 m³ of water contained varying concentrations of boric acid and sodium hydroxide that had been added for criticality control and for pH control. The ion-exchange processes removed the sodium but very little of the boron. In addition, no ion-exchange processes exist that can remove tritium, a radioactive isotope of hydrogen. Even though the most economical and least complex option for disposal was to discharge the water to the Susquehanna River after treatment, the company decided that this option was not the most favourable in light of the City of Lancaster Agreement and public opinion. GPU Nuclear chose to use an evaporator system that would distil the water, remove the boration chemicals and the majority of the radionuclides as a dry solid waste and collect the distilled water containing the tritium for release to the atmosphere. The distilled water was fed to a vaporisation unit that boiled the water and discharged the vapour through a 30 m high exhaust stack into the atmosphere. The dried boric acid waste was packaged in plastic-lined 200 litre drums for shipment and burial.

Processing accident-generated water began on 24 January 1991 and was completed on 12 August 1993. The system discharged the water and produced 158 000 kilograms (190 m³) of dried waste that was shipped as low-level radioactive waste for disposal.

Lessons learnt from TMI-2 accident

- At Three Mile Island, neither the site owner nor the regulator considered the site a suitable location for the long-term storage or disposal of the waste. The common goal of both parties was therefore to remove waste from the site. Lower-activity solid waste that met the criteria for existing commercial low-level burial sites was eventually disposed of at these sites, following a period when the sites were closed to TMI-2 waste. However, existing disposal routes were not available for a large amount of waste generated at TMI-2 (subsequently termed “abnormal waste”),

which exceeded commercial burial criteria. The government and utility worked collaboratively to find a designated route for disposal of this waste.

- The challenge of disposing of waste led to an early and sustained focus on waste minimisation, which provided benefits to the programme.
- Because of the damage to the fuel was so extensive, special nuclear material accountability (SNM) by normal methods was not possible. Therefore, a non-standard approach method to provide the required SNM Accountability was developed and agreed with the regulator.
- The site owner worked with government to find a suitable location for storage of the fuel debris until a final disposal site is opened.

Experience with Chernobyl waste

Requirements for safety of radioactive waste disposal in Ukraine

Ukrainian safety disposal requirements (NRSU-97/D-2000) stipulate that at least the same level of radiological protection is provided for the next generation from disposal of radioactive waste as is provided for the present generation from current activities. Implementation of this principle is achieved by establishing a demand of non-exceedance of harm to the health of future generations in the amount which corresponds to the negligible risk of $5 \times 10^{-7}/y$. Taking into account this requirement, regulations on radiation exposure of future generations (population) are established after release of disposal sites from regulatory control:

- Reference possibility of critical events that may lead to potential radiation exposure – no higher than $1 \times 10^{-2}/y$. If this probability is exceeded, radiation exposure is considered as current.
- Quota limit current exposure dose – is 0.01 mSv/y; reference levels of potential radiation exposure of population during implementation of critical events associated with natural abnormal events and inadvertent human intrusion: 1 mSv/y, non-exceedance of which means that intervention is not justified, and 50 mSv/y which corresponds to the condition of justification for intervention.

This approach, aimed at protecting future generations, somewhat differs from the one accepted in the new IAEA document (SSR-5) and the new version of the Basic Safety Standards. The established level of radiation exposure of the population is too low (0.01 mSv/y), which prevents the use of the principle of limited optimisation. It should be reasonable to bring it in accordance with safety objectives for disposal of radioactive waste, in particular the use of the dose limit of radiation exposure of the population of 0.3 mSv/y from an individual separate radiation source.

It is necessary also to revise the provisions, established in NRSU-97/D-2000 for a “full, restricted or limited to specific requirements” exemption of waste in a storage facility, to reflect the following:

- the statement of “radioactive waste exemption in the storage facility” is not exactly correct; it makes more sense to release the site from the regulatory control;
- the requirement for achieving the levels of “exemption” for each nuclide to grant full exemption of waste in storage facilities seems superfluous (for example, disposal in a surface repository of 10^3 Bq/g of caesium will reach the level of exemption (0.1 Bq/g) in about 400 years);
- “limited or restricted exemption with specific requirements” essentially means extension of control that does not comply with the basic principles and objectives of safe radioactive waste disposal.

Also, radiation exposure scenarios and radiation dose limits should be used, as accepted in the IAEA documents, to determine if exemption of the site is possible.

Final radioactive waste product for long-term storage or disposal

Important documents to be developed and approved are listed below.

- WAC: Should be established by the operator of the storage/disposal facility and approved by the regulatory body. WAC should be based on the safety assessment of storage/disposal facility and should be a part of the “safety case” (in Ukraine – part of safety analysis report).
- Technical specifications for radioactive waste packages to demonstrate:
 - The compliance of conditioned radioactive waste with WAC established for storage or disposal facilities.
 - What methods/measures can be used to check/ensure the compliance of conditioned radioactive waste with WAC? For example, if we have a requirement in the WAC for “no chemical substances in conditioned RW”, methods should be given in the TS to ensure compliance with this criterion.

The TS is developed by the producer of the package of conditioned waste and should be approved by the operator of storage/disposal facility and the regulatory body.
- Passport for RW package. This should be established by the operator of the storage/disposal facility. General requirements for the content of such a passport should be established by the regulatory body.

Conclusions and lessons learnt

- Requirements for the safety of RW disposal in Ukraine are very conservative for some parameters (such as dose limits) compared with IAEA international safety standards.
- There are some incompatibilities between Ukrainian safety regulations and IAEA safety standards related disposal of RW.
- To remove these inconsistencies, Ukrainian regulations related to the disposal of RW need to be updated to take into account IAEA safety standards; at the same time, national RW classification needs to be updated and general acceptance criteria developed.
- Special attention should be given to disposal options for RW of Chernobyl origin that is located in the exclusion zone and is to be disposed of or stored inside this zone. Exclusion zone special dose limits need to be applied to make RW strategy implementation and disposal of RW on the Vektor site a practical possibility.
- The special status of some territories of the exclusion zone should be established (no population at all in the future) to allow the problem of large amounts of RW of origin to be solved.

First experience of disposal of conditioned RW from the Chernobyl nuclear power plant site

The Engineered Near-surface Disposal Facility (ENSDF) for low- and intermediate-level (short-lived) solid radioactive waste is a part of Industrial Complex for Solid RW Management of the Chernobyl nuclear power plant (ChNPP) (Figure 8.2) constructed for conditioned RW arising from the Chernobyl NPP site:

- 200-l drums with cemented liquid RW;
- 3 m³ containers with cemented solid RW.

It is located in the ChNPP exclusion zone at the site of complex Vektor at a distance of 11 km to the south-west of the ChNPP.

Figure 8.2. The Engineered Near-surface Disposal Facility



Source: Tokarevskiy, 2016.

Types of localisation barriers for conditioned (cemented) RW:

- waste form;
- engineering barriers;
- natural barriers.

Basic technical characteristics:

- capacity of disposal facility – 63 200 m³;
- volume of RW inside repository – 55 000 m³;
- number of disposal units – 22 (2 rows of 11 units each);
- overall dimensions of the disposal unit – approximately 25 x 19 x 8 m;
- duration of operation period – 30 years.

ENSDF safety assessment specificity:

- Particularities of ChNPP radioactive waste: partly operational, partly post-accidental.
- Two facilities (solid and liquid treatment plants) produce RW packages: two different types of packages.
- Lack of characterisation data for RW: a great deal of generic and conservative data were used in the safety assessment.
- Strict dose limits, which are set in Ukrainian regulations for long-term radiological protection.
- Individual annual effective dose of the current exposure of critical group of public shall not exceed 0.01 mSv.
- Annual value of the effective dose of the potential exposure shall not exceed 1 mSv.
- The approach used for safety assessment involved conservative scenarios (reference scenarios) and simplified models. No uncertainties or sensitivity analysis was provided.

- The assumed fuel radionuclide composition for all ChNPP RW was too conservative for long-lived alpha-radionuclide content and was not correct for operational RW.
- From analysis of samples of liquid RW, it is now clear that no constant radionuclide composition can be applied for ChNPP RW.

It was recognised that:

- Radionuclide composition could be defined during operation of RW processing facilities by measuring a sample from each batch of liquid RW (receiving tank – 20 m³) in the Liquid Radioactive Waste Treatment Plant (LRTP) and by regular measurement of samples of solid RW before processing.
- Radionuclide content in the particular RW packages will be defined by using these particular radionuclide compositions.
- Radionuclide content in each RW package will be fixed in its passport.

Conclusions and lessons learnt after licensing of ENSDF

- Safety analysis was carried out using limited or practically absent data on specific RW characteristics and applying conservative assumptions and simplified models with a number of uncertainties. That is why relatively established WAC are valid only for the first stage of Lot 3 operation – disposal of RW packages in two symmetric units.
- During the first stage of operation, the Safety Analysis Report (incl. Safety Assessment) has to be revised and improved, essentially taking into account operational experience, real RW characteristics, updated models and justified scenarios of current and potential exposure. Consequently, the waste acceptance criteria have to be revised.
- An operational licence was granted for the first stage of operation with a set of limitations and conditions.
- The producer of RW – Chernobyl NPP – should start the operation of the liquid and solid RW treatment facilities to provide the operator of ENSDF with “real” RW characteristics.

Lessons learnt from the ChNPP accident

Before the Chernobyl accident, there was no experience anywhere in the world in managing the larger amounts of materials resulting from a nuclear accident. The ChNPP case study demonstrates, for the first time, that large volumes of radioactive waste were managed under extreme conditions during and immediately after the accident.

After the Chernobyl accident, a number of sites were created for localising radioactive waste generated from outside the ChNPP site perimeter (see the ChNPP case study in Chapter 4). Only some of such places included engineered barriers to limit migration of radionuclides. There was inadequate waste isolation technology applied at these early facilities, and little characterisation or registration of waste sent to the different facilities. The potential environmental impacts of the storage facilities were not considered. Even today, the majority of the storage sites require in-depth investigation. Retrieval and re-disposal of waste at some facilities is required; in other cases, it is considered more appropriate to undertake safety assessments to determine whether the waste can remain at their current locations. In all cases, it is recognised that these decisions and operations are made more difficult by lack of radiological characterisation information on the stored waste. Currently, works are being performed in the ChNPP exclusion zone to get more information on the waste; this will be used for the safety assessment of RW localisation sites to support the decision-making process related to re-disposal.

Waste generated on the site of Chernobyl NPP resulted from the operational activity of the ChNPP units and waste resulting from clean-up after the accident, and mainly from decontamination activity. These waste are stored in existing (“old”) temporary storage facilities for solid and liquid waste. A set of new facilities were constructed on the ChNPP site to start retrieval, characterisation, sorting, treatment and conditioning of stored waste (see Chapter 6). As a final product, it is expected to produce drums and containers with solidified (cemented) waste to be acceptable for disposal in the near-surface disposal facility.

The first experience of licensing such a facility – the Engineered Near-surface Disposal Facility (ENSDF), – has highlighted the lack of data on radiological characteristics of conditioned waste, because of a lack of information about waste to be retrieved from the “old” storage facilities. This has meant that cautious assumptions are made about the waste to be disposed of at the facility, which in turn means that the facility may not be used to its full environmental capacity. It is recognised that operators of treatment plants should now provide better radiological characterisation of the waste forms being produced.

Moreover, the requirements of the Ukrainian legislation for radioactive waste disposal have been found to be very conservative for some parameters compared with IAEA Safety Standards. One lesson learnt is that Ukrainian legislation should be updated to take account of IAEA Safety Standards, at the same time as radioactive waste classification is updated and general acceptance criteria for future disposal facilities are developed.

Waste destination of Fukushima Daiichi on-site waste

Existing disposal concepts in Japan

There are four disposal concepts for radioactive waste in Japan; trench disposal, concrete pit disposal, subsurface disposal and geological disposal. The depth of trench and concrete pit disposal is less than 10 m. That of subsurface disposal is deeper than 50 m and that of geological disposal is deeper than 300 m (NRA, 2014).

Upper bounds for nuclide concentrations are provided for each disposal concept by law (see Table 8.1). These are determined under general conditions. When designing a disposal facility, the operator has to observe these upper limits. In addition, they have to reselect the nuclides and redetermine the upper limits under actual site conditions.

Trench disposal facilities are considered to dispose the waste from decommissioning work. On July 2015, the Japan Atomic Power Company (JAPC) made the application for trench disposal at Tokai nuclear power plant.

Concrete pit disposal facilities are operated at Rokkasho-mura in Aomori prefecture by Japan Nuclear Fuel Limited (JNFL). The volume of these facilities is about 80 000 m³. This site potentially has a capacity to make facilities with a volume of about 600 000 m³ (JNFL, 2015). Research projects for subsurface disposal were also conducted on this site from 2001 to 2006. However, an application for this facility has not yet been submitted.

For geological disposal, the Nuclear Waste Management Organization of Japan (NUMO), which was established in 2000, has the responsibility. NUMO has started site selection from 2002 but at the present time has not identified any sites. The Japanese government changed the fundamental plan for geological disposal in 2015 (The Cabinet of Japan, 2015). The government strengthened its role and plans to show the scientifically preferable area for geological disposal.

Regulation for clearance was established for concrete and metal waste in 2005. JAPC released 170 tonnes of carbon steel from 2007 to 2010. After the Great East Japan Earthquake in 2011, JAPC stopped these activities and has not yet restarted. Table 8.2 shows a summary of the status of Japanese radioactive waste disposal.

Table 8.1. Upper bounds of nuclide concentration (Bq/tonne of waste) for each disposal concept

	Trench disposal	Concrete pit disposal	Subsurface disposal
C-14	-	1E+11	1E+16
Cl-36	-	-	1E+13
Co-60	1E+10	1E+15	-
Ni-63	-	1E+13	-
Sr-90	1E+07	1E+13	-
Tc-99	-	1E+09	1E+14
I-129	-	-	1E+12
Cs-137	1E+08	1E+14	-
Alpha nuclide	-	1E+10	1E+11

Table 8.2. Status of radioactive waste disposal in Japan

	Disposal depth	Status	
		For waste from NPP operation	For waste from decommissioning work
Trench disposal	< 10 m	Disposal test facility for JPDR waste by Japan Atomic Energy Agency	Under official safety review for Tokai NPP by Japan Atomic Power Company
Concrete pit disposal	< 10 m	Under operation at Rokkasho-mura by Japan Nuclear Fuel Limited	Under planning
Subsurface disposal	> 50 m	Under planning	Under planning
Geological disposal	> 300 m	No	Under site selection

Introduction

Characteristics of the disposal study for Fukushima Daiichi NPP waste are as follows:

- Fukushima Daiichi NPP waste was generated under an uncontrolled situation.
- Available data for the safety assessment of Fukushima Daiichi NPP waste disposal, such as inventory and migration parameters, are limited and include a degree of uncertainty.
- Not only the disposal site, but also the disposal concepts applying to Fukushima Daiichi NPP waste, have yet to be decided.

Under these circumstances, presenting candidate applicable disposal concepts for each Fukushima Daiichi NPP waste and assessment method was set as the goal of the investigation on Fukushima Daiichi NPP waste disposal in 2017.

An approach for the waste disposal study

In the traditional safety assessment of radioactive waste disposal, based on the disposal concept and the assessment method (scenario, model and parameter), release rates through the multi-barrier system are estimated, ending with the transport of radionuclides from the geosphere to the biosphere. These release rates are converted to the dose rate in the biosphere and the disposal safety is demonstrated by comparison with the safety criteria.

On the other hand, in the case of Fukushima Daiichi NPP waste, in order to select applicable options from existing disposal concepts and/or to develop new disposal concepts, requires information obtained from characterisation, waste processing/treatment and waste packaging.

An understanding of the characteristics of the existing disposal concepts and assessment methods includes:

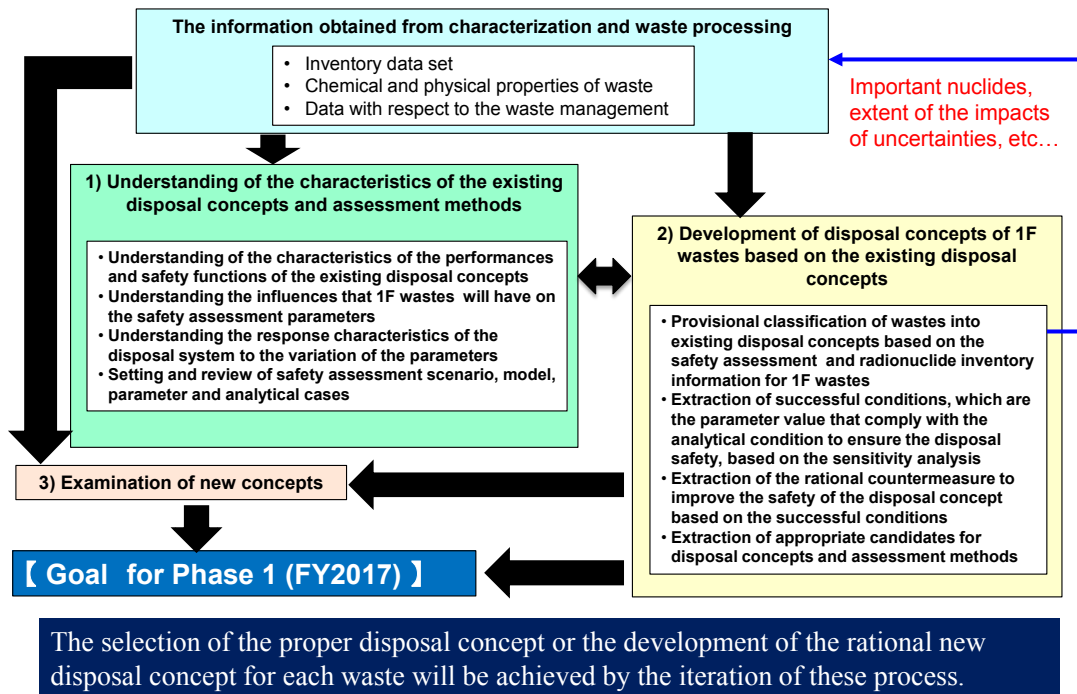
- the characteristics of the performances and safety functions of the existing disposal concepts;
- the influences that Fukushima Daiichi NPP waste will have on the safety assessment parameters;
- the response characteristics of the disposal system to the variation of the parameters;
- setting and review of safety assessment scenarios, models, parameters and analytical cases.

Development of disposal concepts of Fukushima Daiichi NPP waste based on the existing disposal concepts:

- provisional classification of waste into existing disposal concepts based on the safety assessment and radionuclide inventory information for Fukushima Daiichi NPP waste;
- extraction of successful conditions, which are the parameter values that comply with the analytical condition to ensure the disposal safety, based on sensitivity analysis;
- extraction of the rational countermeasure to improve the safety of the disposal concept based on the successful conditions;
- extraction of appropriate candidates for disposal concepts and assessment methods.

The study for meeting these requirements will be conducted in the investigation of the disposal study. The flow of the disposal study is shown in Figure 8.3.

Technical information, such as the requirements of appropriate disposal concepts and technical proposals for developing appropriate disposal concepts are important outcomes of the disposal study undertaken as part of the “examination of waste stream” project. Also, the impacts of potentially safety-relevant radionuclides are extracted under various conditions. This is an important outcome and is very useful for the selection of high-priority nuclides for further study.

Figure 8.3. Flow of the disposal study

Source: Ohi, 2014.

Status of the waste disposal study

Provisional safety assessment based on the existing disposal concepts has been conducted to select disposal concepts applicable to Fukushima Daiichi NPP waste. Identification of important nuclides based on the safety assessment has been carried out. Also, sensitivity analyses have been implemented for the investigation of barrier safety performance (extraction of successful conditions).

Existing disposal concepts of Japan are shown in Figure 6.1.

Waste storage plan at Fukushima Daiichi NPP

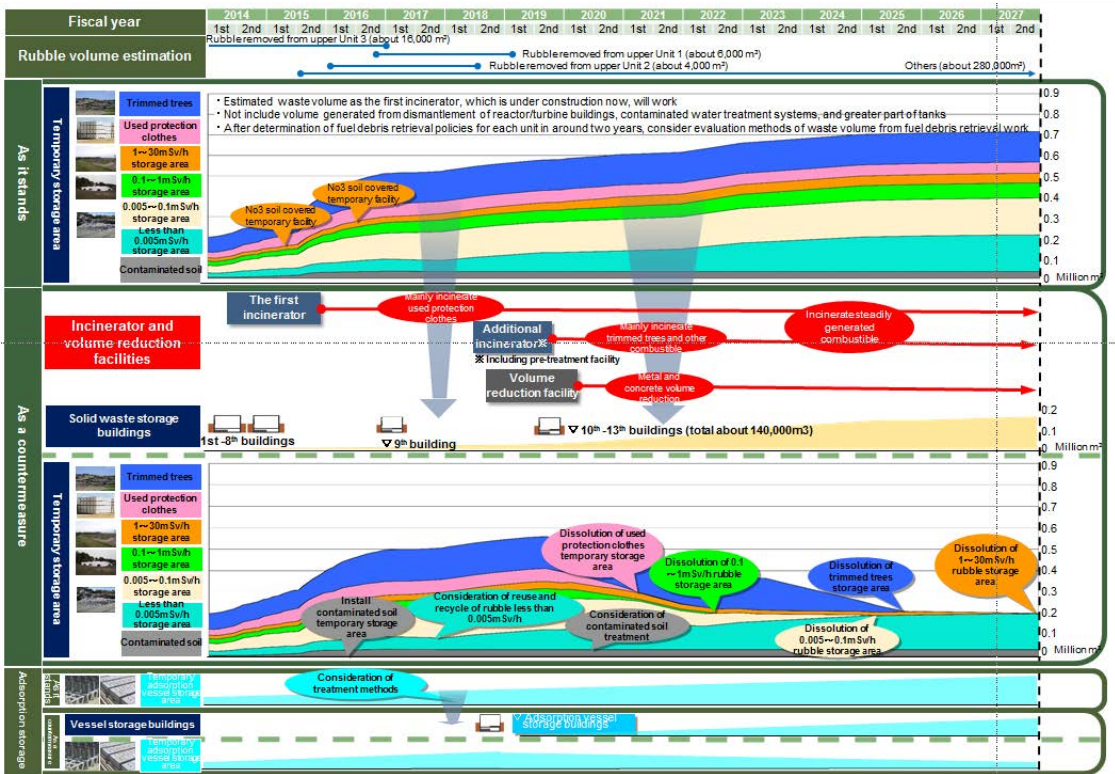
TEPCO has announced an outline of the plan to move waste from temporary outside storage areas to inside storage, installing additional volume reduction and storage facilities. The order of movement is planned to be based on effect on-site boundary dose which is originated from each temporary storage area (Figure 8.4). During movement, as far as possible, combustible waste will be incinerated, and metal and concrete waste will be volume reduced before being stored inside additional buildings. TEPCO will revise this predictive plan, as appropriate, according to how work towards decommissioning would proceed and how much waste would be generated in the future.

The waste volume of “as it stands” cases (Figure 8.4) is estimated to take into account the first incinerator, which started to work from February 2016. Also, this waste volume does not include volume generated from dismantlement of reactor/turbine buildings, contaminated water treatment systems, and the greater part of tanks. In addition, evaluation methods of waste volume from fuel debris retrieval work will be considered after determination of fuel debris retrieval policies for each unit in around two years.

There is about 300 000 m³ waste on-site at this point and will be about 720 000 m³ in about ten years, including trimmed trees, used protection clothes and rubble, as the situation now stands. The first incinerator and No. 9 solid waste storage building are

under construction. Additionally, a metal and concrete volume reduction facility and solid waste storage buildings are planned to be installed. The volume of additional solid waste storage buildings, which are planned to be four buildings from No. 10 to 13 buildings at this point, is planned for about 140 000 m³. These planned facilities will be sufficient for the dissolution of outside temporary storage and most of rubble will be stored inside buildings.

Figure 8.4. Image of solid radioactive waste storage at TEPCO's Fukushima Daiichi nuclear power plant



Source: TEPCO, 2016.

Regarding other waste which is not included in the 140 000 m³, TEPCO will consider how to dissolve temporary storage areas of both less than 0.005 mSv/h rubble and contaminated soil from this time. Concerning rubble of less than 0.005 mSv/h, TEPCO will also consider how to reuse and recycle for contaminated soil; careful prediction of contaminated soil volume and examination of treatment are needed in this case. Finally, concerning secondary waste from contaminated water treatment systems, storage buildings are planned to be installed. By the time secondary waste begins to be moved from temporary storage to buildings, volume reduction treatment or stabilisation treatment will have been considered.

Future issues

In order to improve confidence in the selection of disposal concepts applicable to Fukushima Daiichi NPP waste, the importance of resolving the following issues has been pointed out in the disposal study:

- reduction of uncertainty in the inventory estimation obtained from the characterisation study;

- demonstration of sufficiency and adequacy of the calculation cases in the analyses.

Future information and knowledge obtained from waste characterisation and processing, regarding conformity with regulations, are accumulated and are incorporated into this disposal study. Furthermore, the assessment based on such information and knowledge will be carried out repeatedly in order to develop applicable disposal concepts for Fukushima Daiichi NPP waste.

Lessons learnt from Fukushima Daiichi accident

Concerning solid radioactive waste storage on the Fukushima Daiichi site, continuous work is being undertaken to manage and arrange storage facilities of solid radioactive waste. In the future, TEPCO will install additional radioactive waste storage facilities, as required.

The “basic concept of processing and disposal for solid radioactive waste” should be developed up until FY 2017. Prospects of a processing/disposal method and a technology related to its safety should be made clear by around 2021.

Windscale

Timescales of waste arisings and overview of waste destinations

The earliest radioactive waste resulting from the Windscale fire was generated in 1957 (Arnold, 2007). Radioactive waste continues to be produced at the present time, and further waste will be produced in the future. The important point with regard to this report is that all accident-related waste has been managed using the waste routes in use at the time for normal operational and decommissioning waste. No new storage or disposal facilities were developed specifically for fire-related waste. Any fire-related liquid waste was discharged to sea via the marine discharge pipeline in accordance with pertaining authorisations.

Early years

In the years immediately following the fire, lower-activity solid waste from clean-up operations was disposed of in shallow unlined burial trenches located on the Windscale site (NEA, 2014). Waste was placed into these trenches with little prior characterisation; review of on-site operations and contemporary documents leads to the conclusion that the trenches contain waste that would be considered LLW today (NDA, 2015). The “Windscale trenches” were in operation before the fire and were the forerunners of shallow disposal trenches constructed at the site now known as the Low Level Waste Repository (LLWR), which is located approximately 5 km south-east of the Sellafield site. They are likely to have been constructed and operated in a similar manner; loose waste is likely to have been tipped into the trench and then covered by fill.

In the years immediately following the fire, any solid higher-activity waste produced would have been transferred to existing ponds and silos for storage on the Windscale (now Sellafield) site. Retrieval of waste and decommissioning of these ponds and silos (now designated as the Sellafield “legacy” ponds and silos) is now the top priority for the UK’s nuclear decommissioning programme (NDA, 2011).

The present time

In the UK, a range of disposal routes is available for lower-activity solid radioactive waste. LLW is defined as having less than 4 GBq/te total alpha activity and 12 GBq/te total beta activity. The lower limit of LLW is specified as being greater than the “out of scope value” of Environmental Permitting (Amendment) Regulations 2011. Other than limits on total alpha and total beta activity, there are no radionuclide-specific limits on the definition of LLW. In the United Kingdom, the LLWR is the UK’s primary facility for the permanent disposal of solid low-level radioactive waste.

The lower part of the LLW activity range is a sub-category designated VLLW. This is radioactive waste with a maximum activity of 4 MBq/te¹, and it can be disposed of in certain licensed landfills. The capacity of such landfills is much greater than the capacity of the LLWR, and the cost of such disposals is substantially lower. From the perspective of both cost and “capacity” (i.e. making the best use of the finite capacity of LLWR), it is advantageous to correctly consign VLLW to licensed landfills where possible.

Geological disposal of higher-activity radioactive waste became government policy in 2008. Radioactive Waste Management Ltd, a wholly owned subsidiary of NDA, is responsible for implementing this policy by delivering a GDF and provision of radioactive waste management solutions. The siting process for a GDF is currently at an early stage; therefore, currently there is no disposal route for higher-activity solid waste in the United Kingdom. In the meantime, conditioned and packaged higher-activity solid waste are stored on UK nuclear sites (including at Sellafield) in interim storage facilities. As discussed in Section 8.1, it is a UK requirement that all proposed radioactive waste streams and new packaging proposals are assessed (the “disposability assessment” process) to minimise the possibility that the conditioned and packaged waste is incompatible with the selected geological disposal concept. This is the mechanism by which Sellafield Ltd can demonstrate that all higher-activity waste produced as a consequence of the Windscale fire can be disposed of at a future UK GDF.

Sellafield Ltd, the current operator of the Sellafield site, has undertaken an extensive programme of land quality investigations to characterise and assess the radiological and environmental impacts of radioactively contaminated land on the Sellafield site (Sellafield Ltd., 2012). A routine groundwater monitoring programme at the Sellafield site is also undertaken (see Sellafield Ltd, 2015a and references therein). Through these projects, which commenced in the 1970s, Sellafield Ltd has determined that some radionuclides have migrated from the former low-level burial trenches into the surrounding soil and groundwater.

Sellafield Ltd has determined that the radiological consequences of contaminated land on the Sellafield site, including that generated from the burial trenches, is low (Environment Agency, 2009). The company is implementing a contaminated land management strategy that involves continuing characterisation and monitoring of the site, to better understand the distribution of subsurface soil and groundwater contamination, and modelling, to build understanding of the migration processes and their radiological and environmental impacts (Sellafield Ltd, 2015b). In the future, Sellafield Ltd will undertake large-scale decommissioning of remaining facilities at the site. Remediation of some areas of contaminated land is likely to be a component of this final stage of decommissioning.

Lessons learnt from Windscale case

All accident-related waste has been managed using the waste routes in use at the time for normal operational and decommissioning waste. No new storage or disposal facilities were developed specifically for fire-related waste. Any fire-related liquid waste was discharged to sea via the marine discharge pipeline in accordance with pertaining authorisations. Given this, there are no specific lessons learnt for accident waste.

1. An exception is made for waste containing tritium; the maximum concentration limit for tritium is 40 MBq/te.

8.3. Principal lessons learnt

The principal lessons learnt from the case studies are listed below:

- From the experience of TMI-2 and Fukushima, it is recognised that it is very difficult to justify discharge of liquids into the environment after an accident, even if such discharges would be within operational discharge limits. Stakeholder concerns are the main issues.
- From the experience of major accidents, it has been recognised that the larger volumes of lower-activity waste resulting from accidents exceed the existing capacity of disposal sites for operational and normal decommissioning waste.
- The lack of deep geological disposal facilities in most member countries means that long-term storage of higher-activity solid waste will be an important component of any solution.
- Some of the case studies indicate that storage facilities opened at or shortly after the time of the accident have been developed under extreme time pressures, which have led to limited consideration of their long-term safety and environmental performance. Given this, unless activities of disposal waste are very low, it is highly unlikely that such facilities could be redesignated as disposal sites.
- There is clearly a balance to be struck in storing solid radioactive waste in the aftermath of an accident between “quick” solutions, which immediately reduce radiological doses to the workforce and improve overall safety, and “robust long-term” solutions that minimise the rework necessary to enable long-term storage and disposal of the waste.

8.4. Recommendations

Our principal recommendation to address these lessons learnt is to plan for an accident in terms of long-term waste storage, discharges to the environment and final disposal. It is better to prepare than to improvise in the aftermath of an accident. Key areas where plans should be developed include the following:

- Clear responsibilities between parties need to be agreed and developed within an implementer-regulatory framework. International harmonisation of approaches would be beneficial. A new NEA task group to provide recommendations on preparing for an accident would be beneficial in this respect.
- With forward planning, it would be possible to explore possibilities for on-site disposal of some types of accident waste.
- Plans should also include those for monitoring and assessment of temporary storage sites as soon as is practicable in order to understand their impact on the environment.
- Forward planning would help to address some of the issues associated with developing new near-surface disposal facilities for accident waste, in particular issues arising from the slightly elevated actinide activities expected in some lower-activity accident waste. Near-surface disposal of such waste may not be permitted under existing national waste classifications. Consideration should be given to developing a safety case to allow such accident waste to be disposed of in near-surface facilities.

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Conclusions

The case studies presented in this study offer substantial information on the history of accident site management and lessons learnt, leading to many potentially helpful recommendations. The material provided includes information on:

- state-of-the-art techniques and experience of waste characterisation and classification, including application after major accidents;
- regulatory supervision: regulations, regulatory guidance and regulatory procedures, e.g. review of safety cases;
- the application of international recommendations, standards and guidance.

Every accident is different. The details of any post-accident (after emergency) scenario are unpredictable and specific to the prevailing circumstances. Responding to them requires elements that are not within the usual experience of conventional utility and service management organisations. Managing decommissioning and radioactive waste after a major accident may require a different approach from that used following normal planned operations.

Centralised authority and stakeholder involvement

There is a need for a centralised authority to manage the situation, for example, a high-level governmental commission, to co-ordinate and oversee the planning and implementation of effective measures. Government, industry and research institutions must work co-operatively to plan and implement these measures.

This authority will need to develop a comprehensive strategy with clear objectives to manage the situation, taking into account the interests of a wide range of stakeholders. Effective stakeholder engagement processes are needed to identify those interests.

Implementation strategy

A plan is needed to implement the strategy through a series of tasks designed to meet the stated objectives, identifying who is responsible for implementing each task and providing the powers and resources necessary to those with responsibility for implementation.

A major component of the strategy is connected to the establishment of a regulatory framework for decommissioning and radioactive waste management. This should be based as far as possible on the existing framework for these activities, but specifically modified to account for the special factors linked to the prevailing circumstances arising from the accident, as identified through waste characterisation and other processes.

Special factors include the need to set appropriate reference levels as well as derived standards and monitoring procedures, application of which should result in meeting those reference levels and the ability to demonstrate compliance with them.

A heavily project-focused approach is more effective than a large functional organisation of engineers and designers responsible for small bits of several projects.

While redundancy in organisational functions is expensive and difficult to manage, some degree of redundancy is prudent to ensure that all options and potential problems can be considered.

There is likely to be a need for iteration of the strategy, with more detail added at each stage taking account of the information, including radioactive waste characterisation data, obtained from the previous stage. Responsibilities for implementation and resourcing of tasks in each stage may need to be updated. In the early stages, it may be useful to pursue flexible/parallel approaches. A careful step-by-step approach is in any case strongly advised, so as to reduce the chance of creating legacies requiring future management.

However, it is noted that excessive caution may delay appropriate timing of decisions. Examples include delay of return to normal land use, even though it would be safe to do so, or delay in the introduction of appropriate restrictions, resulting in extended continuation of risky conditions, as well as potential costs increases. This problem should be acknowledged, alongside the need for balance, which should be achieved with the support of stakeholder engagement.

In developing an iterative strategy, it is important to leave time to obtain regulatory approval. Public access to land and normalisation of land use is urgent, providing many hard-to-measure benefits to those who normally occupy the land. The contamination levels can be expected to be relatively low off-site so that remediation work is very extensive but not complex from a technical and safety point of view. However, once the emergency is declared over, decommissioning of the damaged building and remediation of the nuclear site itself is not so urgent. It may also be massively more hazardous and present further risks of repeat accidents. The need to take time for this work should therefore be anticipated, as has been the case at Three Mile Island 2 (TMI-2), the Chernobyl nuclear power plant (ChNPP) and Windscale Pile.

Optimisation

Optimisation is an important aspect of radiological protection and is best done taking into account social and economic factors, not just radiological factors, (e.g. meeting reference levels). Again, the process should be supported by stakeholder engagement. It should be noted that solid waste minimisation, as has been recommended, could be achieved by discharging more waste to air and water, for example by incineration or dissolution, or by creating higher-level waste that is not suitable for shallow land burial. It is not entirely evident that such discharge is the optimum management method, so the choice would need to be supported by a relevantly structured assessment. More generally, it can be noted that the minimisation of one detrimental impact is always likely to result in another detrimental aspect not being minimised to the same extent; and hence the need for a holistic view of optimisation, both as developed in radiological protection and as would be more widely understood by stakeholders.

Storage and disposal

In addition to large quantities of fuel debris, the remediation and decommissioning response to an accident of the type at the Fukushima Daiichi NPP is likely to generate radioactive waste that exceeds limits for near-surface disposal or intermediate depth disposal. This waste needs to be appropriately stored and stabilised until a final disposal solution is developed.

The large quantity of waste created by an accident may exceed existing radioactive waste disposal capacity or be of a waste class for which a disposal solution is not currently available. It may be necessary to create interim stores, but they should be

designed taking into account that final disposal will be needed in due course, and may need to remain effective for extended periods of time while sites for final disposal are identified and licensed. The accident site may not be the location to site this interim storage facility.

Safety analysis

Safety analysis, radiological and environmental impact assessments are necessary to support the identification of priorities, identify feasible management options and select preferred options from feasible alternatives. This process needs to be technically underpinned, but must be informed by stakeholder engagement, particularly as regards local conditions, but also so that the assessments address issues of interest to stakeholders.

These analyses must, to the extent possible, be based on existing regulations and regulatory guidance. Only in exceptional circumstances and based on a safety case that demonstrates compliance with the safety basis of the applicable regulation(s) should exemption from these criteria be permitted.

Thus, the design and content of these analyses and assessments should be specific to the purposes of the assessments, including the interest of the intended audience for each analysis or assessment.

International co-operation

Further development of plans for international co-operation in the event of a major accident would be useful, and could include:

- Further guidance on the application of international recommendations, standards and guidance in the post-emergency phase of a major nuclear accident.
- Pre-planning guidance on decommissioning and radioactive waste management that considers:
 - What planning can be done in advance?
 - What planning cannot be performed until the parameters of the accident are understood?
 - What is the scope for sharing characterisation resources, staff and equipment nationally and internationally?
- Guidance on:
 - the transition from emergency response to normal radiation exposure regulations;
 - stakeholder engagement, with an emphasis on later stages of recovery;
 - communication processes;
 - how to address chemicals alongside the radiological risks.

Annex 1. List of members of the Expert Group on Fukushima Waste Management and Decommissioning R&D (EGFWMD)

Dr Per Strand (Chair)	Norway
Dr Gérard Laurent (Vice Chair)	France
Mr Hiroshi Rindo (Vice Chair)	Japan
Ms Christine Georges	France
Mr Eiichiro Ito	Japan
Mr Norikazu Yamada	Japan
Dr Iuri Iablokov	Russia
Dr Tatiana Kilochytska	Ukraine
Dr Nick Jefferies	United Kingdom
Mr Jim Byrne	United States
Dr Michael Siemann	NEA/RAD
Mr Toshiyuki Koganeya	NEA/SAF
Mr Hiroomi Aoki	NEA/RAD (at the time of drafting the report)

Annex 2. List of abbreviations and acronyms

ALARA	As low as reasonably achievable
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
ChNPP	Chernobyl nuclear power plant
DOE	Department of Energy (United States)
DQO	Data quality objectives
EDR	Exposure dose rate
ENSDF	Engineered Near-surface Disposal Facility
EPRI	Electric Power Research Institute
FCM	Fuel-containing materials
GDF	Geological disposal facility
GPU	General Public Utilities
HLW	High-level waste
ICSRM	Industrial Complex for Solid RW Management
ILW	Intermediate-level waste
INF	Irradiated nuclear fuel
JCG	Joint Coordination Group for SIP Licensing
LFCM	Lava-like fuel-containing materials
LILW-LL	Low- and intermediate-level long-lived waste

LILW-SL	Low- and intermediate-level short-lived waste
LLW	Low-level waste
L RTP	Liquid Radioactive Waste Treatment Plant (Chernobyl)
LRW	Liquid radioactive waste
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual (US NRC)
MOU	Memorandum of Understanding
NEPA	National Environmental Policy Act
NEA	Nuclear Energy Agency
NMS	Neutron monitoring system
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission (United States)
NRS	Nuclear and radiation safety
NSC	New safe confinement
PDMS	Post-defueling monitored storage
PEIS	Programmatic environmental impact statement
RA	Regulatory authority
R&D	Research and development
RFSW	Retrieval facility for solid waste
RW	Radioactive waste
RWM	Radioactive Waste Management Ltd
RWMC	Radioactive Waste Management Committee (NEA)
SDRW	Site for disposal of radioactive waste
SDS	Submerged demineraliser system

SIP	Shelter Implementation Plan (Chernobyl)
SNF	Spent nuclear fuel
SNM	Special nuclear material
SNRIU	State Nuclear Regulatory Inspectorate of Ukraine
SO	Shelter object
SRW	Solid radioactive waste
SSR	Sorting and size reduction
STLRW	Site for temporary localisation of radioactive waste
STS	Site of temporary storage
SWPF	Solid waste processing facility
SWSF	Solid radioactive waste storage facility
TMI-2	Three Mile Island 2
TMIPO	Three Mile Island Program Office
TS	Technical specification
TUE	Transuranium elements
EPA	Environmental Protection Agency (United States)
VLLW	Very low-level waste
WAC	Waste acceptance criteria
WHPF	Waste Handling and Packaging Facility (Three Mile Island, United States)

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Management of Radioactive Waste after a Nuclear Power Plant Accident

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. The group was formed with specialists from around the world who had gained experience in waste management, radiological contamination or decommissioning and waste management R&D after the Three Mile Island and Chernobyl accidents. This report provides technical opinions and ideas from these experts on post-accident waste management and R&D at the Fukushima Daiichi site, as well as information on decommissioning challenges.