IAEA Nuclear Energy Series
No. NW-T-2.10

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Decommissioning after a Nuclear Accident: Approaches, Techniques, Practices and Implementation Considerations

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DECOMMISSIONING AFTER A NUCLEAR ACCIDENT: APPROACHES, TECHNIQUES, PRACTICES AND IMPLEMENTATION CONSIDERATIONS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA NUCLEAR ENERGY SERIES No. NW-T-2.10

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FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world." One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property". The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

Following the accident at the Fukushima Daiichi nuclear power plant, a Ministerial Conference on Nuclear Safety in 2011 adopted a declaration that requested the IAEA to develop a draft action plan on nuclear safety. In September 2011, the IAEA Action Plan on Nuclear Safety was adopted by the IAEA's Board of Governors and unanimously endorsed by the IAEA General Conference. The goal of the Action Plan was to strengthen nuclear safety worldwide.

Item No. 10 of the Action Plan on Nuclear Safety was concerned with ensuring the ongoing protection of people and the environment from ionizing radiation following a nuclear emergency. Under this action, the IAEA Secretariat was requested, inter alia, to collect experience on approaches, techniques, tools and equipment to deal with the cleanup of affected sites and decontamination and decommissioning of facilities after an accident, and to make this experience and information available to Member States. This publication covers post-accident decommissioning aspects including important lessons learned for the safe implementation of the on-site activities.

Three consultancy meetings were held and included the participation of a number of international experts to draft, review, amend and finalize this publication. Additional valuable contributions to the development of this publication were gratefully received from the chairperson of consultancy meetings, Ch. Negin (United States of America).

The IAEA officer responsible for this publication was V. Michal of the Division of Nuclear Fuel Cycle and Waste Technology.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The IAEA regularly provides guidance and information on decommissioning through publications in the field, namely through the Nuclear Energy Series, Technical Reports Series, Safety Standards Series, Safety Reports Series and TECDOC Series. However, only a few of these publications have addressed the various decommissioning aspects to be considered after a nuclear accident [1–7]. New publications were prepared and published in light of the Fukushima Daiichi accident based on the requirements of the IAEA Action Plan on Nuclear Safety [8–10], conclusions of the International Experts Meetings organized by the IAEA [11] or as an initiative of other international organizations, such as the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA) [12]. The interactions between decommissioning and environmental remediation after an accident need to be highlighted, such as the large amount of waste generated and the need for its appropriate management.

1.2. OBJECTIVE

The main objective of this publication is to describe differences in post-accident situations compared to normal decommissioning (i.e. decommissioning after a planned final shutdown) and also to identify significant decision factors. The publication offers an overview of the approaches, techniques, practices and implementation considerations to deal with decommissioning activities after an accident, based on lessons learned from past events. This publication may be useful to Member States to further their understanding of the complexity of post-accident decommissioning and to potentially support their planning of the technical measures necessary to deal with this issue. Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.3. SCOPE

This publication is focused mainly on the on-site decommissioning aspects of a technical nature that need to be addressed after a nuclear accident (International Nuclear and Radiological Event Scale (INES) Level 4–7). Non-technical issues, such as policies and strategies, project planning, organization and management are also covered. The collection of experience on approaches, techniques, practices and implementation considerations is based on lessons learned from past events, including the Fukushima Daiichi nuclear power plant (NPP) accident.

In terms of nuclear facility type, this publication addresses decommissioning of NPPs after an accident. Many aspects covered in this publication are also relevant for non-reactor nuclear facilities and legacy nuclear facilities.

The treatment and storage of large volumes of waste generated during the implementation of on-site decommissioning activities are also considered in this publication; however off-site remediation after a nuclear or radiological accident is out of the scope.

1.4. STRUCTURE

The structure of this publication takes the reader through a variety of post-accident decommissioning aspects that should be considered. Section 2 introduces an overview of decommissioning following an accident with a set of definitions and terms described. Section 3, on post-accident policy and strategy, highlights relevant top level considerations and Section 4 describes project planning and management. Section 5 deals with the key post-accident operational functions. Section 6 is focused on damaged fuel and fuel debris management. Section 7 provides an overview of the considerations and relevant aspects towards the final decommissioning. Conclusions are summarized in Section 8 followed by the list of references cited in the main text. Illustrative cases and examples are given in Annex I on Chernobyl NPP transformation into the ecologically safe system, in Annex II on the role of the United States Department of Energy (USDOE) at Unit 2 of Three Mile Island (TMI) for activities towards decommissioning and in Annex III on damaged fuel and radioactive waste management. Annex IV proposes considerations related to the post-accident use of the international structure for decommissioning costing (ISDC).

2. OVERVIEW OF DECOMMISSIONING FOLLOWING AN ACCIDENT

Definitions and rating of nuclear and radiological events were proposed in INES, which was developed in 1990 and refined later in 1992. The INES User's Manual was issued in 2001 and revised in 2008 to consolidate additional guidance and clarifications, and to provide examples and comments on the continued use of INES [13]. Events are classified on the scale at seven levels. Levels 4–7 are termed 'accidents' and Levels 1–3 'incidents'. Events without safety significance are classified as 'below scale/Level 0'. Events that have no safety relevance with respect to radiation or nuclear safety are not classified on the scale.

Short descriptions of examples of relevant nuclear accidents — Fukushima Daiichi Units 1–4, Chernobyl NPP Unit 4, TMI-2, A1 NPP, Windscale Pile 1 — are provided in several IAEA publications [1–7] and in the following sections and annexes of this publication. Non-reactor nuclear facility accidents in the Russian Federation and the United Kingdom can be found in other publications [14, 15].

Consistent definitions related, among other things, to accidents and decommissioning are offered in the IAEA glossaries [16, 17]. The glossaries and Nuclear Energy Series No. NW-T-2.7 [7] include a description of basic terms such as stabilization, decommissioning (after an accident), on-site versus off-site remediation/clean up, decontamination, damaged fuel and fuel debris, corium, radioactive waste management, decommissioning interim and final end state(s). Although the background and meaning of a variety of specific terms is addressed in the relevant sections and annexes of this publication, the reader may find the following short summary useful:

- — Accident: Any unintended event, including operating errors, equipment failures and other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety;
- — Stabilization: Activities implemented during and subsequent to the emergency response that are needed prior to beginning the intensive post-accident cleanup [7];
- — Characterization: Determination of the nature and activity of radionuclides present in a specified place;
- — Recovery: The process of return of an affected facility and site to a state of normality after a disaster;
- — Decommissioning: Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility;
- — Remediation: Any measures that may be carried out to reduce radiation exposure from existing contamination of land areas through actions applied to the contamination itself (the source) or to the exposure pathways to humans;
- — Fuel debris: Any fuel rod or assembly material that cannot be retrieved as part of a fuel assembly [18];
- — Corium: A melted mixture of components that can consist of nuclear fuel; fission products; control rods; structural materials from the affected parts of the reactor; products of their chemical reaction with air, water and steam; and, if the reactor vessel has been breached, concrete from the structure of the reactor space.

There are typically three phases associated with a post-accident situation at nuclear facilities — stabilization, recovery and decommissioning [19].[1] Stabilization refers to the immediate aftermath of a nuclear accident and it implies controlling of conditions so that impacts to the environment and the general public are minimized. Recovery entails the planning and implementation of activities to limit (and subsequently reduce) the extent of abnormal conditions, and preparation of the plant for the achievement of a longer term, safer configuration. Recovery can be viewed as a precursor to decommissioning.

[1] In the United Kingdom, a fourth phase is also considered, referred to as 'quiescent.' This is the phase during which the only significant activities are associated with the safe and secure storage of materials, pending the availability of final disposal routes and decisions on the site end point [19].

Due to unpredictable conditions and evolution of a severe accident, it is challenging to define specific issues that can be expected during decommissioning of a facility just after an accident. In general, these could be related to the physical state and integrity of structures, systems and components (SSCs), radiation and contamination levels, generation of abnormal wastes and their management, safeguards issues, regulatory responsibilities, availability of decontamination and dismantling techniques for work in hostile environments, records and data management, availability of resources, and organization and management, including stakeholders. It is later in the planning for or in the implementing of decommissioning that the full extent of actual or potential issues are identified and tackled. An initial (provisional) safety analysis would identify major issues that would need to be addressed during the recovery phase to avoid decisions or actions potentially complicating an effective implementation of further decommissioning activities.

An overview of IAEA support for decommissioning and remediation activities after a nuclear accident, such as the development of focused publications, the implementation of a project or direct cooperation with some Member States, is given in Ref. [20]. It is highlighted that the decommissioning and environmental remediation after an accident includes complex technical, safety, managerial, organizational, societal, environmental and economic issues. These need to be effectively addressed and resolved to mitigate the consequences of the accident and to demonstrate that decommissioning and on-site/off-site remediation can be completed even in a difficult post-accident situation.

In this regard, the IAEA international project Decommissioning and Remediation of Damaged Nuclear Facilities (project DAROD) initiated in 2015 should be mentioned to provide lessons learned and benefit from the experiences derived from the decommissioning and remediation of such facilities [21]. The scope of the project spans from the time the emergency is declared over until the decommissioning and remediation of the facility is completed. DAROD project work is being undertaken by three working groups on regulatory issues, technical issues and institutional framework and strategic planning.

The OECD/NEA also addresses decommissioning and radioactive waste management after an accident within its comprehensive programme [22].

3. POST-ACCIDENT POLICY AND STRATEGY

Reference [23] addresses policies and strategies for the decommissioning of nuclear facilities in a general way applicable to all types of nuclear facilities such as NPPs, research reactors, nuclear fuel cycle facilities and laboratories using radioactive material. It refers to policies and strategies for decommissioning after final shutdown in a normal way. Specific emphasis is not given to the decommissioning following post-accident shutdown of a nuclear facility.

The Fukushima Daiichi NPP accident reiterates the need for the development of post-accident decommissioning policies and strategies. Ideally, these should be part of a national decommissioning policy and strategy developed before or during operation of nuclear facilities or soon after an accident occurs. A few references that have already touched on this issue are available (see Refs [2, 11, 24]) and more guidance is expected to be provided in the future (e.g. a particular outcome of the IAEA project DAROD [21]). The following sections offer views regarding the establishment of a post-accident decommissioning policy and strategy, based mainly on the variety of technical aspects.

3.1. POLICY

The development of a national policy at the governmental level with a statement of the objective for managing the accident, the post-accident recovery, site cleanup, waste management and subsequent decommissioning in the aftermath of a civil nuclear accident needs to be considered [10]. The policy should indicate the timescale in which this objective can be met and specify roles and responsibilities of the involved authorities and organizations. The policy may also cover other top level considerations related, for example, to overall long term project management or communication with the public.

3.2. STRATEGY FOR POLICY IMPLEMENTATION

An essential part of strategic planning is to ensure that an emergency plan for design basis accidents is in place by the nuclear site licensee. An emergency plan is a normal site licensing requirement and needs regulatory approval. The plan also describes contingent measures to be taken in light of a beyond design basis accident. Normally, the implementation of site emergency plans should be rehearsed and witnessed by observers from the regulatory body at the site on a routine basis. Hence, strategic planning needs to be developed to inform key activities to be conducted during the early post-accident stabilization phase. The strategy needs to be consistent with, and assist, the implementation of the national policy.

Beyond the phase of dealing with the immediacy of the accident, it is necessary to establish a long term strategy that provides direction for forward post-accident cleanup operations to include:

— Maintenance of a condition of subcriticality, providing for decay heat removal from any damaged fuel present;
— Maintaining and strengthening containment to prevent further releases of activity by use of physical barriers and/or filtered ventilation systems;
— Stabilizing structures to prevent collapse of further degradation hence limiting further potential releases;
— Removal of mobile sources of contamination wherever practicable (e.g. contaminated water to prevent further spread);
— General decontamination as appropriate to extend working time of decommissioning staff and introduction of measures to prevent recontamination of cleaned areas;
— Identification and preparation of access routes to support stabilization and future decommissioning activities;
— Identification of options for the management of wastes that may well require various treatment options and be present in quantities very much larger than those from normal operations;
— Development of procedures for the radiological protection of workers, the public and the environment.

The availability of funds as well as technical and human resources are key issues for the development of any decommissioning strategy [23]. Post-accident decommissioning is very demanding and significant extra resources are needed to cover expenses that were not expected or planned before the accident. International support may also be needed varying from expert support provided by international organizations or specialized companies to the considerable financial support of donors.

3.2.1. Resource efficiency

The resources that are immediately available to conduct the post-accident project are likely to be limited in the short term to the owner/operator staff assigned to the plant and home office support personnel. Detailed plant knowledge will be essential in formulating a recovery project team. For this reason, additional expertise may be needed from within the organizations that originally designed the systems and that managed construction of the plant.

Specialized resources that may be needed may be obtainable from the following:

— Government institutions with experience in management of nuclear and radioactive materials, such as laboratories for analysing highly radioactive samples and facilities for testing conditioning process concepts;
— Companies worldwide with capabilities for specific situations, such as systems for caesium and strontium removal from water, decontamination equipment and high radiation measurement instruments;
— Individuals with experience that makes them experts in the subject matter can be found and brought in as consultants or advisors on an as needed basis;
— The IAEA, through independent reviews of plans, technical methods, operations, and other aspects of the projects resulting in assessments and recommendations to the plant owner/operator.

Effective management of resources requires prioritization. In the short term, there will be essential and urgent activities that are of the highest priority over requests for less important issues. In the case of the TMI-2 accident, the primary technical priorities were to achieve cold shutdown of the reactor, maintain reactor system stability and protect the health and safety of the public. In the case of the Fukushima Daiichi accident, the top priority was the safety of local citizens and workers. The Chernobyl NPP accident was very special in this regard as its radiation consequences were extreme and heavily affected follow-up on-site and off-site actions. For the Windscale Pile 1 accident, the principal objective was to extinguish the fire in the graphite core of the reactor and this took several days to bring under control. Further technical details regarding this topic are provided in this publication, mainly in Section 5 on post-accident key operational functions.

In addition to essential and urgent activities, there will be many requests and demands for information and additional actions. For example, research agencies will want characterization measurements and sample analyses that may be beyond what is required to support operation decisions. It is therefore extremely important for management to prioritize all activities that will consume limited resources.

The degree to which such requests and demands are to be considered beyond near term urgency need to be also managed carefully. Useful considerations regarding long term knowledge and information management can be found in Ref. [25].

3.2.2. Communications

A publicly acceptable communications programme, based on openness and transparency, should be developed to keep stakeholders fully informed and to maintain trust. It is essential that credible communication channels with the media, municipalities and other involved counterparts in the affected areas are established rapidly, otherwise speculation may develop on an unjustified basis.

The primary objective of communication is to inform those affected so that contingency measures can be applied rapidly (e.g. use of potassium iodide tablets to reduce thyroid doses from ^{131}I). Such measures assist in minimizing fear and anxiety by ensuring understanding of and compliance with protective actions. It is essential for the site operator to provide immediate, realistic and frequent briefings to the media/local community. To manage briefings, the site operator and key organizations may need to establish a communications office(s). In the case of the Fukushima Daiichi accident, the site operator, Tokyo Electric Power Company (TEPCO), created a new communications office.

In the case of Fukushima Daiichi and TMI-2, the development of a Roadmap [26] proved to be a highly effective tool in helping develop a common understanding. The use of a Roadmap is described in Section 3.3.

3.2.3. Harmonization of relevant organizations

Harmonization of approaches and activities initiated, for example, by national government, site operators, regulators and vendors (manufacturers) who are assisting in the post-accident decommissioning is of great importance. This harmonization may be facilitated through a coordinating committee and all relevant papers, reports and minutes of the meetings should be easily accessible to the public. Harmonization needs to be a significant part of an effective communication framework [11].

It is important to establish a comprehensive (total) management system to ensure implementation of the long term project activities through cooperation of various organizations, bodies and industries [7]. A top level management body needs to be appointed and duties and responsibilities clarified for all organizations involved.

In the case of Fukushima Daiichi, a council was established to accelerate the decommissioning work and enhance collaboration between on-site work and the Government (the Council for the Decommissioning of TEPCO's Fukushima Daiichi NPP).

In the case of TMI, the United States Nuclear Regulatory Commission (NRC) branches of nuclear reactor regulation, and inspection and enforcement were deployed to the site within days of the accident. During the first few months, NRC staff at the site played an important part in establishing and harmonizing crisis teams, and putting emergency systems in place.

3.3. STRATEGIC PLANNING/ROADMAP

The use of a Roadmap can be a highly effective tool to communicate the mid-term and long term points of a nuclear decommissioning programme. The Roadmap includes the phases of the project on the journey towards the completion of decontamination and safe storage/disposal of radioactive waste. Figure 1 illustrates an example of the TMI Unit 2 (TMI-2) Roadmap.

It is necessary to continually update the strategy, taking into account changes to the on-site situation. Regular updates to plans can increase understanding and promote public trust, thereby minimizing the impact of the accident and facilitating the determination of the end state [7]. Introduction of dates for the completion of particular tasks needs to be carefully considered to avoid unrealistic expectations that may lead to criticism of the overall strategy.

3.3.1. Post-accident programme and project phases

Effective programme and project management helps improve direction, control, costs, productivity and overall delivery of nuclear decommissioning projects.

One example includes the decommissioning of A1 NPP in Slovakia. The Bohunice nuclear site suffered an accident during refuelling in February 1977. After a relatively long preparatory time, the facility operator developed (and the regulator approved) a five stage decommissioning programme (see Fig. 2). It started with securing a safe radiological state (Stage I), decommissioning of the outer active objects and their reconstruction for decommissioning purposes, radioactive waste and contaminated soil management (Stage II) and decommissioning of technological parts, components and systems in the main production building with a graded approach (Stages III–V). The goals and the end of each stage are to be defined for each particular stage. Sufficient financial provision is a basic precondition for the beginning of each decommissioning stage.

Following the accident at Fukushima Daiichi NPP, the Government of Japan and TEPCO compiled a Roadmap in December 2011 and have been developing efforts towards decommissioning based on the Roadmap. Based on the progress of the work, the revised Roadmap was initiated in July 2012 [26]. An illustration of the progress and future challenges of the mid- and long term Roadmap towards decommissioning is shown in Fig. 3.

3.3.2. Optioneering

The post-accident situation significantly differs from the final shutdown and transition period after normal operation and therefore implementation of the decommissioning activities may not be straight forward. To be prepared for these situations, it may be necessary to develop strategic options [11].

For example, if a plan to remove fuel assemblies and debris in the Unit 2 spent fuel pool cannot be achieved using existing equipment, optioneering can address the issue by developing alternative plans to construct a container (a kind of containment) on the upper level of the reactor, when the building has sufficient seismic resistance. This is discussed in Ref. [26] and illustrated in Fig. 4.

3.3.3. Technology application

The value of adapting existing technologies should not be underestimated and the effectiveness may outweigh the costs (e.g. complex emerging remote handling solutions). In the case of the Chernobyl NPP and TMI accidents, adaptation of existing technologies proved highly effective in capturing on-site and in-reactor building images and data.

However, if adapting existing technology is not effective, the establishment of a short term, specialized R&D team might be implemented. This R&D team can be tasked with gathering international expertise and new technologies to help deal with unexpected post-accident situations. New technologies need to be demonstrated to prove their effectiveness before being used in the decommissioning programme. It is also essential that workers are fully trained to operate the new technologies.

Post-accident decommissioning activities at several damaged reactor units and legacy sites worldwide will be implemented for many decades in the future. Time overlapping may be of benefit in some cases from the viewpoint of sharing lessons learned or even transfer of specific technologies that were developed and used at one site to another site. The same applies for damaged fuel and fuel debris management and for radioactive waste management.

A comprehensive overview of decommissioning technologies and identification of further R&D needs, including applications specific for Fukushima Daiichi, can be found in Refs [27–29].

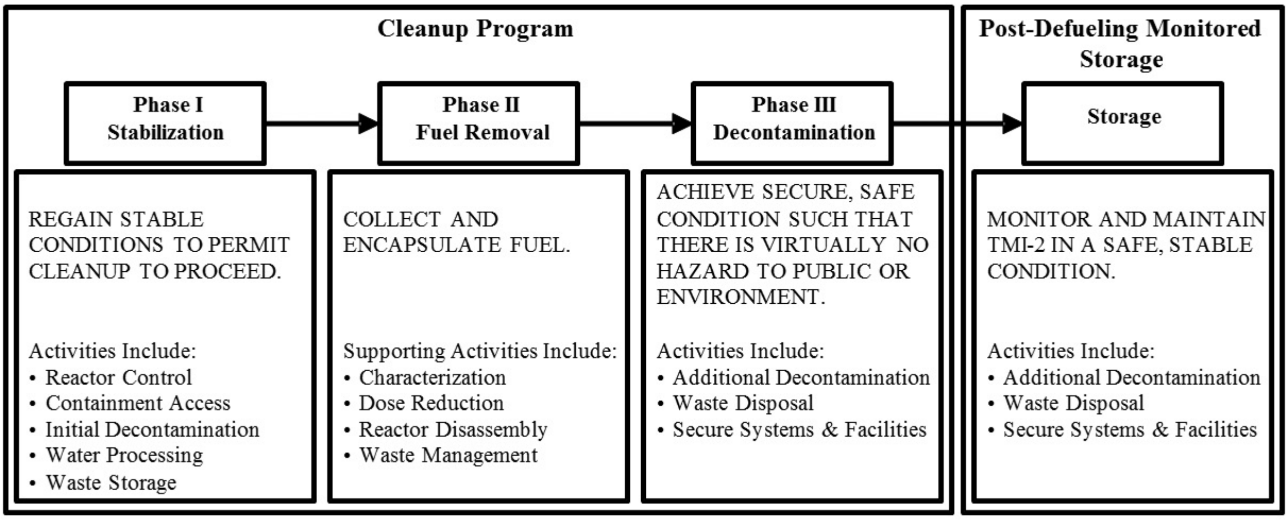


FIG. 1. TMI Roadmap. (Adapted from Electric Power Research Institute (EPRI), with permission).

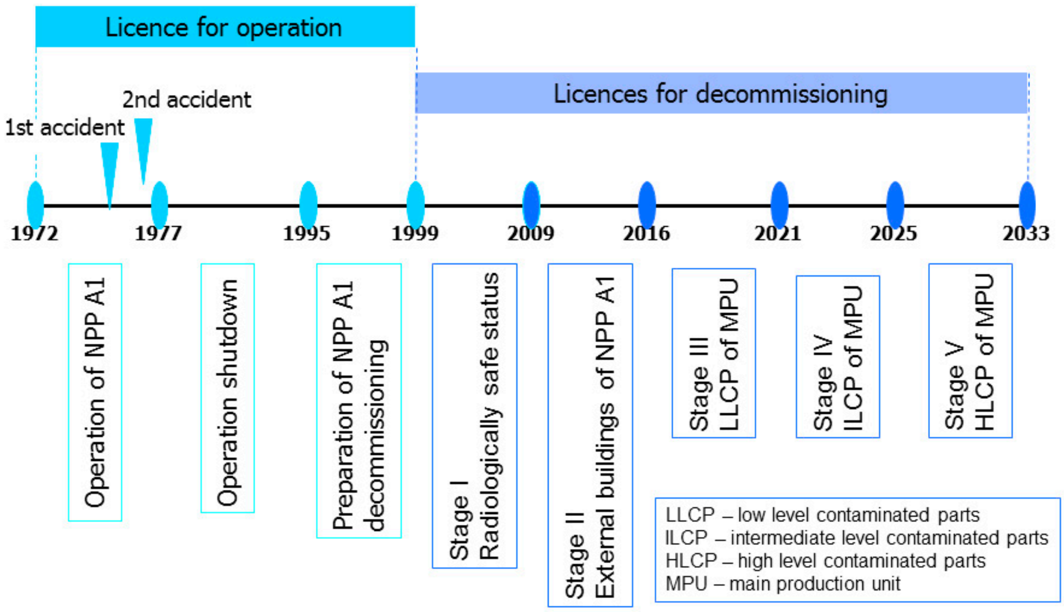


FIG. 2 . Stages of A1 NPP decommissioning. (Courtesy of JAVYS, a.s., Slovakia.)

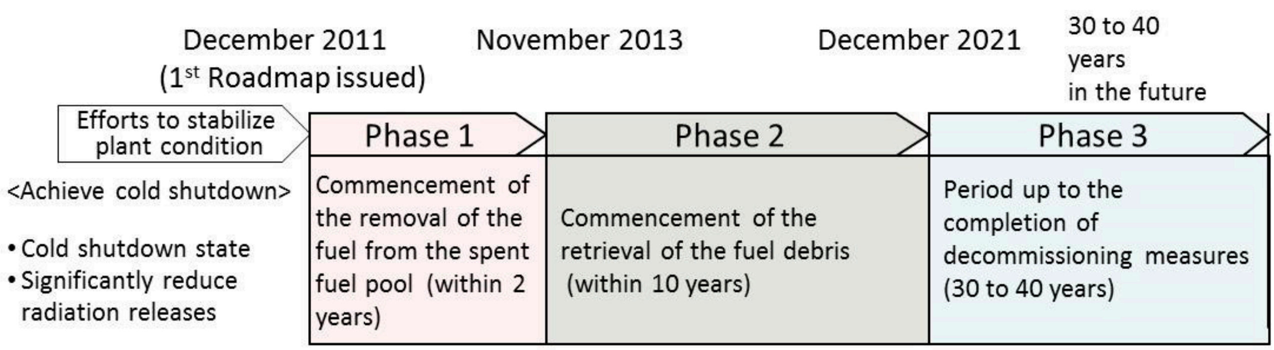


FIG. 3. Mid- and long term Roadmap towards the decommissioning of the Fukushima Daiichi NPP Units 1–4. (Courtesy of the Ministry of Economy, Trade and Industry, Japan.)

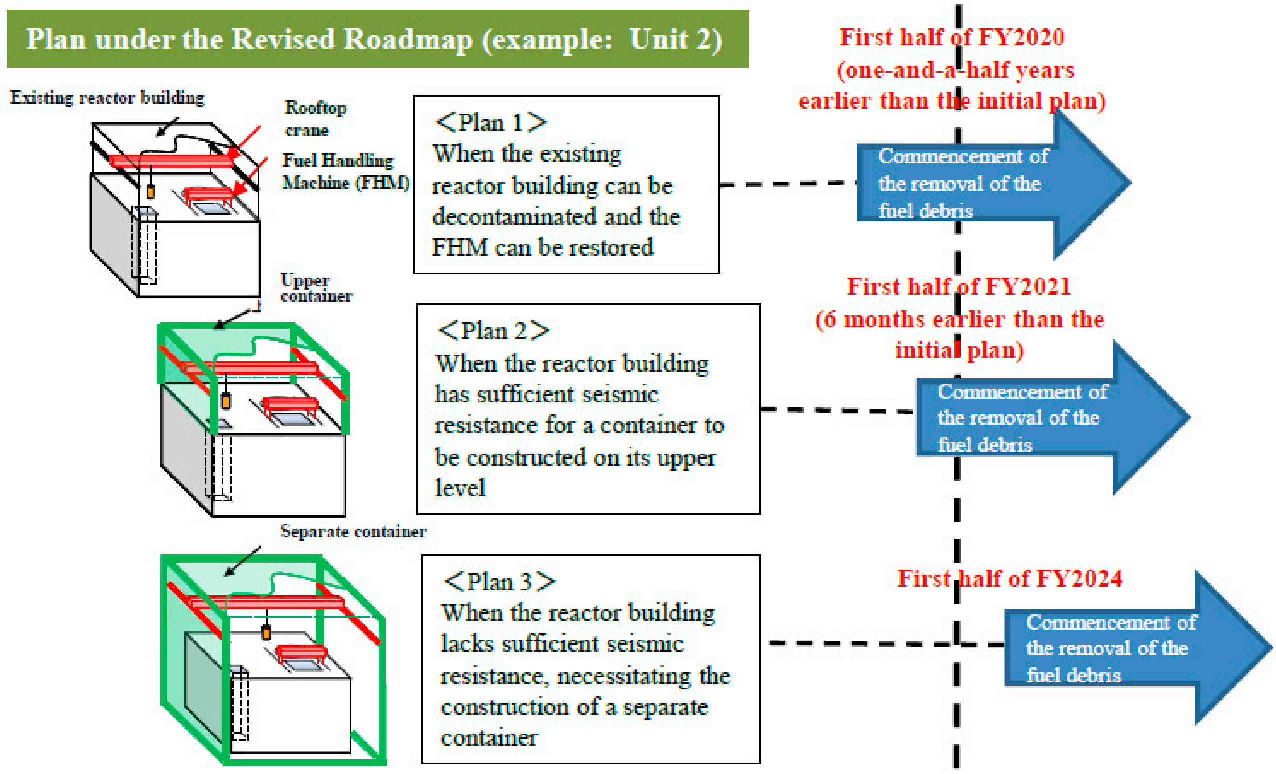


FIG. 4. Optioneering example under the revised Roadmap for Fukushima Daiichi NPP Units 1–4. FY: fiscal year. (Courtesy of the Ministry of Economy, Trade and Industry, Japan.)

3.4. REGULATORY ASPECTS

This section offers a brief overview of specific regulatory aspects to be considered regarding use of technologies, methods and approaches for post-accident decommissioning that significantly vary from those used in the case of decommissioning after a planned shutdown.

Nuclear accidents that happened during earlier stages of nuclear industry development indicated that regulatory approaches in post-accident situations should be based on specific considerations, focused guidance and flexible case-by-case solutions. Past accidents and relevant follow-up activities were difficult also for regulators as there was limited experience with post-accident situations and no, or very limited, specific legislation. Lessons learned and practical examples are now available for regulators in countries with nuclear infrastructure to consider. However, each accident case is unique and needs a specific regulatory approach.

The special status of post-accident unit(s) may be an option to enhance flexibility of regulatory influence and decisions in mid-term and long term perspectives. A variety of technical aspects need to be addressed by the national regulator as well as some non-technical issues that might need to be considered, such as the preservation of information and records for further long term decommissioning processes [30]. In some cases, it may be necessary to introduce a new regulatory framework for the post-accident facility and its implementation should be supported via above standard cooperation and dialogue between regulator and operator.

Some examples of challenges for regulators include specific decommissioning planning for the A1 NPP (Section 3.3.1), licensing of ^{85}Kr release from TMI-2 (Section 5.1.4.1), the need to develop regulations for the use of high integrity containers for TMI-2 radioactive waste disposal [31] (Section 5.7.2), or the decision to build an impermeable ice wall as a part of specific measures to decrease the ingress of underground water to Fukushima Daiichi reactor buildings, discussed in Ref. [32].

A useful practice is the direct exchange of experience among regulators from different countries who are invited to be involved in providing advice related to the regulatory aspects of decommissioning after an accident. An example of this is the Japanese national regulator officially inviting former chairpersons from regulatory authorities in France, the United Kingdom and the United States to support Japan's Nuclear Regulation Authority activities [33].

4. PROJECT PLANNING AND MANAGEMENT

4.1. ORGANIZATIONAL ASPECTS

Decommissioning after a severe accident is likely to require different organization and management than that required following a planned shutdown [7, 34–36]. Aspects such as those listed below will be different and would affect organizational and managerial decisions [36]:

— Technical competences, contractors and specialists;
— Planning;
— Stakeholders;
— Timing;
— Expenses.

The above mentioned aspects are addressed in this section. Planning and managerial aspects are discussed in Section 4.2.1 and cost estimation is discussed in Section 4.2.2.

Following a severe accident, the expertise required will go far beyond that required under routine decommissioning operations. This is owing to, for example, the need to deploy, and be trained in, novel (unproven) technologies. One such technical area could be robotics for use in hazardous environments. Another could be the management of abnormal, unique waste. The pool of contractors and specialists in those technologies is more limited than for routine circumstances [36]. In some cases, there may be very few companies with enough experience/expertise in certain technologies. If so, the identification and selection of these companies may not follow the usual tender and bid process.

In post-accident decommissioning, the planning process will be slower than after a planned shutdown, taking into account the numerous uncertainties to be expected. It is inevitable that any such decommissioning plan will be less detailed and include more contingencies and more flexibility than in more usual circumstances. Deviations from the preliminary plan will be common in the course of the decommissioning work.

In post-accident conditions, both regulator and operator/implementer are challenged and have to maintain certain dynamics due to exceptional conditions. Typically, there may only be general and short term, immediate post-accident regulatory requirements. As abnormal conditions (that may be anticipated or arise unexpectedly) create unusual safety concerns, new regulatory oversight and requirements will be needed. For each such instance, the operator will normally conduct a safety analysis specific to the concern and submit it to the regulator for independent analysis and review. Such reviews may result in additional or modified actions. In general, the regulator will also inspect or conduct surveillance as the operator carries out the required actions.

While participation of a wide range of stakeholders is a desirable component of any decommissioning project, it is likely that post-accident decommissioning will see the active, even anxious involvement of many more stakeholders (see example provided in 5.1.4.1). These may include delegations from segments of the general public, either locals or public opinion groups; the media; international bodies; shareholders; and funding bodies. In turn, the decommissioning organization (and the regulatory body, the government) will have to set-up dedicated centres in charge of communicating information and responding to the concerns of all stakeholders. Whereas, in principle, decommissioning after a planned shutdown is well received by the majority of stakeholders, decommissioning plans and activities after an accident may be met with a challenging attitude by some stakeholders who may feel reluctant to cooperate with those perceived to be responsible for the impacts on their lives [36].

Public distrust after an accident translates to understandable public and political opposition and non-constructive intrusion in recovery work. The overriding lesson here is that people don’t trust (or distrust) a technology, they trust (or distrust) those who implement it. This perception is likely to inject complexities into the decommissioning project before trust can be regained [36].

The presumably long times of a decommissioning project after a severe accident (at A1 NPP, Chernobyl NPP or at Fukushima Daiichi the expected duration of decommissioning spans over several decades) will heavily impact the organization and management of the project. It can be expected that considerations applicable to deferred dismantling will be in order, including staff turnover and regulatory changes. In addition, unlike deferred dismantling of a plant shutdown under planned circumstances, the high radiation and contamination levels expected to remain in a post-accident state will require the long term involvement of the organizations responsible for the plant [36]. This can also be the case for early nuclear facilities (left in quiescent phase), which, although they didn't have an accident, nevertheless have potential significant radiological challenges.

A comprehensive (total) management system should be established to ensure the implementation of long term activities that involve the cooperation of various organizations, bodies and industries. A top level management body needs to be appointed and duties and responsibilities of all involved organizations need to be clarified as they are necessary factors influencing the success of the project. In view of the above considerations, there will be a need for long term capacity building in terms of human and technical resources — see Sections 4.4 and 4.5.

4.2. PROJECT MANAGEMENT

Management of post-accident nuclear facility decommissioning projects is challenging as additional cross-cutting safety, technical and organizational aspects, different from those related to facility shutdown after a normal operation, need to be considered. Support for planning, cost estimation and on-site project management towards the decommissioning end state is related, inter alia, with the development of specific analytic and practical approaches and focused technologies. Useful experience and lessons learned are available from implementation of post-accident decommissioning activities in several countries.

4.2.1. Planning and managerial aspects

The planning and management of decommissioning projects for nuclear facility shutdown after normal operation is usually based on well-known physical and radiological status, detailed project scope, schedule and cost estimation [7]. Planning and project management of a facility after a nuclear accident is much more complicated due to the many cross-cutting safety, technical and organizational issues. Some of the most significant aspects of planning and management for decommissioning after an accident include:

- Uncertain physical conditions that may be in place in hard to access areas of the facility.
- Characterization data and information that is needed for further planning may be difficult to obtain, because the nature of parameters to be measured is different compared with a standard situation.
- Characterization may be done gradually as decommissioning activities are progressing and the results of gradual characterization can influence implementation of further particular decommissioning activities.
- Detailed planning and cost estimating can be restricted by the lack of detailed information.
- Existing organizational and technical infrastructure, standard analytical and practical approaches and available technologies for support of decommissioning planning and project management may not be suitable for post-accident needs. There may be a need to establish a focused organizational unit to manage everyday activities, or an R&D programme may be needed in the long term plan towards final decommissioning.

Table 1 points out some of the differences in standard management functions for post-accident situations.

A comparison of decommissioning planning, managerial and organizational aspects of nuclear units shutdown after an accident and after normal operation can be found in Ref. [35]. The A1 and V1 NPPs at the Jaslovské Bohunice site were compared with regard to the development of decommissioning strategies, preparation and planning for decommissioning, management of decommissioning projects and other specific aspects. The main conclusions can be generalized as follows:

- The duration of the active phase of decommissioning may be shorter than the real operational lifetime of a nuclear facility that is shutdown after a normal operation. This however cannot be expected in the case of post-accident shutdown. A prolongation factor of about 2.5 was identified for A1 NPP decommissioning in comparison with the V1 plant.
- Many more R&D activities and additional safety measures are needed for a post-accident nuclear facility, including the need for the use of unique or adapted procedures, techniques and equipment for decommissioning.
- Decommissioning costs are significantly higher for a post-accident nuclear facility (see details in Section 4.2.2 on cost estimating).

TABLE 1. DIFFERENCE OF REQUIREMENTS FOR A POST-ACCIDENT SCENERIO COMPARED WITH NORMAL PLANT OPERATION

Management function	Difference of post-accident requirement from normal plant operation
Communications and public interface	While this is primarily a corporate responsibility, focused support will be needed to collect and provide information and data to support this function.
Regulatory	Normal rules, regulations and requirements may not apply to the conditions; there might be a need for a certain level of flexibility to deal with consequences of special situations.
Operations for safety and stability	Change from power operations to operations to maintain a safe and stable state. Operations for safety and stability are ensured by the plant SSCs that remain in operation and SSCs are to be specifically introduced to maintain safe and stable conditions and functions.
Operations for cleanup and recovery	A larger scale needed than for normal plant operation, including SSCs added for decontamination, fuel and debris recovery and management, waste management.
Engineering	Wide range of skills necessary to support all functions.
Safety	Trending and analysis of facility conditions, accident analyses for the changed conditions.
Chemistry	Will need to set-up and conduct on-site capability for many analyses not normally needed, both radiological and chemical. In particular, the high radioactivity of many samples will present special handling challenges.
Decontamination	Small and large scale decontamination needs will be ongoing and different amounts and types of secondary waste from post-accident decontamination will be generated.
Tools and mock-ups	Special shops and facilities will be needed to support operations. Many of them will be based on short notice demand, such as fabricating customized end effectors for debris removal or component disassembly.
Radiological controls and worker health and safety	This will be a highly intensive activity to support the large amount of personnel working in radiological and airborne contamination areas.

Planning and management of Chernobyl NPP and Fukushima Daiichi NPP decommissioning are special cases because of the necessity for comprehensive R&D and the critical relationship between managing large amounts of radioactive waste and off-site remediation activities. The establishment and maintenance of specialized, robust technical and organizational infrastructures were necessary to support planning and implementation of on-site decommissioning, and, to some extent, radioactive waste management [37, 38].

The structure of various organizations supporting, implementing and supervising Fukushima Daiichi decommissioning is quite complex. International involvement in establishing an organizational infrastructure, through invitation of external experts as special advisors or hosting of peer reviews to evaluate decommissioning plans, may be considered as a useful practice [39–41].

4.2.2. Cost estimating

The costs of decommissioning a reactor after a severe accident are difficult to estimate as they are subject to a number of uncertainties, such as the lack of specific information needed for decommissioning planning, changes of initial conditions and limited cost-based precedents for many of the unique activities. Decommissioning costs will be ultimately much higher than after a planned shutdown (according to some estimates, even an order of magnitude higher).

To ensure a regular cash flow, it will be crucial for funding bodies to guarantee the availability of funds throughout the entire decommissioning project. In the absence of such assurance, it is likely that post-accident decommissioning will last even longer, ultimately resulting in higher costs if spread over a longer time. The inevitability of uncertainties during decommissioning is likely to add unforeseen expenses (e.g. the discovery of hidden or underground contamination) and cause further delays.

ISDC [42] is a cost structure model that was devised by the OECD/NEA, IAEA and the European Commission to help with the production of cost estimates associated with decommissioning activities. ISDC has been adopted by several countries to manage cost estimates or to map national estimates for purposes of comparison.

The hierarchical structure of ISDC is illustrated in Fig. 5. Activities at Level 2 represent a subdivision of the Level 1 activities. Activities at Level 3 provide a further subdivision of activities to the level of typical decommissioning activities, which may be identified in any decommissioning project. Level 3 is the reference level with links to the cost items identified in cost structures other than ISDC. Level 3 activities represent the basic building blocks for developing the overall cost estimate.

The ISDC can also be used to estimate decommissioning costs after an accident, using the principal activities shown in Fig. 5. Additional ISDC hierarchical levels will be required to account for increased non-standard decommissioning activities that may be needed after an accident. Examples of additional cost activities for decommissioning after an accident include:

— Damaged fuel and fuel debris management;
— Extensive use of radiological and environmental management;
— Management of non-standard types and quantities of radioactive waste;
— Safety system management;
— Construction of emergency temporary infrastructure, including new shielding and facilities.

Further considerations related to the post-accident use of the ISDC are summarized in Annex IV.

Because of uncertain, unique and changing conditions, much of the cost estimating for cleanup and recovery will be of a rough order of magnitude. Estimates will constantly be adjusted based on actual costs.

A standardized unit of cost approach will not be applicable for many activities for the reasons stated above. In these instances, labour cost estimating will be modified to judge the number of workers and other personnel on a task by task basis using an average burdened unit cost labour rate for the skills needed. In addition, the crew composition and best estimate of each task's duration will be needed. Where there is significant uncertainty, an estimate of the level of effort over an extended time can be used in lieu of the duration of tasks.

4.3. WORKER HEALTH AND SAFETY

The safety of workers involved in decommissioning activities is one of the highest priorities. Providing workers with a better work environment is essential for the safety and success of the decommissioning project. This will lead to the acceleration of the site work and an increase in reliability.

There are a variety of ways to improve the working environment, such as appropriate shielding, the use of robotic devices, air-conditioning, the establishment of rest space and simplifying working equipment. The following are examples of the on-site work environment measures used by TEPCO [43]:

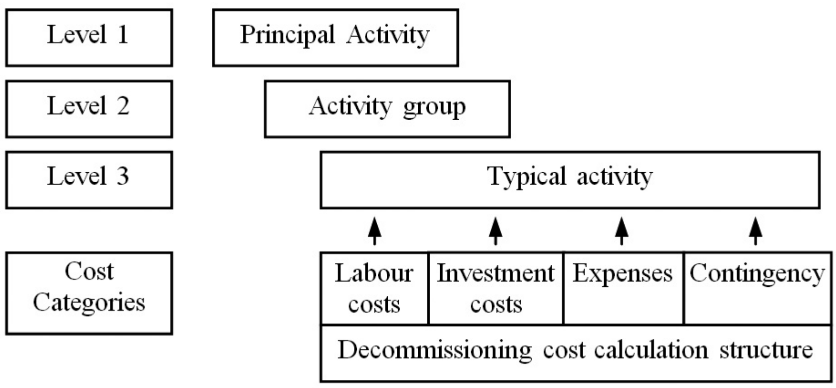


FIG. 5. Hierarchical structure of the international structure for decommissioning costing.

— Measures to reduce exposure dose (see below) and to improve the work environment (e.g. proper classification of working areas or expansion of areas without full face-covered mask restrictions to speed up decontamination work).
— Improvement of worker facilities (e.g. creation of Fukushima Daiichi new office building, large refreshment room and lunch cooking centre).
— Countermeasures against heat stress[2] (drinking water and wet paper towels supplied nearby, see Fig. 6). Examples of portable cooling vests are discussed in Refs [44–46].
— Measures to protect against the combination of radiation protection and heat stress [47].
— Special personal protective equipment (see Fig. 7) and personnel monitoring dosimetry.
— Long term contracts for workers.

Exposure dose reduction examples are discussed below and include various shielding measures introduced at the Fukushima Daiichi Unit 4 refuelling floor before the removal of stored fuel, and the removal of rubble from Unit 3. The first example is the installation of steel and lead panels and protective screens that contributed to about 70% reduction of exposure dose. The second example illustrates remote deployment of standard mechanisms (e.g. crawler cranes) that were used for removal from small pieces of rubble to a 20 tonne fuel handling machine [48].

Examples of TEPCO's management to increase safety and assure quality are as follows [43]:

— Creation of operational procedures in line with the actual work on the site, thorough participation in hazard prediction activities, and enhancement of communication with cooperative companies.
— Clarification of where the responsibility lies in the chain of command on the site, including the relationship with subcontractors.

FIG. 6. Countermeasures against heat stress: A car fitted with an air-conditioner; inside the car is a water cooler and rest area. (Courtesy of TEPCO, Japan.)

[2] Heat stress is when the body is unable to cool itself by sweating. Several heat induced illnesses such as heat cramps, heat exhaustion and the more severe heat stroke can occur during heat stress.

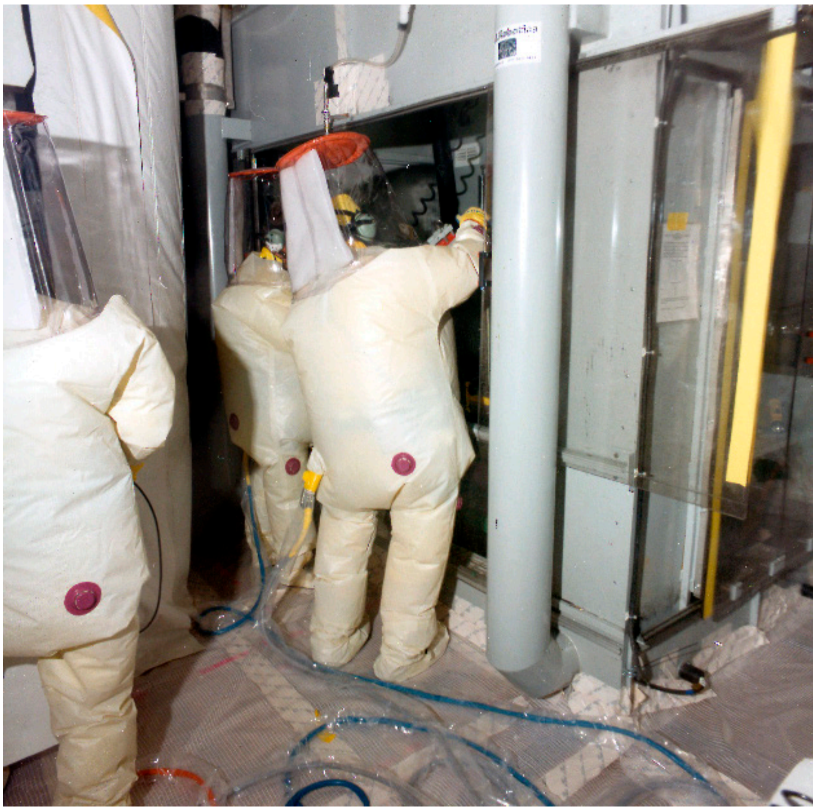


FIG. 7. Example of special personal protective equipment, Rocky Flats. (Courtesy of USDOE, USA.)

— Reinforcement of the structure and personnel of the safety and quality control departments through:
  - Creation of a function in charge to supervise the safety and quality control departments of the head office and the nuclear power stations;
  - Increase in the number of staff members for the safety and quality control departments in the nuclear power stations.

4.3.1. Examples for working environment improvement at TMI-2

At TMI-2, personnel entry into the containment building was necessary over several years to load the damaged fuel and debris into canisters. This activity was performed using long handled tools from above the surface of the water within the reactor building. There were several methods employed to minimize the exposure and improve the comfort of workers, including the following:

— Decontamination methods, such as floor scabbling leading to the reactor building entrance.
— Water washing of the building walls from the top to below the working level.
— Shielding of stairways, high radiation areas, air coolers, floor drains, floor hatch openings and components. Lead blankets were commonly used (Fig. 8). In one case, cylindrical polymer water columns were used; however, because of leakage, the water was replaced by sand.
— Operations were performed from within the refuelling canal above the reactor. The canal was left empty to shield the operators from reflected radiation emitted from the basement.

One approach to worker comfort provides a useful lesson. Ice vests were initially tried to reduce heat stress; however, they proved to be cumbersome for workers. To resolve the heat issue, the containment building was air conditioned. This had the negative impact of creating additional contaminated water as a result of condensation within the air coolers, although processing of this additional waste was a small price to pay when compared with the need to establish acceptable working conditions.

FIG. 8. Specialized radiation shielding including lead blankets. (Courtesy of Lancs Industries, USA.)

4.4. HUMAN RESOURCE NEEDS

As described in previous sections, post-accident decommissioning needs special organizational, managerial and technical approaches, which are different from normal decommissioning approaches. Human resources need to be well prepared and trained to implement those special approaches appropriately.

Any post-accident scenario is unpredictable and will require a programme that includes continual investigation and development. Conventional operating plant and utility management and architect–engineers, turnkey suppliers and responsible constructors need to be prepared to work in the unconventional manner required by projects that result in an R&D type environment.

The functional needs of an organization in a less severe case of fuel damage may be handled mostly by plant staff. In the most severe cases, this will require a large additional organization. Almost any incidence of fuel damage will require outside expertise and additional personnel. The organization should be flexible and respond quickly to the need for change when the actual requirements and priorities of the project need to be modified [1].

A special issue is the establishment of a training programme for many workers of all skills who do not have nuclear decommissioning experience. Many workers may need to be recruited from non-nuclear fields in the case of post-accident decommissioning. The conditions following a nuclear accident will require special training for the more severe conditions. It is important to include not only the technical skills, but also human aspects affecting worker attitudes and behaviours towards safety integration, in the post-accident training programme.

Another aspect that is very much related to organizational and human resources issues of post-accident decommissioning is the right motivation of staff. Motivational factors directly affect the safe and effective decommissioning of nuclear facilities, this is especially important in post-accident cases. With motivated staff, it is not only the level of output that improves, but higher standards of quality can be achieved by both on-site staff and subcontractors. Leadership of and initiatives by managing staff to encourage worker motivation are also important.

4.4.1. Management functions

Specialized, focused organization, including executive management, will be needed for many tasks to establish and maintain safe conditions, and to handle accident conditions for the duration. At some point, in parallel, this organization will need to manage, plan and execute the cleanup and recovery of damaged fuel, fuel debris and related radioactive waste. In addition, there are a few examples of the specialist management support functions that will need to be modified for the post-accident situation, such as:

- A public communications interface to work with the many outside influences. An effective communications group is needed to translate complex technical language into explanations suitable for less technical or less involved individuals.
- Regulatory interface to address the many unprecedented operations for which safety analyses followed by the regulators' review will be conducted and approval obtained.
- Work with off-site research and development organizations dealing with challenges unique to the post-accident.
- Input from advisory groups consisting of highly experienced individuals to conduct independent reviews of plans and activities followed by appropriate recommendations based on international standards and experience from other post-accident decommissioning.
- Management of records and documents (general knowledge and information management) is important for both future decommissioning planning and information sharing with stakeholders.

4.4.2. Operations

There are many facets of operations such as those provided in the following non-exclusive list:

- Maintaining the existing plant operational organization and infrastructure as there will be many operational needs for systems and components not directly affected by the accident. Filtered ventilation system throughout the facility is one example [1].
- Operations similar to normal, but with a need for expanded skills because of the different characteristics of materials and systems. Examples include, but are not limited to, physical and radiological characterization, water processing, waste management, radiation protection and source management.
- Commissioning and operation of special construction, facilities and equipment to support implementation of the post-accident decommissioning activities (see examples in Section 5).
- Robotic and other remote technology operations.

4.4.3. Engineering and special technical skills

The following non-exclusive points may be considered as required engineering and special technical skills:

- Development, design and specification of case-by-case equipment and components on an as needed basis. Examples include the construction of a new safe confinement (NSC) covering Unit 4 of Chernobyl NPP, an underground frozen wall surrounding Fukushima Daiichi Units 1–4, and retrieval of damaged fuel and fuel debris from Fukushima Daiichi Units 1–3.
- Interface with remote technology suppliers.
- Shielding design and implementation by procurement or on-site fabrication.
- On-site machine shop needed for fast turnaround of tools and end effectors that require custom configurations.

Another specific enhanced need may be in an analytical service provided through laboratories and facilities for radiochemistry and other methods of sample characterization as well as focused analysis of visual characterization created by methods such as video, laser imaging, sonar and others.

4.5. MANAGING TECHNOLOGY NEEDS

The needs for sophisticated technological application and development in post-accident situations are driven primarily by the following two overriding factors:

- Exposure: High dose rates, airborne radioactivity and other noxious atmospheres, high temperatures, and/or other environmental conditions that prevent direct human access;

— Feasibility and accessibility: Too narrow spaces, underwater operations, lack of visibility due to space configurations, heavy loads that are beyond human capabilities.

Other incentives for the use of technology include enhanced project productivity and efficiency. Examples are devices that gather in situ characterization data and speed up its processing, which can greatly enhance the overall schedule.

In particular, development and deployment of remotely operated and robotic technologies from other industries, in addition to those from nuclear operations and decommissioning, have brought a tremendous amount of relevant experience. Nevertheless, compared with most non-accident situations where direct hands-on equipment operations by workers can be prevalent, severe post-accident conditions need substantially more technological solutions. In deciding upon specific technology needs, remotely operated or otherwise, project managers deal with the complexity of each unique application. Valuable inputs can be given by operators, engineers and technicians who will be responsible for deployment of technologies. Further, on-site or near site organizations are an essential part of the project team that may include responsible vendors and chosen suppliers. Organizations should be staffed with those who are responsible for the ultimate application of the technologies and who are able to provide the site interface for designers and manufacturers located elsewhere.

Choosing to go down a path of technology that requires significant development will depend on the severity of the conditions. In situations such as those that initially existed at TMI-2 and still exist at Fukushima Daiichi, remote devices are essential. In less severe situations, there are trade-offs for the degree of sophistication needed. For situations similar to those at Paks and A1 NPPs, initial considerations without a substantial amount of evaluation can include, but are not limited to, the following factors:

— Choosing among existing tools and devices that can be applied with reasonable adaptation.
— Schedule optimization; in particular, the more urgent treatment requirement, the less incentive there is to use a specifically developed device.
— Large quantities of required consumables (e.g. abrasives, filters, ion exchangers) or waste generation will favour the use of already existing tools, for example with commonly available spare parts, over more sophisticated devices.
— The profile/skills of available operators versus the need for specialists or extensive training.

Where conditions are severe and it is clear that sophisticated technologies are needed, a host of other factors come into play. In selecting remote technologies, some of the important considerations and evaluation factors include the following:

— A phased feasibility study and conceptual design approach may be needed because every situation is different. In particular, final design may have to wait for data that characterizes the actual physical conditions. In fact, specific remote technology development may be needed to first obtain such data.
— In determining the feasibility, available remotely operated equipment and tools should be investigated before the design phase; either for direct use or with some adaptation to the conditions at hand.
— Opportunities to use innovative, and possibly unproven, technologies are to be considered with caution and with the understanding that a significant amount of proof testing will be needed.
— The realization that some degree of complexity is inherent in order to provide the performance for all functions; however, the design of equipment doesn't need to be overly complex in the range of tasks a single machine can conduct. For example, the same device to cut underwater cannot be expected to perform the same function in air. Put another way, designing for both environments may lead to complexities that result in unreliability and increased costs.
— A user friendly man–machine interface; here is where the project team approach can be very useful.
— Technical aspects that need improvement are to be recognized; for example, less vibration during cutting and cable (tether) management.
— The ability to maintain and decontaminate equipment needs to be integral to the design.
— When using commercially available equipment, procuring spare parts for the most important modules (e.g. cutting tools or robotics arm) will benefit the schedule with regard to needs for decontamination or repair of damaged module(s) as they arise.

— Use of mock-ups and software simulation for operator training is of great benefit for testing various operational modes and rehearsing specific actions.
— For manually operated systems, limiting human presence means shorter operating time and thus productivity. Operation from outside the hazardous zone may be necessary. Remote manual operation can also improve the safety for humans.

5. POST-ACCIDENT KEY OPERATIONAL FUNCTIONS

5.1. MAINTAINING SAFE CONDITIONS

Preparation towards decommissioning of an accident damaged facility first involves establishing stable conditions to prepare for subsequent cleanup and decommissioning activities in the case of a severe accident, or return to service in cases where that is feasible [7]. The general term 'stabilization' is associated with this post-emergency phase. Stabilization involves ensuring there are SSCs in place to reliably maintain a stable condition for the long term or until they are no longer needed towards the major objective of removing nuclear fuel and the nuclear fuel debris in the case of significant reactor core damage. Stabilization also serves to reduce radiation levels within the work areas and contributes to lower off-site doses.

There has been significant variability in the conditions of past nuclear accidents. As a result, it is not useful to define formally the stabilization phase such that it would be applicable to all post-accident conditions. However, many of the essential functions needed to carry out stabilization are similar among events and are described here. First described are infrastructure systems and facilities (Section 5.1.1). These are differentiated from those needed directly to support stabilization of the nuclear plant.

Stabilization objectives for the nuclear plant itself include, but are not limited to, the following functional requirements:

— Infrastructure systems and facilities;
— Monitoring facility conditions;
— Decay heat removal;
— Ventilation and gas control;
— Controlling radiation and spread of contamination to off-site;
— Ensuring essential facility structural stability.

Each of these is described separately in the following subsections. Maintaining and monitoring nuclear subcriticality is also an important stabilization function that is discussed in Section 6.4.

5.1.1. Infrastructure systems and facilities

Infrastructure includes functions typical of any industrial facility, such as non-essential water supply, electrical power, communications, administrative buildings, roads, worker facilities and others. An important first step following an accident is to make an assessment of the existing site infrastructure to ascertain the status. It will be necessary to identify the following:

— Systems that are still functional and can be used to support stabilization, cleanup and subsequent decommissioning operations;
— Systems that have been damaged, but are still required and need to be modified or replaced to support the programme;
— Systems that are damaged and are no longer required and can be removed in early operations;
— New systems that will be needed to support stabilization, cleanup and subsequent decommissioning operations.

This last point illustrates that existing infrastructure requirements to support the various phases of stabilization, cleanup and the subsequent selection of a decommissioning option may be insufficient to deal with the problems of an accident damaged reactor. The facilities for the storage of waste are unlikely to be sufficient to deal with the quantities involved during cleanup and decommissioning; as an example, facilities for the storage and pre-processing of reactor operational wastes will normally be present but their capacities may be inadequate to deal with an accident damaged situation. The requirement for additional facilities will be particularly evident if an immediate dismantling scenario is selected requiring a need for increased waste handling capacity early in the project. On-site waste storage should be considered to reduce transport requirements. This is discussed further in Section 5.7.

Other aspects of infrastructure will need to be developed. There may be a need to construct additional access routes via demolition of existing structures or the construction of new roadways and railheads. Utility services may need to be replaced, extended and/or re-routed. A few examples include the following:

— At Chernobyl NPP, additional electrical power was required to maintain services following the shutdown of Units 1–3. At Fukushima Daiichi, new electrical substations were built because of the damage caused by the tsunami.
— Essential communications such as phones, internet/Wi-Fi access, and radio communications with worker crews. Constant communication with workers within radiological or confined areas is essential.
— Worker facilities for change out, dosimetry and monitoring. At Fukushima Daiichi, an existing training facility, approximately 12 km from the site, serves as a marshalling station for workers and visitors on their way to the site. In addition, a facility dedicated to simultaneous whole body counting was established.
— Offices for management, planning, engineering and administration. At TMI-2, an office building was constructed for the management and support staff.
— Material storage is a logistical need for many events, either with existing or added warehouse space.

A prime example is the construction of an incinerator at the Fukushima Daiichi plant to burn the low level waste (LLW) generated from the cleanup and decommissioning of the site. The 3170 m^2 facility is to be used to reduce the volume of LLW, including such things as clothing, gloves and building materials (see Fig. 9).

Additional infrastructure requirements are also likely to be needed to deal with accumulations of both undamaged and damaged fuel. Additional wet or dry fuel storage and temporary storage/laydown areas for plant equipment and materials are discussed further in Section 6.

5.1.2. Monitoring facility conditions

A severe accident causes the destruction of many of the plant's normal monitoring and measuring instruments. Both during the emergency phase of accident response and for subsequent recovery activities, many of the functions for which these instruments were designed must nevertheless continue to be monitored. It is essential to decide on important parameters to be monitored (such as temperature, neutron multiplication, water levels).

Methods are to be established to measure those parameters, either with installed instruments for direct detection, or by indirect methods (such as monitoring of short lived noble gas release to detect possible nuclear criticality, as described in Section 6.4). With time, monitoring in some areas may no longer be needed, and in other areas it may need to be improved. For example, as access is gained to fuel debris, it may become possible to directly monitor neutron multiplication with instruments designed for that purpose. Some parameters will be more difficult to measure than others; for example, establishing automatic measurement of water level can be a challenge when the sensors are difficult to reach because of radiological conditions at their locations.

Reliability of monitoring is important. Examples of means to accomplish this reliability include redundant systems and components, diversity of detection methods, battery backed power supplies and duplication of monitoring centres at remote locations. Human observation and remote cameras (with or without interpretive software) can also serve as diverse means of detection.

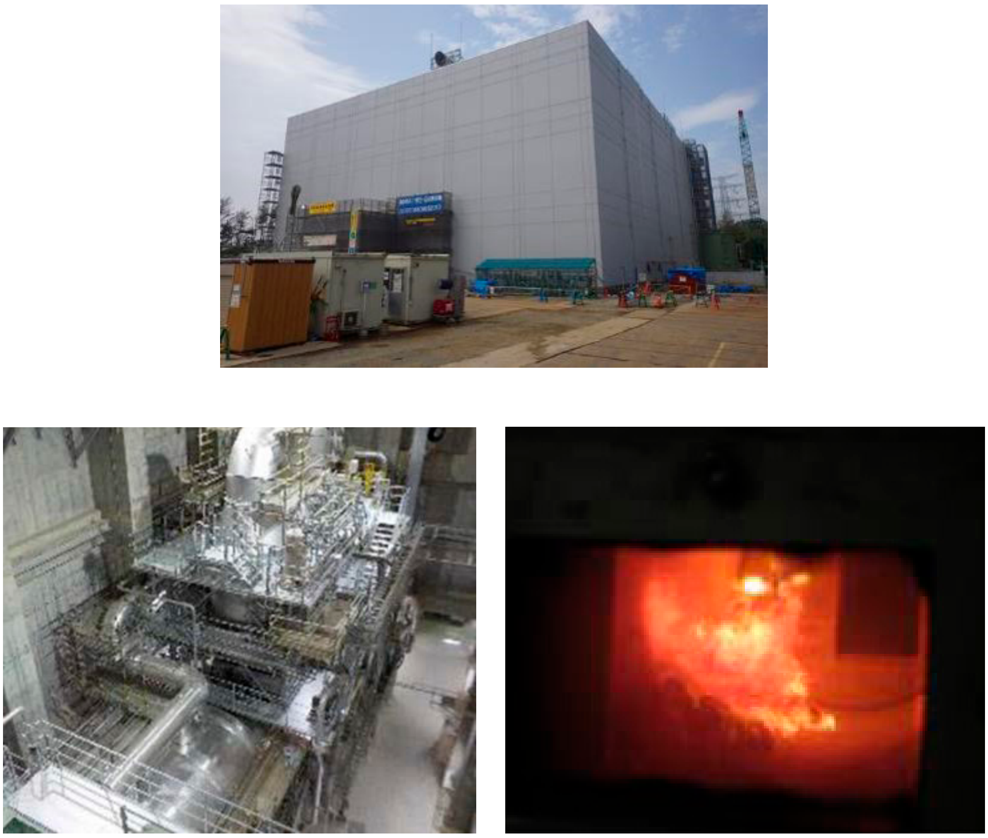


FIG. 9. Fukushima Daiichi incineration facility. (Courtesy of TEPCO, Japan.)

5.1.3. Decay heat removal

After shutdown of a nuclear reactor, either following normal operation or after an accident, heat continues to be generated as a result of radioactive decay of fission products within the nuclear fuel [7]. The overall rate of heat generation depends on the operational history prior to shutdown; that is, the concentration of fission products within the fuel. Removal of decay heat is needed to maintain a cold shutdown condition. Nuclear reactor plants have systems specifically for this function.

Regardless of the configuration of the nuclear fuel following an accident, decay heat continues to be generated. Its removal is especially important so as to prevent further disruption or damage resulting from increases in temperature and/or pressure. Over time, the rate of decay heat generation will become low enough that passive removal via pathways to the environment will be sufficient and active systems will no longer be required. For example, at TMI-2, which had operated for a few months prior to the accident, passive decay heat removal was possible in 16 months. In other cases where the reactors operated for longer periods such that fission product concentrations are higher may require several years to reach passive removal.

When normal cooling systems can be maintained in operation during the post-accident phase, they may be used for removal of the decay heat. In some cases, additional or substitute measures may be needed. If the original function is completely lost, installation of new cooling systems will be needed. The following are three examples that illustrate the variety of methods that have been used.

At Windscale Pile 1, normal air cooling or carbon dioxide injection could not extinguish the fire; therefore, water was injected. This also served to remove some of the decay heat.

At TMI-2, both primary and secondary systems were functional after the accident. However, use of the normal decay heat removal system would have circulated highly radioactive water within the auxiliary building, thereby creating high radiation areas for which human access was needed. Therefore, an alternate method for maintaining cold shutdown was established using natural circulation in the primary system to reject heat via the steam generator tubes, and circulating water through the secondary side of the steam generators for heat removal. The steam generators, which are located within the containment, performed as a radiation barrier (see Fig. 10).

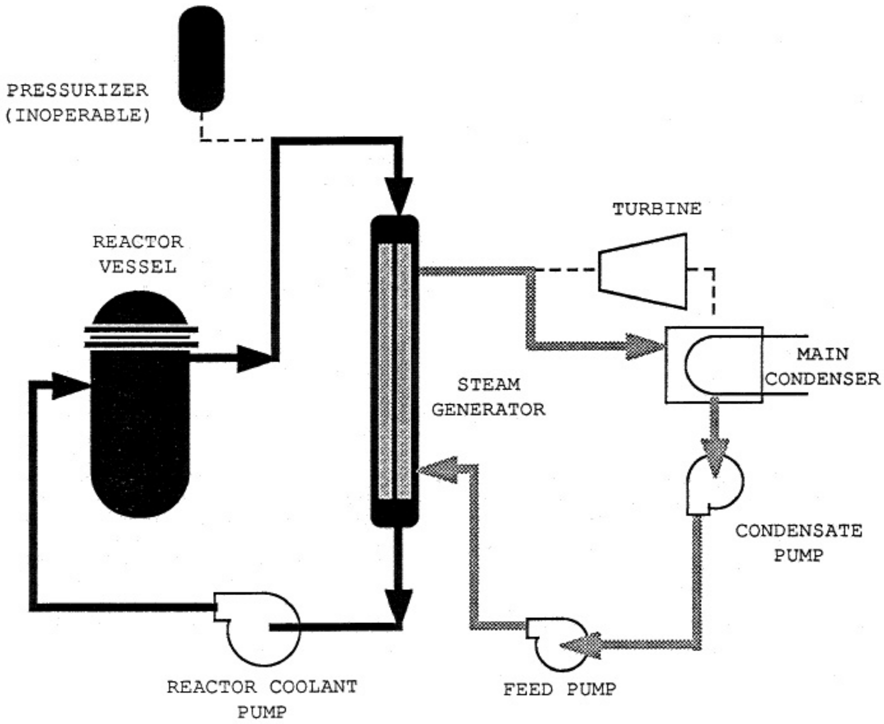


FIG. 10. TMI-2 decay heat removal pathway. (Courtesy of EPRI, USA.)

At Fukushima Daiichi, decay heat removal was required in two areas within the facilities. Undamaged spent fuel was stored in the Unit 1–4 spent fuel pools located within the damaged reactor buildings. Although their heat generation was much less than those within the reactor, it was necessary to keep the pool water well below boiling temperature. Temporary cooling and desalination systems for spent fuel pools were installed and placed in operation. The spent fuel in these pools is being moved to a common storage pool away from the reactors, which is separately cooled as may be needed [41]. The destroyed fuel and fuel debris in the Unit 1–3 reactor vessels and containment buildings is cooled by water circulating using an interim circulating injection cooling system loop about 4 km in length that was constructed during the stabilization phase. With this system, the contaminated water is pumped from the turbine building and returned to the reactor for injection cooling after decontamination and desalination. For reliability over the duration for which it will be needed, this system is being shortened and improved [7].

Desalination is related to the sea water that was initially used for the cooling of fuel. It may impact materials of SSCs and the fuel itself, for example through increased potential for their corrosion. An important lesson to be highlighted in this regard is the need for alternative and reliable cooling options to be installed that are capable of operating over prolonged periods of time [49].

5.1.4. Ventilation and gas control

During a reactor accident with fuel failure or core disruption, fission product gases (noble gases such as krypton and xenon and volatile fission products such as caesium iodine) are released from the reactor core. These gases may need to be removed to reduce the airborne radiation hazard. In addition, provisions are to be made to capture iodine after the accident to protect workers and the public.

Hydrogen is generated as a result of the radiolysis reaction of steam with zirconium alloy. Trapped hydrogen and hydrogen that continues to be generated needs to be removed to prevent explosions [11].

When the original containment system is intact, controlled release of the gases to the atmosphere can be conducted in compliance with regulations. At National Research Experimental, a research reactor in Chalk River [50], Windscale Pile 1 [51] and A1 NPP [52] the original containment functions were not lost. The radioactivity released to the environment was mainly in the form of gases from the discharge stacks during or following the accident.

5.1.4.1. TMI-2 gas control

Although the TMI-2 containment was not breached, the hydrogen that accumulated detonated and caused a brief pressure spike. Some damage was observed later, but it was not serious enough to impede operations. Access to the containment at TMI-2 was limited by an inside atmosphere with levels of ^{85}Kr that were unacceptably high for any sustained occupation (37 000 Bq/cc measured shortly before venting). To proceed with the cleanup safely and quickly and also to reduce risk of the unpredictable and uncontrollable leaks to the environment, the gas had to be removed. After a long process of preparation and review, the containment was vented of approximately 1700 TBq of ^{85}Kr and the first entry was made approximately 16 months after the accident.

Seven months of intensive licensing and legal effort was required to obtain the regulatory approval to vent. During this period, an environmental assessment was prepared, a citizens' monitoring programme was established that recruited area residents for monitoring activities during the purge, and the public was involved in a number of other ways.

The public process resulted in at least three schemes as alternatives to venting. One was selective absorption with charcoal, another was a scheme for a balloon-supported sleeve, and a third was for jet assisted boosting of the vent effluent to higher elevations. None of these were seriously considered by the regulatory agency. A fourth option, called the selective absorption process, was conceptualized at a USDOE laboratory. An independent technical evaluation of the process was conducted. It was concluded that purging is preferred in all respects, including feasibility, effectiveness, practicality, health and safety, psychological stress on the nearby population, schedule and cost.

Stress and other psychological impacts led to a consideration by the regulatory agency to allow faster venting than was originally proposed. The venting operation technical specifications allowed the use of real time meteorological data to compute off-site doses. This permitted the project team to take advantage of optimum dispersal conditions by increasing the release rate when meteorological conditions allowed, and thus complete the venting more rapidly while still meeting the requirements for release limitations.

Two existing systems were used for the purge. The hydrogen control system (modified with a higher capacity fan, new controls and interlocks) was used to vent at a rate up to 0.28 m^3 per second while the containment atmosphere was rich in ^{85}Kr. The containment air purge and purification system was used for rates up to 8.7 m^3 per second during later stages when the concentrations were lower. The flow rate was controlled based on the off-site integrated dose criteria. All releases were through the station vent, which contained monitoring instrumentation. An extensive off-site network of monitors and samplers was established for the purge.

Slow rate purges were conducted over an 11 day period followed by 4 days of fast purging. The operation was accomplished without incident. During this period, the ^{85}Kr concentration within containment dropped to approximately 2.2 Bq/cc. There were also a number of smaller purges later to vent the ^{85}Kr subsequently released from the water in the containment basement.

Following the ^{85}Kr venting, two technicians, heavily laden with protective gear and instruments, made the first post-accident entry into the dark and dripping wet containment building. With access, the project team was finally able to evaluate fully the damage to the plant and to work directly on the systems and equipment that had been most affected. More than 2000 entry days were to follow.

5.1.4.2. Chernobyl NPP gas control

At the Chernobyl NPP, the accidental explosion substantially destroyed the reactor building. The sarcophagus of Chernobyl NPP Unit 4 (called the shelter) was equipped with ventilation shafts for convection and ventilation systems.

A stationary dust suppression installation (SPP) was commissioned at the end of 1989 for the purpose of limiting the spread of radioactivity from the shelter to the outside. The SPP was made up of one distribution pipe header with 14 nozzles, covering the central part of the under roof space. The installation was designed to reduce the concentration of radioactive aerosols within the shelter premises and prevent their spread into the environment. During the first years after its commissioning, the average activity of aerosols inside the shelter premises was reduced by a factor of ten. While in use (from the middle of 1990s), film forming compounds (fixatives) were applied to seal in place small quantities of dry residue.

Based on the results of SPP operation, it was found that the system was not operating efficiently enough because dust suppression was effective only within a limited area of the central hall (approximately one third of the total area). In 2003, the SPP was upgraded by extending the system coverage area to the whole under roof space of the shelter, and by optimizing the applied fixatives and modes of their application. The effectiveness of SPP operations on the radiation situation eventually decreased. However, its shutdown in the long term could result in increases of release and deteriorating radiation conditions both inside and outside the shelter.

Additional upgrades to the dust suppression system were done in 2004. The improvement resulted in the enlargement of the spraying area from 1500 m^2 to 5200 m^2. The number of spray nozzles was increased from 14 to 49 and; the network of collectors and pipelines was enhanced.

A full check of the upgraded SPP pilot industrial operation was conducted during 2004–2005, including the operation of nozzles in the shelter under roof space. Optimizing modernized SPP (MSPP) operation modes drastically reduced the number of leakages of fixatives and solutions into the bottom premises of the shelter and helped to decrease the consumption of dust suppression composition.

The MSPP was put into routine operation in 2006 and its additional safety function (protection function) is the reduction of the effective neutron multiplication factor (K_{eff}) in fuel containing material (FCM) accumulations located in the reactor hall. The reduction of K_{eff} is achieved by spraying a neutron absorbing solution of gadolinium nitrate onto the surfaces of those accumulations. The solution is sprayed when the safe operation limits are exceeded over the parameters measured by the systems controlling the condition of FCM accumulations. It was further proposed to establish an additional operation mode of MSPP that would allow for radioactive aerosol deposition from the air during emergency situations, including in negative temperatures.

In 2006–2008, the MSPP was used for the preparation of work areas in the shelter under roof space during shelter stabilization activities. Attention was given to the application of a protective polymeric coating on the localized dust forming surfaces to ensure reliable immobilization of dust forming substrates (sand, construction chippings). High immobilization effect was confirmed especially for alpha and beta contaminants. Various dust suppression compositions were tested and the optimal one, marked AK-510 based on siloxane acrylate, was selected before MSPP commissioning. This composition ensured all necessary characteristics, including the following:

- Required thickness of surface up to 200 mm;
- Drying time up to grade 3;
- Necessary adhesion and uniformity of thickness distribution;
- Chemical inertness;
- Water resistance;
- Flame spread rate and smoke developed index;
- Time of protective effect, radiation resistance.

The MSPP can be used for adding neutron absorbing materials (0.1% of gadolinium solution) in accumulations of FCMs, both in addition to and independently from the standard system of adding the gadolinium solution to the MSPP upgrade. The addition of neutron absorbing materials resulted in the following:

- Improved shelter radiation and nuclear safety during the current operation and emergency situations;
- Improved working conditions of personnel;
- A protective polymeric coating (an essential preventive measure of safety in case of an accidental collapse of shelter structures).

The MSPP operations succeeded in reducing the shelter environmental impact by decreasing (more than twice) the release of radioactive aerosols from the shelter and by decreasing (more than four times) the loose surface contamination in the under roof space.

5.1.4.3. Fukushima Daiichi NPP radioactive gas control

At Fukushima Daiichi, for each of Units 1, 2 and 3, which still have fuel materials in the core, a system was installed to control the gas discharge from the containment vessel by keeping the gas at a negative pressure with filtered circulation. It also has a defence in depth function of nitrogen injection to prevent the possibility of explosion from hydrogen that continues to be released, although in much smaller quantities compared with those during the accident.

Building covers were installed on Units 1, 3 and 4, where the upper parts of the reactor building were destroyed during the accident [9]. The covers consist of frames and panels. The cover for Unit 4, where all the fuel was within the spent fuel pool, used the remaining reactor building as the support. It was completed in 2013; fuel removal started in November of that year, and was completed in December 2014. The covers for Units 1 and 3 are self-supporting. Construction of more reliable containment that enables good working conditions for full scale cleanup activities, such as fuel debris retrieval, will ultimately occur [7].

5.1.5. Controlling radiation and off-site spread of contamination

In the case of an INES Level 6–7 reactor accident, the containment function can be severely breached due to high pressure or explosions and unrestricted release of radioactive gases might occur if supplemental measures against severe accidents, such as emergency venting, are not available or fail to work properly. To limit further releases in the period immediately after the accident, short term closure measures can be anticipated. Some examples follow.

At Windscale Pile 1, large area filters were placed atop the discharge stacks to minimize the amount of significant quantities of radioactive particles released to the environment during an accident or from fuel failures during operations. During the 1957 accident, this did not inhibit the release of fission gases (mainly ^{131}I). Following the accident, the discharge stacks were capped to restrict the release of particles from burnt fuel residues.

At the Chernobyl NPP, an enclosure (called the sarcophagus or shelter) was constructed over a period of six months following the accident. The shelter was equipped with electricity, ventilation, fire extinguishing, monitoring and other features. It has been working as an effective containment since its construction. However, because a limited time was allowed for design and construction, problems have arisen, such as inaccessibility to many important structures. The lifetime of the shelter is estimated to be about 30 years and therefore a project to construct the NSC was initiated in 2007. The primary function of the NSC is to confine radionuclides for a minimum of 100 years. The NSC will also enable good working conditions for full scale decommissioning activities, such as fuel debris retrieval and waste treatment, when the decision is taken eventually to proceed with these activities [7, 53].

At Fukushima Daiichi, an early objective following the accident was the reduction of radiation levels to limit the dose to the public in the area immediately surrounding the plant. The dose reduction plan included control and management of gaseous and liquid releases combined with measures to reduce radiation from on-site waste and operating facilities. In addition to the radioactive gas control previously discussed, measures included:

— Limiting the liquid radioactive release to the environment with a focus on discharges of contaminated water to the ocean.
— Decontamination outside of the buildings.
— Limiting radiation from radioactive materials generated during recovery from the accident.
— Establishing a covered area and constructing a storage facility for temporarily storing accident contaminated debris and on-site timber. The area first had a protective sheet, then soil and then a top cover sheet.
— Transferring debris and waste containers to locations away from the site boundaries.
— Installing thin lead sheets to shield liquid processing components such as slurry transfer pipes and cross-flow filters.
— Using thicker shields and concrete box culverts for high activity processing components such as spent caesium absorption towers. The passageways linking box culverts use sandbags for shielding.

5.1.6. Ensuring essential facility structural stability

As buildings may be damaged after an accident, establishing and maintaining the integrity of structures and buildings is essential for ensuring safe isolation of radiation and confinement of contaminated material as recovery, cleanup and dismantling operations are being conducted. Following a major accident where buildings have been impacted by accident forces and/or have been contaminated, it is important to inspect, assess the damage, evaluate structural integrity, and take practical measures (such as reinforcement) to maintain integrity throughout the stabilization, preparation and decommissioning processes.

5.1.6.1. Inspections and surveys

Inspecting facilities and buildings for structural damage or weakening is needed to determine if and what measures may be needed to reinforce and maintain their integrity and confinement functions. These functions may be temporary at first because of urgent needs. More permanent measures based on inspections and evaluations conducted by experts with knowledge of current and future requirements for the facilities will follow. Evaluations include consideration of future degradation resulting from the impacts of the accident, such as deformations and displacements. It is also important to understand the consequences of emergency response measures; for example, evaluating the corrosive effect on long term integrity of introducing sea water.

Operational personnel should be trained to conduct regular surveys to assess structural integrity. This begins when structural integrity has been assured (with or without reinforcement measures) and continues to the end of the need for the buildings and structures. The survey methods need to be created with consideration of the site features (configuration, exposure), the kind of risks (or combined risks) that could increase the damage, chemical corrosion, erosion, settlement, weathering and other types of ageing.

This continual monitoring of the building and structures needs to be conducted along with periodic expert inspections and evaluations. Analytical and experimental research of the chemical and structural effects may be necessary to understand the impacts of the measures on long term integrity [7]. Conclusions of surveys should be recorded for trend evaluations. These records are also to be provided to the regulator.

5.1.6.2. Assuring the structural integrity of buildings

Assuring structural integrity of buildings over the long term is part of a broader need to assure performance of major safety functions. Where the accident has caused damage to SSCs that are needed for safety, actions might be taken to assure performance during occurrence of natural hazards. The effects of ageing need to be evaluated where the stabilization and recovery actions will require a schedule of many years.

Reinforcement may be required. For example, the lifetime of the Chernobyl NPP sarcophagus was limited to approximately 30 years. Because of challenges to the structural integrity, an NSC became necessary. The confinement structure is now being installed to provide ongoing containment of radioactivity to last at least 100 years [7].

SSC safety functions should not be lost as a result of natural events such as earthquakes, tsunamis, heavy rain, typhoons and tornadoes. It is especially important that loads created by the accident conditions are combined with those analysed for the most severe conditions during natural events. To provide examples of the verification of the capability of SSCs to withstand an earthquake, a tsunami, typhoons and heavy rain, tornadoes and ageing degradation, assessments and evaluations were conducted following the Fukushima Daiichi accident as described below [9].

Earthquake

SSCs that are needed for safety functions need to be provided with the appropriate seismic resistance in accordance with the seismic design guidelines. Diversity needs to be considered as needed. Based on evaluation results, TEPCO determined that there was no significant damage from the Great East Japan Earthquake to the reactor buildings, turbine buildings, and primary equipment and piping which requires seismic safety. TEPCO conducted additional assessments to determine if the buildings could withstand future large earthquakes. The assessments were conducted in accordance with the design basis earthquake ground motion concept based on the seismic design review guidelines.

Computer earthquake simulations that considered the situation of buildings were also performed. Based on these analyses, TEPCO concluded that the reactor buildings would be capable of withstanding a future large earthquake. To further enhance the safety margin at Unit 4, TEPCO installed a support structure at the bottom of the spent fuel pool. This increased the safety allowance (earthquake resistance strength) by an additional 20% [54].

Tsunami

TEPCO took measures against a postulated tsunami with the maximum height of 7–8 m caused by an outer rise earthquake. As a result of this study, all the pumps for water injection into Units 1–3 reactor pressure vessels were moved to higher locations by July 2011. The mobile emergency power sources, fire engines and other related equipment were also moved to higher locations. TEPCO also constructed a temporary sea wall of varying height of 2.4–4.2 m on the grounds at the 10 m level to protect the major buildings. In addition to the countermeasures for earthquakes and tsunamis, redundant and diversified facilities and equipment (e.g. power trucks, fire engines) were established for response to other events such as multiple equipment failures or off-site power losses. Based on the experience of the Great East Japan Earthquake, the Nuclear Regulation Authority developed new regulatory requirements. TEPCO determined the new earthquake ground motion and tsunami heights that needed to be considered as part of the backfit requirements for Fukushima Daiichi and implemented these as appropriate [54].

Typhoons and heavy rain

Buildings that contain highly contaminated water, such as reactor and turbine buildings, are evaluated using past meteorological data, according to the Building Standard Act. The buildings, and the systems installed in the buildings, are evaluated for their ability to maintain their function during typhoons. Regarding heavy rain, an estimation of the amount of contaminated water accumulated in the basement of the building was conducted. From the results, even if it is assumed that 1000 mm/day of rain falls, which exceeds past meteorological data in Japan, the water levels were evaluated to remain sufficiently low to avoid overflow.

Tornadoes

Buildings that have reinforced concrete structures are not expected to be damaged by tornadoes. The pumps for the reactor pressure vessel/primary containment vessel (PCV) water injection system are distributed to dispersed areas and it is considered that the risk for loss of the function of all pumps at one time due to a single tornado is very low. Even in the case that all the pumps simultaneously lost their capability, water injection could be provided by fire engines.

As for the power supply system, diesel generators are within reinforced concrete buildings. Motor control centres are within reinforced concrete or steel beamed buildings at dispersed areas to avoid simultaneous loss of function. Outdoor cables are installed to have multiple routes because they could be directly damaged by tornados. In the case that all the cables are damaged, dedicated generators are located at dispersed locations and could supply power to the SSCs. In addition, mobile power units are available. Similar evaluations are conducted for important safety related SSCs such as the spent fuel pools and the water treatment systems.

Ageing degradation effects

Assessments and evaluations are needed to verify the capability of SSCs to respond to ageing degradation effects over a long time period. Repairs and reinforcements of SSCs to ensure integrity need to be conducted when necessary and may include development of a corrosion control system for the key components, or other specific measures.

5.2. DEVELOPMENT AND DEPLOYMENT OF REMOTE TECHNOLOGY

The development and deployment of highly automated remote technology was in its infancy at the time of the accidents at A1, Windscale Pile1, TMI-2 and Chernobyl NPP. At TMI-2, manual defueling was chosen over development of complex mechanized equipment because the development time needed and the uncertainty for achieving a highly reliable, remotely operated system presented too much project uncertainty.

Since those accidents, there has been much progress in remote technology evolution and today there are many applications within the nuclear power industry to support operations and maintenance. For post-accident cleanup and decommissioning, Fukushima Daiichi will set many precedents for the application of remote technology to manage the condition. The sophisticated application of robotics will be a major aspect of decommissioning of

Fukushima Daiichi for the long term. The strong need for R&D and innovations for decommissioning of nuclear facilities, including those after an accident, are addressed in Refs [28, 29].

5.2.1. Project management perspectives

Current decommissioning systems typically employ many of the advanced approaches used in other nuclear and non-nuclear industries. This includes advanced electronics, the design of robots, virtual reality simulation and software that makes robots intelligent and adaptable to a variety of tasks. The development of industrial robots and remotely operated equipment in non-nuclear sectors has provided technological advancement, especially in those sectors that deal with hazardous materials (the chemical industry) and special and difficult tasks (the defence or space industries).

The application of remotely operated equipment for any decommissioning work requires a complex combination of highly skilled resources and a programme to conceptualize, design, fabricate, test and place this equipment in operation. Compared with repair and replacement work at non-accident NPPs, this complexity can be more demanding for post-accident work because of extreme conditions involving radiation and physical damage. Situations that make human access impossible include, but are not limited to, the following:

(a) High gamma radiation and alpha/beta contamination;
(b) Atmospheric conditions with noxious gases;
(c) High temperatures;
(d) Difficulty of access, such as in narrow spaces;
(e) Lack of visibility;
(f) Heavy loads.

There are non-accident situations that can be equally challenging, for example in fuel reprocessing facilities or nuclear legacy facilities. From the perspective of the project manager and project engineer, obtaining remote equipment for decommissioning tasks can be a comprehensive project on its own. Weeks or months may be required, the total duration depending on factors such as complexity of the task, whether or not adaptable components are available, or if considerable development will be needed. Once it is clear that the situation confirms the impossibility of human operations and the need for remotely operated equipment, evaluations are to be conducted for the technology to be applied. In order of preference, the first choice is to use the available commercial equipment, the second is to adapt existing equipment and systems for the conditions to be encountered. A final resort is development from the ground up for the specific need. The complexity of such undertakings is illustrated by the steps in Fig. 11.

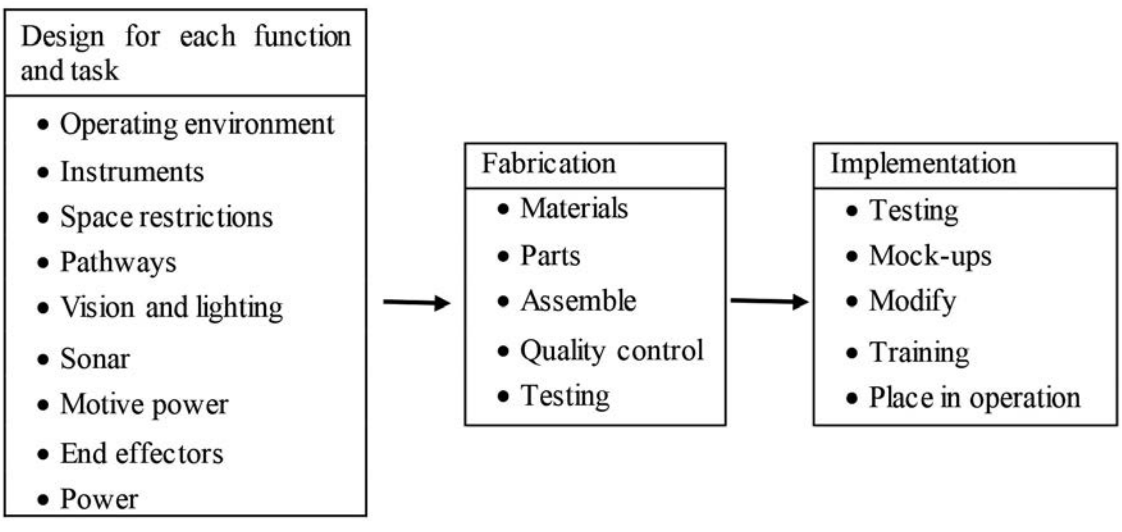


FIG. 11. Project steps for each remote technology application.

Some of the technical information needed to obtain remotely operated equipment includes the following:

- Specification of the operating method and scenarios that define the needed performance, functionality, safety and operating conditions;
- Identification of required supply (power, gas, hydraulic) that needs to be available;
- Radiation exposure, source term and associated dose rates to be taken into account;
- Space constraints, such as limited access, interferences, pathways to be traversed;
- Special needs, such as radio-controlled command, instruments (cameras, dosimeters) and types of end effector tools.

5.2.2. Considerations for choosing remote technology

From the preceding discussion of project management perspectives, it is not difficult to imagine the increased challenges for post-accident situations where extreme physical conditions and the many unknowns that may exist prior to deployment of equipment. In this regard, insights for what is needed are often best provided by those at the site who will use the equipment. Such individuals need to participate with engineers and suppliers at each step of design, procurement, development and deployment.

For each remote technology to be adapted or developed, in addition to photographic, video, radiation and other characterizations, the use of simulations, 3-D models, and physical mock-ups are key to development and demonstration of reliable performance prior to operation in the accident environment. These tools also provide a means to train operators in control of the remotely operated equipment, both generally and for specific tasks to be conducted in situ. Models and mock-ups should be retained as long as the situation for which they are built continues to exist.

The development of remotely controlled manipulators to be used in a highly contaminated facility includes the following:

- Recognition, characterization, quantification and sample analysis resulting in the creation of 3-D disposition and 3-D radiation models.
- Drafting of a preliminary decontamination plan including definition of the requirements for tools, supporting construction and use of remotely controlled manipulators.
- Decontamination and dismantling simulations for:
  - Specification of tools and supporting construction;
  - Specification of geometrical, mechanical and kinematics parameters for manipulators.
- Generalization of requirements, screening of existing tools, equipment and manipulators:
  - Modification of existing tools, equipment and manipulators;
  - Detailed specification, basic design and development;
  - Purchasing of commercially available tools, equipment and definition of requirements for their modification.
- Modification, improvement or manufacture of tools, equipment and manipulators (in cooperation with vendors).
- Factory acceptance tests to ensure that specification parameters and functions are met and to confirm reliability of equipment.
- Mock-up tests and operator training to provide:
  - Detailed descriptions of decontamination and dismantling operations;
  - Advice for operations;
  - Familiarity with tools, equipment, manipulators — design, operation, maintenance;
  - Training of decontamination and dismantling tasks — obtaining operating skills.
- Performing decontamination and dismantling tasks — feedback, improvement and further development.

5.2.3. Post-accident applications of remote technology

Applications of remote technology in post-accident situations are focused heavily on, but not limited to:

(a) Characterization using direct observation, data collection and sampling;
(b) Decontamination of walls, floors, structures, and equipment;
(c) Size reduction and retrieval of damaged fuel and fuel debris.

A few examples of characterization and decontamination are presented in the following sections. Many other devices for these functions exist for use throughout the nuclear power industry. Removal of damaged fuel and fuel debris is more unique to a post-accident situation [7].

5.3. CHARACTERIZATION AND MEASUREMENT

Characterization is a very important technical step in the planning phase of any decommissioning project — whether the facility was shut down following normal operation or after an accident. This section discusses how methods and techniques used to characterize facilities and plants have assisted with the planning and execution of decommissioning projects following accidents.

Following an accident, understanding of the structure of a facility and its radioactive content through the acquisition of knowledge, records and through data capture is essential in order to plan for safe, reliable and cost effective decommissioning. Hence, the characterization of a facility is an aggregation of many data sources. Characterization studies are normally conducted in two aspects.

Radiological characterization is conducted to determine the radiation levels to estimate dose levels for worker deployment, to determine the radionuclides present including fissile materials to estimate the types and quantities of wastes in the various categories, and to plan for waste transport and compliance with waste acceptance criteria for waste disposal [6].

Physical characterization under accident conditions is conducted to determine the status of damaged fuel and debris initially by visual inspection methods, and subsequently by sample removal and analysis. Without such characterization, it is not possible to plan removal operations in any detail. Additionally, it is necessary to determine the post-accident configuration and stability of existing structures to determine the route forward for decommissioning after the accident. For example, if safe storage is an option, it would be necessary to determine the integrity of structures that are essential for containment of the radioactive inventory in the longer term and alternatively, for immediate dismantling. Access routes for remotely operated equipment need to be planned in detail.

In an accident scenario, radiological characterization and physical characterization are often compromised by limitations on access created by the accident conditions. Firstly, the physical disruption of normal access routes may be evident, so it may not be possible to get to regions where inspections, measurements or sample taking can be made to support radiological characterization [55]. Secondly, the presence of high radiation fields may preclude even limited human access, necessitating the need for special remote tooling often constructed on a one-off, ad hoc basis to meet immediate and specific needs. Post-accident, assuming catastrophic fuel failure, the gamma radiation fields will be dominated largely by ^{137}Cs (~30 year half-life) with the result that the radioactive decay will be much slower than normally experienced in a reactor system shut down after normal operations where the decay will be dominated by ^{60}Co (~5.3 year half-life) in the near term. This means that even after 100 years in the accident situation, the gamma fields will only have reduced tenfold. Additionally, the presence of ruptured fuel will have spread actinide (alpha) contamination around the plant, necessitating special precautions for decommissioning operations, such as the use of air suits. In an accident scenario, it will be necessary to carry out characterization using an iterative approach as plant details emerge, access routes are developed and an improved understanding of the nature of the situation becomes evident. Regular review will be necessary; hence, the need for multiple approaches to characterization as the general picture builds.

While characterization is just a specific case of the potential for the need for innovative deployment and access solutions to support sample taking and measurements, it is worth stating more generally that, wherever possible, the philosophy should be to use commercially available, proven equipment that can be adapted to the job at hand rather than opting for costly and time consuming development work. In case of time constraints, it may be necessary to use improvised systems of available equipment for rapid deployment to enable basic data to be obtained quickly. On this basis, approaches can then be refined, once scoping observations have been made. In special circumstances, specific R&D may still be required since each accident will encounter unique challenges that will require adapting existing technologies to conduct characterization [7]. Such endeavours will lengthen deployment timescales.

5.3.1. Examples of characterization approaches under accident situations

Examples of how these various factors have determined the routes forward for characterization at the various accident damaged plants under consideration are given below.

At TMI-2, an extensive programme was applied including sample acquisition and examination to develop and implement a test and inspection plan that completes the characterization of the TMI-2 equipment that may have been damaged by the core damage events, and the TMI-2 core fission product inventory. The characterization programme included both sample acquisitions and examinations and in situ measurements. Fission product characterization involves locating the fission products as well as determining their chemical form and determining material association [56].

At Windscale Pile 1, the spread of contamination from failed fuel, mainly dominated by ^{137}Cs, precluded initial inspection of the fire damaged zone of the reactor core. At a later stage it was possible to use remote TV inspection of the discharge face of the reactor using a manipulator arm deployed into the fuel discharge void of the reactor. The inspections provided an initial view of the extent of the fire damage; the fuel condition also provided an early input to the plans for decommissioning operations. Much later, it was possible to deploy endoscopic inspection equipment into the fuel channels to determine further the condition of the remaining undischarged fuel [57].

For the Windscale Pile 1 reactor at Sellafield, the novel use of laser scanning equipment was used to generate a 3-D map of the internals of a contaminated cell to facilitate decommissioning planning [58]. Such technologies are very useful in the absence of as-built drawings or situations where drawings of an older plant can no longer be reproduced. In situations where facility entry is difficult because of high radiation fields and/or limited physical access for personnel, laser scanning in conjunction with gamma cameras can provide essential data on structures and radiation fields to facilitate decommissioning planning.

At Bohunice A1 NPP, various advanced technologies were used for support of physical and radiological decommissioning characterization. A 3-D laser scanner, SOISIC, together with modelling software packages, EUCLID and 3Dipsos, were used for the creation of 3-D models and also for the verification of the as-built state of selected systems and premises. Interactive Graphics Robot Instruction Program software was used for computer simulations of some difficult dismantling and decontamination tasks [59].

Later, within a Slovakian national IAEA Technical Cooperation project, a more advanced type of 3-D laser scanner, Callidus, was delivered together with appropriate software for creating virtual reality type representations of indoor hostile rooms. A 3-D laser scanning-panoramatic photo, cloud of points and 3-D model is shown in Fig. 12. In addition, further radiological modelling and dose predictions used the 3-D ALARA[3] tool called Visiplan, based on available information about radioactive sources and contamination of the plant structures [60]. This approach was necessary because of missing drawings and the fact that existing documentation did not reflect the as-built state, in addition to the issues of high levels of contamination and dose rate in the scanned areas. The benefits were the possibility to directly create 3-D as-built models, to simulate and optimize dismantling activities, to substantially reduce occupational doses and to save the time needed for implementing interventions.

At Fukushima Daiichi, it is difficult to access some parts of accident affected buildings and PCVs because of damaged facilities and rubble caused by the tsunami and hydrogen explosion. Accumulated contaminated water originating from leaking reactors, the in-flow of underground water and high radiation from these obstacles also makes the access difficult.

Several steps are necessary for both radiological characterization and physical characterization, including the following:

[3] As low as reasonably achievable (ALARA) is the process of determining what level of protection and safety makes radiation exposures, and the probability and magnitude of potential exposures, as low as reasonably achievable, economic and social factors being taken into account.

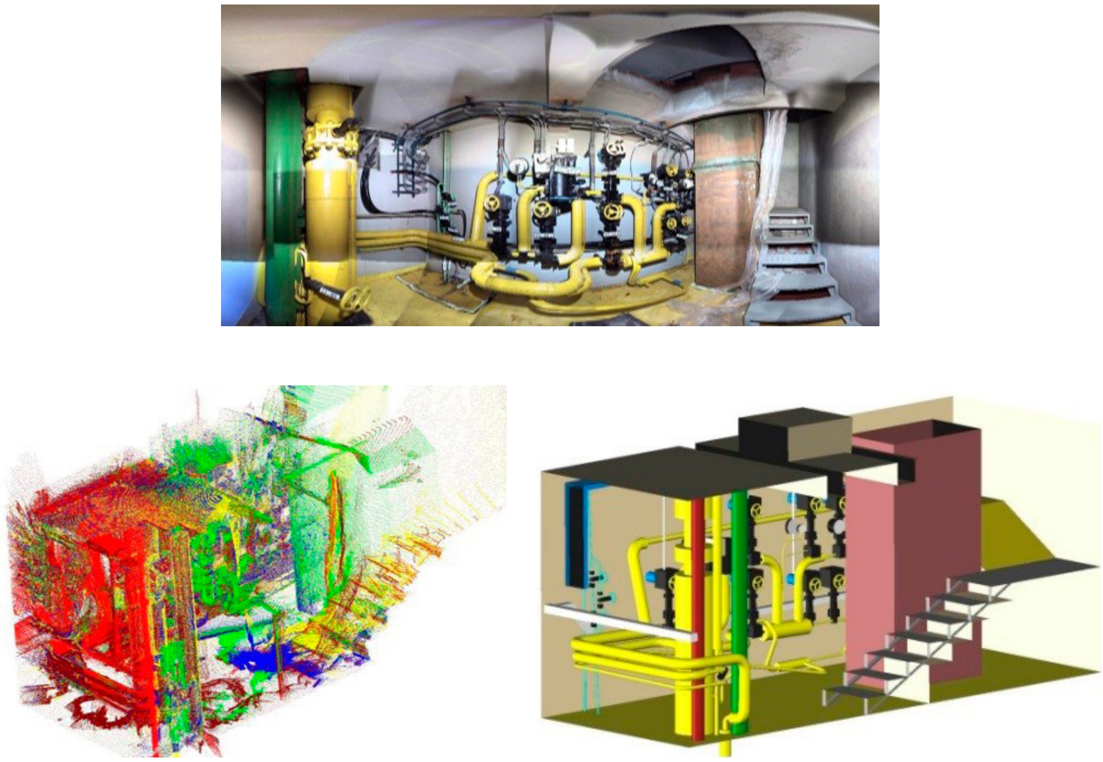


FIG. 12. 3-D laser scanning-panoramatic photo, cloud of points and 3-D model. (Courtesy of Nuclear Power Plant Research Institute (VÚJE) Trnava, Slovakia.)

— Identification of damaged points in buildings and repair work to stop underground water in-flow;
— Identification of damaged points in PCVs and repair work to stop reactor cooling water leaking;
— Pumping up of accumulated contaminated water and drying out of buildings;
— Removal of rubble;
— Repair work of facilities;
— Measurement of radiation and decontamination in buildings;
— Access to PCVs;
— Investigations in the reactor pressure vessels and PCVs.

Another example of 3-D modelling based technology that could support careful planning of decommissioning activities at Fukushima Daiichi Units 1–4 is discussed in Ref. [61].

5.3.2. Remote technology applications

Two early examples of mobile devices were used at TMI-2 for the characterization of areas in air and underwater. TEPCO began using a high-access survey robot to survey the lower floors of the reactor building of Unit 2 at the Fukushima Daiichi NPP in June 2013. The robot was specifically developed for the task by a private company and Japan's National Institute of Advanced Industrial Science and Technology [62].

The high-area accessible crawler work platform was developed by Japan's National Institute of Advanced Industrial Science and Technology, while a private company developed the survey performing robot arm installed on it and equipped it with 3-D display of the surrounding structures, simultaneous control of 11 joints and shock absorption in case the arm touches obstacles. The high-access survey robot can be operated through a 400 m long fibre scope or wireless communication from a remote place for less radiation dose to operators. It can continue working for five hours by using the battery. The crawler work platform — measuring 1.8 m by 0.8 m and weighing some 1100 kg — can travel at speeds up to 2 km per hour. It has a height of 1.8 m while being transported or travelling, but the 1.7 m robot arm can extend up to 7 m [62].

The private company that developed the robot arm said that it “can easily approach hard-to-see objects that are behind other objects in a structurally-complex environment in the reactor building by applying simultaneous control on multiple joints” [62]. The robot employs a zoom camera, laser range finder and dosimeter at the tip of the arm to create detailed images, collect 3-D data and identify sources of radiation. The information gathered through the use of the robot will be used for the future planning of decontamination, shielding, PCV investigation and repairing works [62].

A remote-controlled crane was developed to lift fuel assemblies out of a storage pool at Unit 3 of the Fukushima Daiichi NPP. The two arms of the robotic crane can also remove rubble that might block its path to picking up fuel assemblies and has the versatility to be able to cut some rubble into smaller pieces [63].

The remote-controlled robot crawler crane (see Fig. 13) with gamma cameras is an example of robot development to accomplish tasks to investigate areas significantly contributing to radiation dose rates (hot spots) on the first floor of the reactor building Units 1–3: This crane measures 2.3 m by 0.7 m with a height of 1.4 m, weighing 1250 kg and can travel at speeds up to 1.5 km per hour.

An additional three examples of devices (Rosemary, Sakura and Quince) developed by TEPCO for characterization purposes at Fukushima Daiichi are shown in Fig. 14.

These were designed for:

— Investigating radiation sources on the second and third floors of the reactor buildings (Rosemary and Sakura [64]);
— Conducting radiation measurements and 3-D laser scanning inside the reactor buildings to create images of structures and other items;
— High reach for video imaging of the upper floor of the reactor buildings such as the operations floor (Quince).

More examples of robotic technologies developed in Japan are available through information sources and references of IRID [65, 66].

A remotely operated mini submarine developed by France’s (then) AREVA was delivered to Japan. The submarine carries a probe and takes readings underwater so that a map of contamination can be drawn. Teams from the Japan Atomic Energy Agency are being trained to operate the submarine, after which it will be used to map the lakes, reservoirs and rivers within the 20 km evacuation zone around the Fukushima Daiichi NPP [62].

Another example of a remotely operated device comes from the decommissioning of A1 NPP. Figure 15 shows one of several robotic arms, designated MT-80A, during cutting of small pipes for dismantling of the heavy water evaporator. Six different cutting and end effector tools are possible: handling effector, hydraulic shears, reciprocating saw, electromagnetic effector, circular saw and sampling device.

FIG. 13. Remote-controlled robot crawler crane. (Courtesy of International Research Institute for Nuclear Decommissioning (IRID), Japan.)

FIG. 14. Fukushima Daiichi's Rosemary, Sakura and Quince. (Courtesy of TEPCO, Japan.)

FIG. 15. Remotely operated robotic arm MT-80A. (Courtesy of VÚJE, Slovakia.)

5.4. DECONTAMINATION OPERATIONS

Decontamination within the site and facilities needs to be done to create a working environment that allows even limited access for decommissioning staff to implement further actions. Contamination type and on-site hot spot locations are not known post-accident, so proper characterization needs to be done beforehand. Various decontamination techniques may be used; however, those more appropriate to be deployed remotely are preferable, such as high pressure water jet decontamination, dry ice blasting or metallic abrasive blasting.

There may be significant sources of contamination located outside the reactor buildings or inside on the floors and walls in the form of debris and dust generated during an accident. These need to be remotely and mechanically removed in the first instance to allow further decontamination of various surfaces. At the Chernobyl NPP, there were several mobile machines adapted for placement of highly contaminated debris; two of these are shown in Fig. 16. Figure 17 is a modern version of the scabbler used for decontamination at TMI-2 for scabbling concrete and collecting it into drums with a vacuum system.

5.5. DISMANTLING AND MANAGEMENT OF GENERATED WASTE

Dismantling activities need to be planned and effectively implemented as an integral part of the entire decommissioning process, taking into account further waste management in terms of amount and types of non-radioactive and radioactive waste [10, 11].

There is a need to establish a waste management strategy and a long term waste management plan by estimating volumes, types and characteristics of different waste streams and by identifying optimized waste management scenarios even though end points for waste might not be defined yet. The required/applicable end state for the site and affected areas (greenfield versus brownfield) will affect quantity and type of cleanup wastes. It is important to address interdependencies between particular steps and setting of long term priorities.

FIG. 16. Remote-controlled decontamination machines used at the Chernobyl NPP. (Courtesy of Chernobyl NPP, Ukraine.)

FIG. 17. TMI-2 scabbler.

Interfaces with other relevant programmes (nuclear materials accounting and safeguards) and with other bodies (civil emergency management authorities, local and State ministries, neighbouring States, international organizations) need to be identified. There will likely be a need for a specific and focused R&D programme. Good practice versus bad practice and lessons learned from previous nuclear and non-nuclear events need to be considered [7].

Waste acceptance criteria need to be defined for each step of the waste management programme including the management of non-compliance with existing criteria and/or new criteria for new facilities. Generic waste acceptance criteria may be needed if the end point is not known yet. The availability of sites for disposal need to be considered taking into account the use of existing alternate disposal routes (e.g. municipal or industrial landfills), management of large volumes and diverse waste types, the need for new facilities and time frames for implementation.

Infrastructure (on-site and off-site) needs to be developed or further updated including the possibility of adapting nuclear and non-nuclear facilities, equipment and vehicles for new purposes. Comparisons and feasibility studies of centralized or local solutions versus mobile/transportable technologies are advised to be developed, noting the status of existing and new dedicated facilities (e.g. mobile treatment systems, incinerators, cementation, compactors or sorting and segregation systems).

Staff availability and training need to be planned well in advance, taking into account various aspects such as dose constraints, qualifications, availability of contract staff to augment local staff or accommodation for new staff.

Optional cost estimations (cost–benefit, affordability) need to be proposed and appropriate financing needs to be ensured. Communication and public involvement need to be addressed and considered during the decision making process [11].

5.6. WATER/LIQUID WASTE

When the integrity of reactor systems remains intact, as has been the case for INES Level 4–5 reactor accidents, contaminated water is usually retained within the plant's containment, other buildings, systems and tanks. When the reactor cooling system remains operable, it may be used for cooling the core and fuel debris. However, even when the water is contained, installation of special equipment and systems may be needed for cooling and for water treatment during the phases of stabilization, damaged fuel recovery and plant cleanup. These will require engineering and operational actions to process the water for radionuclide removal and to manage the water inventory in terms of its amount. Eventually, such water might be cleaned to release standards with the resulting concentrates stored or considered to be disposed.

In the case of INES Level 6–7 reactor accidents, as a result of temperature and pressure excursions, the reactor vessel and systems, containment and other buildings may be damaged. Contaminated water can leak into various substructures resulting in significant complications of subsequent activities. Immediate actions are required to mitigate and prevent the release of the contaminated water throughout the plant or into the surrounding land and water. Measures are required to find pathways for release and leakage. These pathways are then isolated through a variety of methods. Depending on ease of access and radiation levels in these locations to be isolated, the only options may be detection, characterization and sealing with remotely operated equipment.

At these accident levels, influx of water from rain, groundwater and sea water to the plant and reactor presents its own set of problems. This clean water can become co-mingled with contaminated waters, thereby increasing the volume of water that must be treated and managed. Introduction of sea water will potentially cause a corrosive effect on systems and tanks.

5.6.1. A1 NPP

An example of special needs in treatment of non-standard liquid waste generated during the post-accident phase is the Slovakian A1 NPP case. Spent fuel assemblies were stored in storage casks, which were filled with a coolant. 'Chrompik' (a water solution of potassium chromate and dichromate) was used as a cooling medium in some casks from the 1970s until 1995. Later, an organic coolant 'dowtherm' (organic liquid mixture of biphenyl and biphenyl oxide) was used because Chrompik proved to be a very unsuitable storage medium for spent fuel assemblies. It caused corrosion of cladding and, because Chrompik is a water based solution, radiolysis led to hydrogen generation and also caused cladding damage to many spent fuel assemblies. These defects and incidents led to severe contamination of coolant and stored assemblies.

Spent fuel assemblies were repacked and transported to the Russian Federation for reprocessing in the 1990s. The Chrompik was transferred to storage tanks. The stored Chrompik is a liquid radioactive waste that in 2015 contained fission products (^{137}Cs and ^{90}Sr) with a volumetrically concentrated beta activity of 100 GBq/L and alpha emitting nuclides (^{238}Pu, ^{239}Pu, ^{241}Am and others) with a volumetrically concentrated activity in the range of 4–6 MBq/L. There is sludge at the bottom of the Chrompik storage tank that primarily contains chemical corrosion products of the fuel element cladding, chromium (III) from the reduction of chromate from Chrompik, metallic uranium and corresponding amounts of plutonium (depending on the stage of fuel element burnup).

The dowtherm contained dispersed metallic uranium together with a smaller amount of corrosion products in the form of a fine dispersion. Liquid waste from spent fuel storage casks was of a specific character, requiring special measures and the application of other technological procedures for its treatment and conditioning. These measures and procedures differ from those for processing common radioactive liquid of this type of waste owing to the following:

— Relatively high specific activity;
— Presence of plutonium and other transuranic elements;
— Non-homogeneity (considerable amounts of solid particles in the form of sludge);
— Presence of an organic liquid (dowtherm).

Part of the Chrompik with activity up to 1 GBq/L has been vitrified because of its radioactivity level and chemical composition. Chrompik with higher activity will be processed in a vitrification line, modified and technically improved based on the previous operational experience and needs to enhance shielding because of high radiation.

Dowtherm with activity up to 200 MBq/L was incinerated. If the activity was higher, it was decontaminated by ion exchange resins and then incinerated. Spent adsorbents were solidified with geopolymer within containers.

A certain volume of dowtherm and sludge remains in casks that contained spent fuel. After retrieval of the residue from the casks, it will be characterized and a suitable matrix for the conditioning of these residues will be chosen. Several matrices such as cement and geopolymer are being assessed for sludge solidification.

5.6.2. Three Mile Island Unit 2

At TMI-2, the core disruption did not seriously damage the reactor vessel. The rupture disk of the reactor coolant drain tank did burst during the accident, and several thousand cubic metres of coolant spilled onto the containment basement floor, and then to the basement of the auxiliary building. Influx of river water into the building through the service (technical) water system relief valve increased the volume of this contaminated water. During the course of cleanup activities, the volume of the contaminated water reached about 10 000 m^3, which was recycled via cleanup processing for shielding and cooling [7].

The emergency stabilization measures of the first few months were successful and the water was contained in tanks, systems and sumps. Initially, the most important issues were tracking the water volumes and minimizing any increase because of the dire shortage of spare tankage. When this was accomplished, the challenges were how to collect the fission products in the water for safe handling, storage, reuse and disposal. Subsequent water management can be viewed as four phases:

(1) The project team acted to stabilize the situation by isolating and controlling the water, transferring and cleaning it when possible, and analysing its characteristics to increasing levels of accuracy. This phase lasted until the project team realized that control had been established.
(2) A large scale cleanup was needed to process the water and capture the radionuclides. Over a two year period, the project team constructed two new water processing systems. One of these was a resin ion exchange system and the second was a zeolite system primarily for capture of radioactive caesium. Over 3.7 million L of new tankage for processed water and 1120 m^3 of new solid waste storage space for the resin and zeolite vessels were established.
(3) The water had to be maintained in an acceptable condition for reuse or disposal. Reuse was primarily for shielding within the reactor vessel during fuel debris removal and for decontamination throughout the plant. This meant continued operation of the above two systems plus constructing a third new system to process the water used during defueling.

(4) Twelve years after the accident, open cycle evaporation was chosen to dispose of the accumulated water, instead of direct discharge of water even though it could meet regulatory standards. The reason was because of stakeholder concerns about tritium content. The basis for deciding on evaporation is described in the discussion of tritiated water disposal at the end of this section.

5.6.3. Chernobyl NPP

Although the upper part of the Chernobyl NPP Unit 4 reactor building was blown off by the explosion, more than half of the fission product inventory and most of the fuel materials remained in the core. To cool these materials and cover them to control the release, lead, dolomite, boron carbide, clay and other materials were dropped from helicopters. These materials hit the remaining building structure. They may have affected the composition of the accumulated water in the shelter [7].

During the years after the accident, rainwater and groundwater washed into the core of Unit 4, via various paths. It reacted with the lava-like FCM and structures, creating a huge amount of contaminated water. The influx of rainwater and melted snow into the shelter was estimated to be on the order of 2200 m^3 a year. In addition, almost the same amount of water was accumulated due to the condensation of humidity and water spray operation inside the shelter to suppress the dust. The total amount of the accumulated water inside the shelter is estimated to be about 10 000 m^3, and several hundred cubic metres of that is thought to have leaked into the ground. As a countermeasure to protect the Prypyat river from contamination by the radioactive sources in the evacuation zone, including the shelter, a clay underground water shielding wall of about 13 km in length was constructed downstream of the facility. Also, many wells were sunk along the wall to pump out groundwater. But the effect was not clearly confirmed since the area concerned was too wide to be covered by this wall [7].

Approximately 20 000 m^3 of liquid radioactive waste has been accumulated and stored in special facilities at the site. Plans are to process this liquid over a period of 10–15 years using cementation, followed by disposal in a near surface repository.

Sources of moisture penetration inside the shelter object are both natural and human-made and include the following:

(a) Atmospheric precipitation penetrating inside the object through leaks in the encasement with the area of about 100 m^2. Annual precipitation is estimated in approximately 2200 m^3.
(b) Condensate formed in the summer in a volume up to 1650 m^3 due to differences in temperature and moisture content of the ambient air and indoor air in the lower levels of the unit.
(c) Operation of the regular dust suppression system during which the volume of the solution sprayed into the space under the central hall roof is about 270 m^3 per year.

Water penetration into the shelter object premises can create safety and operational problems. These include migration of contamination, structural degradation of the shelter, and disruption of diagnostic systems. Also, water enters into a chemical reaction with the backfill materials, construction materials and FCM and destroys them. This is followed by dissolution and transport of long lived radionuclides and fissile elements. These processes form highly alkaline carbonate solutions, so-called ‘unit’ water. According to studies in 2002–2003, there was an increase in the concentration of radionuclides (except for ^{137}Cs) and fissile elements in the unit waters, caused by the lava-like FCM destruction process, followed by leaching of radionuclides. These processes result in increased migration of radionuclides within the shelter object premises. The highest danger is uncontrolled flows of unit water into the Unit 3 premises. This water poses a real environmental threat in case of leakage from the shelter object.

This water is collected in the lower levels of the shelter object and contains high concentrations of transuranic elements and organic dust suppressant compositions used in the shelter object dust suppression system. A flocculation/coagulation method is used for capture of transuranic elements.

Uncontrolled water flowing from the shelter object into the bottom levels of the neighbouring Unit 3 represents the highest hazard. This is about 300 m^3 per year; it does not meet the acceptance criteria at the existing liquid radioactive waste management system at the Chernobyl NPP due to high content of organic substances and transuranics.

With the support of the IAEA and the European Commission, a facility is being created for pre-treatment of water that comes from the shelter. The method used is polymer flocculation. After transuranic isotope pre-treatment, this water will eventually be processed in a conventional facility to capture the remaining liquid waste with grout as a disposal media. In 2008–2011, a successful project was conducted to build a pilot installation and technology for liquid radioactive waste purification from transuranics and organics; the pilot equipment is shown in Fig. 18.

Design, installation and testing of a pilot facility was conducted for purification of shelter object water not meeting the acceptance criteria at evaporation installations and liquid radioactive waste storage facilities. A promising technology for shelter object contaminated water purification that can handle both the dust suppression organics and transuranics was developed at this facility. The developed technology is based on coagulation and deposition processes and ensures the following purification factors:

(a) Purification factor from transuranics is > 10 000;
(b) Purification factor from organics is > 1000.

A facility for pre-treatment of water coming from the shelter object was designed on the basis of successful results obtained in the pilot facility. After pre-treatment to remove transuranic isotopes, this water will be treated at a conventional plant for liquid waste.

The Chernobyl NPP water purification plant reduces the water volume by evaporation. During this process, transuranic concentration is further increased, and organic concentrate settles inside the evaporator. Over time, the evaporator becomes inoperative.

Existing and newly generated radioactive waste includes evaporator concentrates, spent ion exchange resins and perlite filter pulp. These materials are planned to be treated at a liquid radioactive waste treatment plant and solidified with cement. Most of the liquid radioactive waste is evaporated concentrate. Prior to its transfer for solidification, evaporator concentrate will be further evaporated at the liquid radioactive waste treatment plant.

5.6.4. Fukushima Daiichi NPP

Sea water was injected into the core of Units 1–3 of the Fukushima Daiichi NPP, with fire engines or cement pumping cars, since the normal cooling systems were lost. As a result, the impact of the salt on the corrosion of structure materials had to be assessed and measures were taken accordingly to keep the integrity [7].

> “When the containment at Fukushima Daiichi Units 1–3 was breached during the accident, highly contaminated cooling water leaked and flooded in the basement of reactor buildings and turbine buildings. It then leaked into the drainage ditches discharging to the quay. The leakage paths were quickly determined and blocked with sealants. To preclude further distribution of leaked water, silt fences were set in the quay.

FIG. 18. Pilot facility for shelter object water treatment. (Courtesy of Chernobyl NPP, Ukraine.)

“Subsequently, it was realized that undetected paths existed between the accumulated water in the basement and the groundwater around the reactors; the accumulated water was pumped out to keep its level lower than the groundwater. The pumped water was sent to a treatment system for decontamination, which is described below. A part of the treated water was sent back to the core for injection cooling. The remaining part was stored as excess” [7].

The inflow of the ground water was about 400 m^3 per day as of 2013 (and beyond), and this raised the volume of the treated water rapidly causing a significant challenge to storage construction [7].

The treated water volume stored in on-site tanks was more than 1 100 000 m^3 as of November 2018, and is increasing. Additional contaminated water has decreased step by step to approximately 200 m^3 per day owing to the multiple countermeasures against the groundwater ingress into the damaged facilities [67] .

There are many challenges involved in management of such a significant amount of accumulated water, such as the increase in the quantity of water requiring storage, difficulty in getting approval of government and other stakeholders for discharge of treated water, and the management of secondary waste from water treatment. It is necessary to find a sustainable solution to the problem of managing contaminated water and this would require consideration of all options, including the possible resumption of controlled discharges to the sea. An assessment of the potential radiological impact to the population and the environment arising from the release of water containing tritium and any other residual radionuclides to the sea needs to be performed in order to evaluate the radiological significance and to have a good scientific basis for taking decisions [39–41].

TEPCO’s strategy for managing contaminated water accumulating in the buildings includes pumping of the water from the turbine buildings to the centralized radioactive waste treatment facility buildings and then treating to remove caesium isotopes using two parallel systems, namely two caesium adsorption apparatuses. Removing the gamma emitting caesium isotopes as the first step facilitates further use and management of the treated water. These two systems have been enhanced by adding a strontium removal capability. Following caesium removal, the water is treated, to remove salt (NaCl) using the reverse osmosis process. Approximately half of the feedwater is desalinated and used for cooling of the damaged cores of Units 1, 2 and 3. The remaining half is a concentrated salt solution that is highly radioactive, containing mainly ^{90}Sr. This solution is stored in above ground tanks. Three multinuclide advanced liquid processing systems (existing, improved and new high performance advanced liquid processing systems) are being used to treat the highly radioactive water to remove 62 radionuclides (including ^{90}Sr) to below or near detectable levels (see Fig. 19). A number of additional systems have also been deployed to remove ^{90}Sr alone from water stored in the bolted flange type tanks. The plan is to treat all the amounts of stored water using caesium removal equipment, the desalination facility and multinuclide (except tritium) removal systems [41].

“To keep Units 1–3 in cold shutdown, a circulating injection cooling system loop of about 4 km in length was constructed and has been operating. With this system, the contaminated water is pumped out from the turbine building and returned to the reactor for injection cooling after decontamination and desalination. Absorption and/or precipitation methods are used for decontamination. It should be borne in mind that processing and disposal of the used absorber and sludge of fairly high radioactivity will ultimately pose a significant challenge for the waste treatment system. After decontamination, the water is desalinated by ion exchange or evaporation. Dealing with the excess water is a significant and ongoing issue. Also, storing and processing the condensed salt water remains an issue” [7].

“To cope with this problem, further measures are being considered and tested, such as applying further decontamination to the excess non-radioactive water, which is then routed to temporary storage tanks pending a decision on whether it could be discharged safely into the environment, or constructing a groundwater bypass from the upstream to the sea to reduce incoming flow. In addition, a water sealing wall is being constructed at the quay as a precaution. As a way of improving containment, future plans include locating the leakage path and appropriately sealing the vessels or the buildings, and installing a closed-loop cooling system within or alongside the reactor buildings. To realize this plan, it will be necessary to establish a good working environment for equipment installation and other activities by providing effective cleanup and/or shielding” [7].

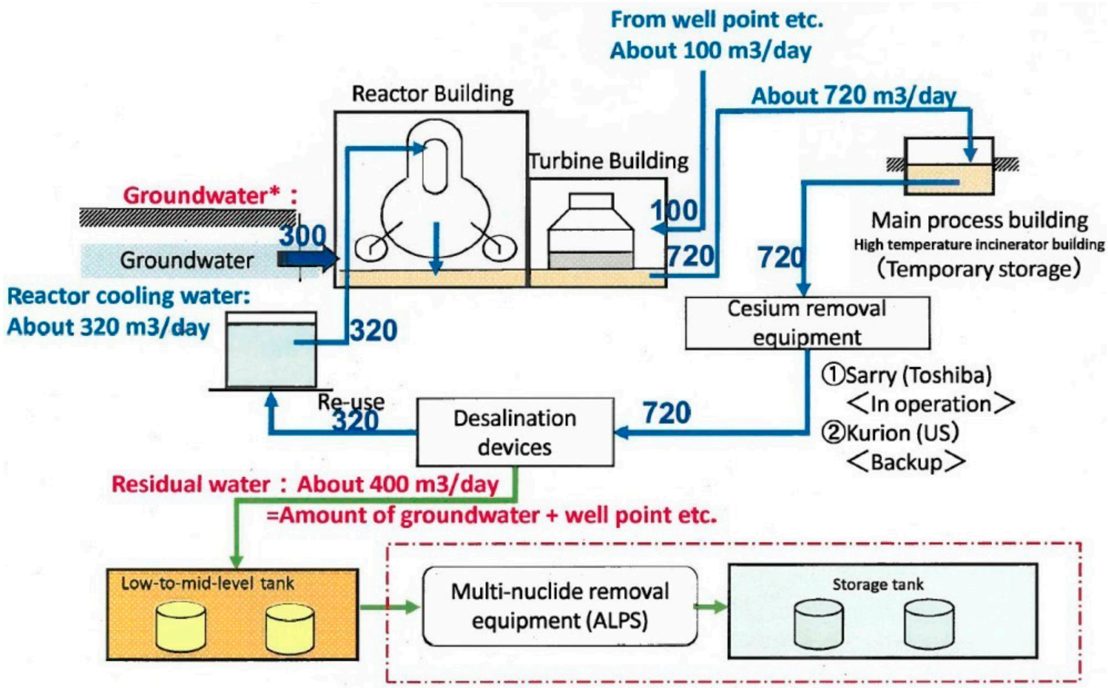


FIG. 19. Overview of contaminated water management showing water volumes as of March 2015, Fukushima Daiichi, Japan. (Adapted from Ref. [41].)

Later in 2013 and 2014, new leakages to the quay were found. It was suspected that some of the contaminated water left in the drainage ditches came out due to the groundwater flow that does not go into the reactor/turbine building. To address this problem, immediate countermeasures such as bypassing some of the groundwater, removing the contaminated water from the ditches, and enclosing the contaminated soil with sodium silicate walls were taken, in addition to fundamental measures [41].

5.6.5. Storage of radioactive water and liquid waste

At both TMI-2 and Fukushima Daiichi, concerns about the discharge of water containing tritium created a need to construct water storage tanks. At TMI-2, because the total inventory of water within the facility did not increase significantly after the stabilization phase, the added tank capacity, which occurred in two steps, was all that was needed. The first step was installation of tanks within an empty spent fuel pool to hold highly radioactive water as a stage in processing through the zeolite system. The second step was the added capacity of four million litres in two large outdoor tanks that held processed water.

At Fukushima Daiichi, the water situation is vastly different than at TMI-2 for a combination of reasons. The magnitude is greater as there are three damaged reactors. Much of the fuel was operated for years, the result being that the fission product content was considerably higher. There are water leakage paths outside the primary containment for some of the reactors. And there has been, and continues to be, a significant influx of uncontaminated groundwater and precipitation that becomes mildly contaminated and must also be processed [41].

This greater volume combined with not being able to release processed water containing some tritium to the sea has resulted in about 1000 tanks being built at Fukushima Daiichi to hold the water. Individually, the tanks are smaller than the TMI-2 tanks; however, the greater number of tanks has occupied a very large part of the area at the Fukushima Daiichi site.

Other types of facilities used for collection and storage of different types of liquid radioactive waste streams (underground tanks) were used during operation and the post-accident phase of A1 NPP in Slovakia. Several underground outdoor tanks (diameters ranged from 6 m to 16 m, different internal structure) were constructed from concrete with a special polyester glass reinforced laminate coating. Tanks were used for storage of water contaminated during operation of A1 NPP and later for collection of waters from decommissioning works. The integrity of the tanks became an issue after many years of operation and leakage occurrence of liquid radioactive waste to the environment was a possibility. Moreover, a layer of sludge was created on the bottom of each tank. Based on physical characterization, it was found that some other small pieces of solid material and objects were thrown into the tanks during their operation [68].

A special remote handling technology was developed to retrieve the waste and decontaminate the underground tanks. It was a challenging task considering the very small input opening for the inspection access (approximately 540 mm × 540 mm), through which a manipulator was inserted into a tank, and the need to develop remote handling manipulators flexible enough to decontaminate the set of underground tanks with various diameters and internal structures. A special long-reach manipulator named DENAR-41 and appropriate robotic arms were constructed, tested and successfully used to implement necessary retrieval and decontamination activities [69, 70].

Generally, the stored liquid radioactive waste and bottom sludge were non-homogenous, with very adhesive properties and high specific activity. Therefore, a special movable cementation facility (MCF) was developed, designed and manufactured for treatment of waste stored in the underground tanks [71].

The MCF can be easily transported and erected near any tank containing sludge. The main components are installed in four ISO certified containers, which are provided with autonomous ventilation and filtration systems. The entire MCF is remotely controlled from the control desk located in a separate container. This configuration allows treatment of sludge with relatively high activity. The MCF is designed for a maximum final dose rate on the surface of drums with immobilized sludge up to 30 mGy/h [71].

Radioactive sludge is pumped from underground tanks by the sludge pump, which is attached to the arm of the DENAR-41 manipulator. This manipulator can manoeuvre the sludge pump over the entire bottom surface of the tank. The MCF allows waste immobilization in different kinds of inorganic matrices. It employs in-drum mixing resulting in a final waste product, fixed in a stable matrix, stored in 200 L drums [71]. Figure 20 shows the manipulator DENAR-41 installed above underground tank and its connection to the MCF.

5.6.6. Tritiated water issue

Once water has been treated to concentrations below regulatory release limits, the approach to its release may be similar to the release during non-accident situations. A prudent technological approach is to establish monitoring frequencies and time delays for off normal concentrations and flow rates that will provide a higher degree of assurance of proper release of treated water. This may require more sophisticated instrument and control systems than would otherwise be the case.

FIG. 20. Long-reach manipulator DENAR-41 and MCF at the A1 NPP. (Courtesy of VÚJE, Slovakia.)

5.6.6.1. *Tritium in treated water*

Each atom of tritium in contaminated water substitutes for one of the hydrogen atoms in a water molecule. Water with tritium is referred to as tritiated water. Standard processing methods such as ion exchange for removal of radionuclides from contaminated water will not remove tritium.

Over time, the on-site quantity of tritium and tritiated water is reduced because of radioactive decay; tritium has a 12 year half-life. Another method for reduction experienced at TMI-2 was that the spraying of treated water for decontamination resulted in some evaporation, which was then removed via ventilation exhaust pathways. Because the concentration of tritiated water molecules is extremely low, evaporation removes it preferentially compared with the decontamination spray water that did not evaporate and remained to be treated again.

The concentrations of tritium are low in water contaminated as a result of an accident, and with dilution and metred release, can be discharged to rivers and oceans well within regulatory limits, as is the normal practice of facilities such as those reprocessing spent fuel.

5.6.6.2. *Stakeholder concerns*

At both TMI-2 and Fukushima Daiichi NPP, there have been stakeholder concerns for discharge of water with tritium regardless of the low concentration. At TMI-2, the downstream communities were against the prospect of tritium in their drinking water regardless of regulatory limits and negligible health effects. Because of stakeholder concerns, a decision was made to not discharge water to the Susquehanna River, even though it could have been done within the drinking water standard concentrations. Although it cost more operationally, the uncertainty of an outcome, the cost, and the time to resolve the legal conflict led to the decision to use the evaporation method [72].

At Fukushima Daiichi, there is concern for disposal of processed water to the ocean related to consuming fish from the area. The decision on how to dispose of the water is beyond the scope of this publication. The impact of not being able to discharge processed water at Fukushima Daiichi is much greater than it was at TMI-2 because of the greater quantities, continual in-flow from groundwater and precipitation that is adding to the total, and the resulting need to continue building storage tanks.

5.6.6.3. *Options for tritiated water disposal*

At TMI-2, several methods were considered for disposal. Nine options for which the environmental impacts were evaluated by the NRC are listed in Table 2. The owner decided to use one of these, which was to evaporate the water using an open cycle evaporator [72].

An additional fifteen alternatives were considered, but were eliminated from further evaluation as being less desirable from a technical standpoint, or clearly inferior to the other alternatives that received more detailed consideration [72]. Some of the options that the owner rejected without further evaluation included the following:

(a) Open air evaporation ponds. These have been used at the Savannah River Site in the United States of America. This method was rejected at TMI-2 because without creating a complex facility and systems, there couldn’t be absolute assurance that rain and snow would not overflow the ponds to the river.

(b) Transport by truck to the Atlantic Ocean and disposal in accordance with standards. The logistics of trucking combined with routes through densely population areas and the anticipation of public outcry led to rejection without serious consideration.

(c) Deep well injection on-site. This would have required knowledge of the geology and pathways below the area of injection, but this information was not available. Deep well injection off-site had the same trucking logistic issues. This method has been used at the Hanford site in the United States of America, where feasible.

(d) Isotopic separation of tritium. This method was determined to not be economically viable for large volumes of water with low concentrations. This conclusion was reached for the TMI-2 tritium based on discussions with staff at facilities in Canada.

TABLE 2. OPTIONS FOR TMI-2 TRITIATED WATER DISPOSAL EVALUATED BY THE NRC [72]

Option evaluated	Tritium	Borate
Evaporation, solidification of concentrates and disposal at a licensed burial ground	Atmosphere at TMI	LLW burial ground
Evaporation, solidification of concentrates and retention on-site	Atmosphere at TMI	TMI site
Distillation, solidification and disposal of concentrates followed by river discharge	Susquehanna River	LLW burial ground
Off-site evaporation at a government site	Atmosphere at the government site	Burial at the government site
Permanent on-site storage of solidified waste	Atmosphere at TMI	Ground at TMI site
Solidification and disposal at a commercial LLW site	Atmosphere at TMI	LLW burial ground
Long term river discharge	Susquehanna River	Susquehanna River
Short term river discharge	Susquehanna River	Susquehanna River
Liquids storage in tanks (the no-action alternative)	TMI site	TMI site

5.6.6.4. Health and environmental impacts from tritiated water disposal

At TMI, the evaporation of 8.7 million L began in December 1990 and was completed in August 1993. According to the Pennsylvania Department of Environmental Resources, the total activity during evaporation was 24.3×10^{12} Bq of tritium resulting in a 0.01–0.013 mSv dose to the public. The ranges of evaluated impacts are summarized in Table 3 for the nine alternatives listed in the Table 2.

5.7. SOLID WASTE

Nuclear accidents produce large volumes and varieties of very low to high activity wastes that are to be disposed of as solid waste. In general, much of this can be similar to that of non-accident nuclear waste, although the quantities are greater and the radionuclide mix and levels of contamination can be quite different. Normal waste sources include radiation protection disposables, decontamination materials, contaminated tools and equipment, process media and miscellaneous items. Accident wastes that can be very different from normal include trees, ion exchanger media used to absorb substantial amounts of fission product radionuclides such as caesium, filters for the same purpose, and a host of equipment used in the course of recovery and cleanup.

Waste management methods and technologies are also much the same in function; however, the more severe conditions often present special challenges. For this publication, three cases are described that illustrate post-accident unique situations. They are as follows:

(a) The need to establish a substantial capacity for interim on-site waste storage at Fukushima Daiichi as a result of the huge amount of contaminated items and materials;
(b) The initial and continued use of high integrity containers at TMI-2 to meet the challenge of disposal of process media without removing it from the process vessels;
(c) Planning, designing and building of waste management facilities at the Chernobyl NPP to accept a large amount of waste yet to be removed for final disposal.

TABLE 3. RANGES OF EVALUATED IMPACTS FOR EVAPORATION OF TMI-2 TRITIATED WATER [73]

Impacts	Range of impacts
Bone dose to the off-site population	0–14 person-rem total population (0–0.14 person-Sv) 0–0.4 mrem to the maximally exposed off-site individual (0–4 µSv)
Total body dose to the off-site population	0–3 person-rem total population (0–0.03 person-Sv) 0–5 mrem to the maximally exposed off-site individual (0–50 µSv)
Thyroid dose to the off-site population	Up to 6 person-rem total population (0–0.06 person-Sv) Up to 4 mrem to the maximally exposed off-site individual (0–4 µSv)
Estimated number of radiation caused cancer fatalities to the off-site population	0–0.0004
Estimated number of radiation caused genetic disorders to the off-site population	0–0.002
Occupational dose	0–25 person-rem (0–0.25 Sv)
Estimated number of radiation caused cancer fatalities to the worker population	0–0.003
Land commitment	0–49 000 square feet (0–4552 m^2)
Radioactive waste burial ground volume	0–460 000 cubic feet (0–13 026 m^3)
Cost to the licensee	US $100 thousand to US $41 million
Time to complete	0–36 months
Number of traffic accidents	0–12
Estimated number of traffic fatalities	0–0.8
Maximum individual dose from accidents	0–60 mrem total body (0–600 µSv) 0–3000 mrem bone (0–30 mSv)
Population dose from accidents	0–0.7 person-rem bone (0–70 mSv) 0–0.02 person-rem total body (0–0.2 mSv)

5.7.1. Interim on-site waste storage at Fukushima Daiichi NPP

Recovery from any accident will require interim on-site waste storage. The experience at Fukushima Daiichi is presented here as it represents a wide range of waste types needing such storage. An overall picture of the Fukushima Daiichi waste management challenge is in Ref. [9].

The different types of contaminated waste material that are managed within the Fukushima Daiichi on-site areas include trees, buildings and other debris, and very large amounts of contaminated water/secondary water treatment waste. The three waste types are managed independently. Debris and felled trees are segregated and stored in areas based on dose rates and contamination levels.

The initial on-site waste management strategy focused on providing safe, temporary storage for the wastes associated with stabilization and dose reduction efforts. Planned actions included:

(a) Construction of temporary storage facilities (with soil coverage of debris);
(b) Construction of soil covered temporary storage for cut down trees;
(c) Relocation of temporary storage facilities to reduce dose rates at the site boundary;
(d) Construction of a temporary cask storage facility to support removal of nuclear fuel;
(e) Construction of storage facilities for secondary waste from water treatment (e.g. absorption media).

It is conservatively estimated that a total of 560 000 m^3 of contaminated material will be generated until the end of fuel debris removal, planned for 2027. A new centralized storage facility is being planned with a capacity of approximately 160 000 m^3. The difference between the estimated amount of waste and the planned capacity of the storage facility highlights the expectation that waste segregation, volume reduction and recycling will reduce the volume of waste requiring long term management (storage) as radioactive waste.

Contaminated trees are a significant waste stream. As of early 2014, 79 300 m^3 of trees were being stored on-site. The trunks are managed separately from the branches, leaves and roots, as higher activity levels are present on the branches, leaves and roots than on the trunks. It is estimated that the volume distribution is roughly 30% trunks, 40% branches and leaves, and 30% roots. Further segregation of bark from the trunks could be effective at leaving minimal contamination on the trunks. The tree trunks are temporarily stored in stacks with limitations on the height and measures to ensure airflow in the stacks to reduce the fire hazard. Temperatures are also monitored to further protect against fire.

The branches, leaves and roots are placed in covered temporary storage facilities that include multiple barriers with retaining walls and soil along the sides, and soil and impermeable high-density polyethylene sheets above the waste for shielding and to control the infiltration of water. Ventilation and temperature monitoring reduce the possibility of fires. Figure 21 is a schematic for a covered storage facility for branches, leaves and roots.

Contaminated building parts and other debris must be stored on the site. Activities to stabilize and to reduce dose rates from the damaged reactors resulted in large amounts of debris being collected (see Fig. 22). Debris also continues to be produced as a result of ongoing recovery activities. Some of the debris has high levels of contamination and its management requires additional radiation protection measures for workers. These conditions create challenges for accessing, characterizing and removing debris. Debris has been transported by means of heavy equipment. The waste is transported to collection areas and then stored in different types of facility based on the measured dose rates. The storage types in use include storage buildings, soil covered temporary storage facilities, storage tents and outdoor storage covered by sheets to limit water infiltration (Fig. 23). For soil covered temporary storage, the soil and high-density polyethylene sheets provide shielding and reduce the amount of infiltration of water into the stored waste.

Debris is segregated based on the surface dose rates and type of material (e.g. concrete, steel). Owing to the implementation of planned measures such as volume reduction, recycling and incineration, the volume of stored waste could be significantly reduced. At the same time, the management of solid waste is complicated because in the absence of waste acceptance criteria derived from a disposal concept, only temporary stabilization and storage can be implemented [67].

Guidelines as shown in Table 4 were established to ensure that debris with surface dose rates greater than 1 mSv/h is stored in storage tents, soil covered temporary storage facilities and solid waste storage buildings. These guidelines were developed by the site operator (TEPCO) and approved by the regulator (Nuclear Regulation Authority of Japan) in view of worker protection and to support the maintenance of a dose rate of 1 mSv per year at the site boundary. Several thousand cubic metres of contaminated soil have been generated and are stored separately from other debris. Operational waste (e.g. HEPA filters) is managed similarly to debris.

The third type of waste that needs interim storage is the secondary waste generated by the several water processing systems. The two primary types of waste generated are sludge and used vessels from the caesium removal processes. The processes result in the accumulation of shielded steel vessels containing spent zeolite that has been used to capture caesium and other contaminants such as oil, strontium, technetium and iodine (see Fig. 24).

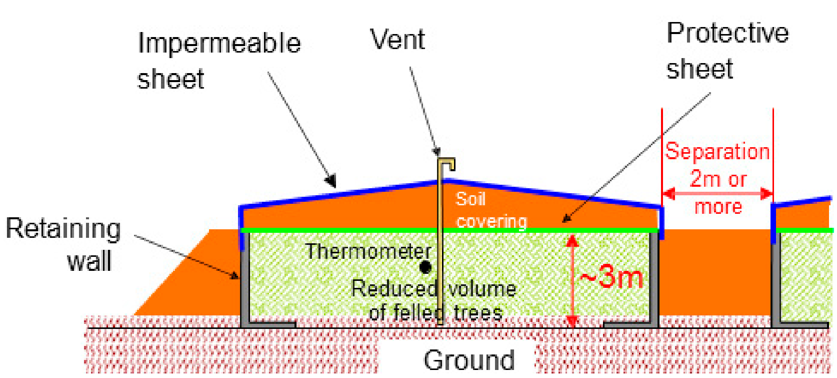


FIG. 21. Schematic of a soil covered storage facility for branches, leaves and roots. (Courtesy of TEPCO, Japan.)

In parallel with ongoing activities, research and development activities focusing on technical solutions for the treatment of salt containing wastewater are being undertaken. The objective is to produce waste forms that are suitable for long term storage on the site. Several techniques such as direct cementation, drying and storage, and drying and subsequent cementation have been investigated. This includes various practical tests with the aim of increasing the salt content in the cement matrix as much as possible, while still meeting the required mechanical strength and homogeneity.

FIG. 22. Whole view of Unit 3 reactor building north side (before large debris removal). (Courtesy of TEPCO, Japan.)

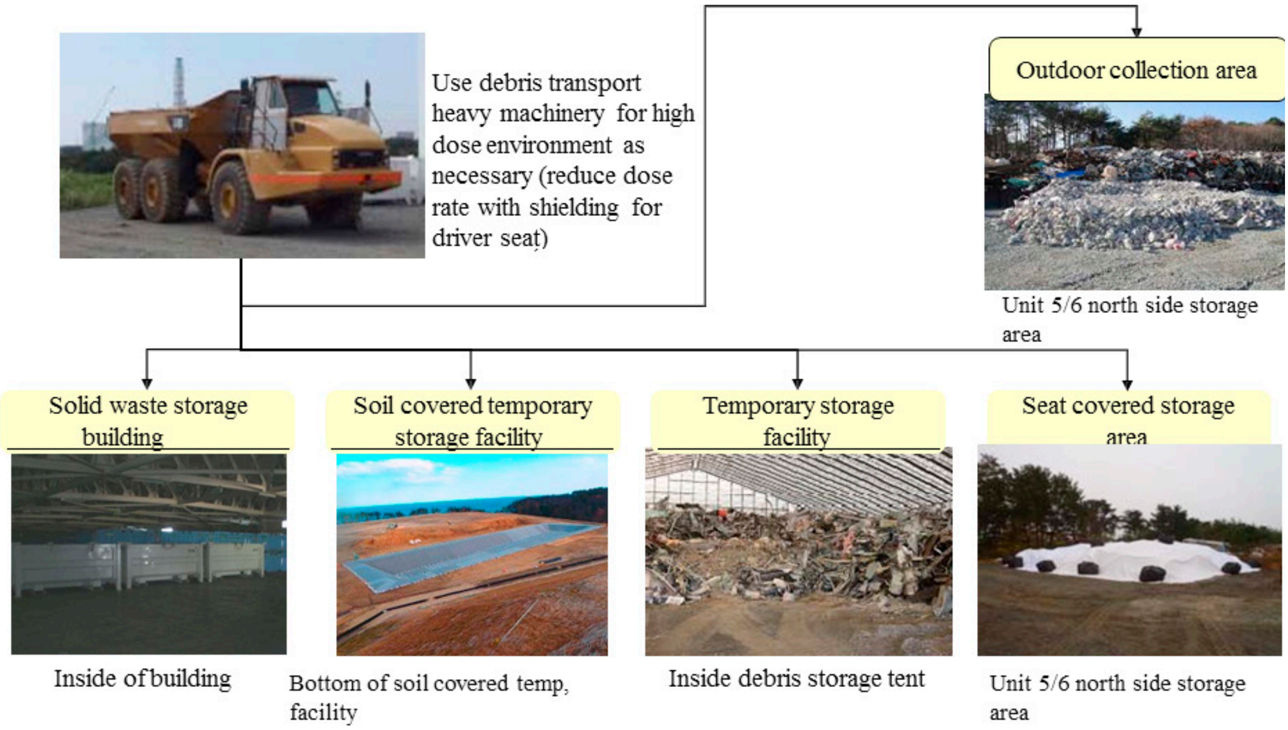


FIG. 23. Examples of facilities constructed for the management and storage of debris. (Courtesy of TEPCO, Japan.)

TABLE 4. GUIDELINES FOR SEGREGATION AND STORAGE OF DEBRIS

		Surface dose rate of debris (guide value) *			
		0.1 mSv/h or less	0.1–1 mSv/h	1–30 mSv/h	More than 30 mSv/h**
Storage approach	Shielding	None	None	Concrete wall, soil, containers	Containers and building
	Prevent dispersion	None	Sheet cover	Tent, soil, containers	Containers
Temporary storage method		Outdoor collection	Sheet cover	Containers storage Temporary storage facility Soil covered temporary storage facility	Container storage Solid waste storage building

* Review guide value as appropriate considering on-site air dose rate.

** For those over 1 Sv/h, temporarily store in containers, solid waste storage building or shielded area.

FIG. 24. SARRY vessels in the operating facilities. (Courtesy of TEPCO, Japan.)

5.7.2. High integrity containers initiated at TMI-2

The concept of high integrity containers (HICs) was introduced around the same time as, but not as a result of, the TMI-2 accident. It was considered by the NRC as an acceptable alternative to solidifying waste. These containers were called high integrity because, compared with very LLW containers, greater strength and leak tightness would be needed for the types of waste for which they would be used. They would need to be equivalent to encapsulation. This concept was adopted at TMI-2 to eliminate the risks of high personnel exposure and spills of the very high radioactive materials that could result from solidification activities. In addition, there were high uncertainties associated with demonstrating that the solid mass in the disposal container was uniformly mixed. The use of HICs subsequently took on a variety of designs and materials suited to the waste characteristics combined with handling safety and the economics of fabrication.

HICs have been used at both TMI-2 and Fukushima Daiichi for two significant forms of abnormal waste resulting from processing high activity accident generated water. They are called abnormal because of the high concentration of fission products and/or small quantities of fuel materials. One form consists of resin media within demineralizer vessels and the second is zeolite based systems used primarily to capture ^{137}Cs, also within vessels.

At TMI-2, the need to dispose of these waste forms was significant in that it paved the way for the use of HICs. At Fukushima Daiichi, HICs are used for interim storage after the process media (resins or zeolites) have been chemically exhausted. At TMI-2, they served to meet disposal site waste acceptance criteria. This is one of the factors that led to the standard use of HICs [74].

The major benefits of HICs include the following:

(a) Eliminate the personnel exposure intensive job of sluicing the extremely radioactive resins for conditioning (e.g. during the TMI-2 accident, highly contaminated coolant water passed through the tanks, whose resins captured roughly over 400 TBq of radioactive caesium [75]). At TMI-2, the initial options for disposal of these resin media vessels focused on solidifying with cement. However, conceptual designs for cementation indicated that a facility for doing so would be very expensive, dose intensive for workers, and in itself would create another waste stream.
(b) Avoid the technical complications of process vessels with a sacrificial mixer for in situ solidification. This was also rejected at TMI-2 primarily for the inability to demonstrate that the resulting form would be homogeneous within the vessel. In addition, there was concern that if any of the vessels would need to be opened to recondition the material, it would be dose intensive and would be challenging to prevent spread of contamination.

Therefore, for TMI-2 resin vessels that could not meet the criteria for LLW disposal, a special HIC was designed that consisted of a reinforced concrete cylindrical container (Fig. 25). Leakage was prevented by a corrosion resistant steel liner that was coated inside and out with phenolic paint. The durability of the HIC was enhanced by the amphoteric material placed on the inside bottom of the container for pH control. After loading, the HIC lid was sealed and bonded to the body using a bead of adhesive gel and flowable grout material. A vent system allowed gas produced by radiolysis to escape. Without venting, the HIC had sufficient burst strength to contain the gas that may be generated within a 300-year lifetime. The concrete container attenuated radiation from the enclosed media by a factor of approximately nine, which was not enough shielding for manual handling but was sufficient to simplify handling procedures.

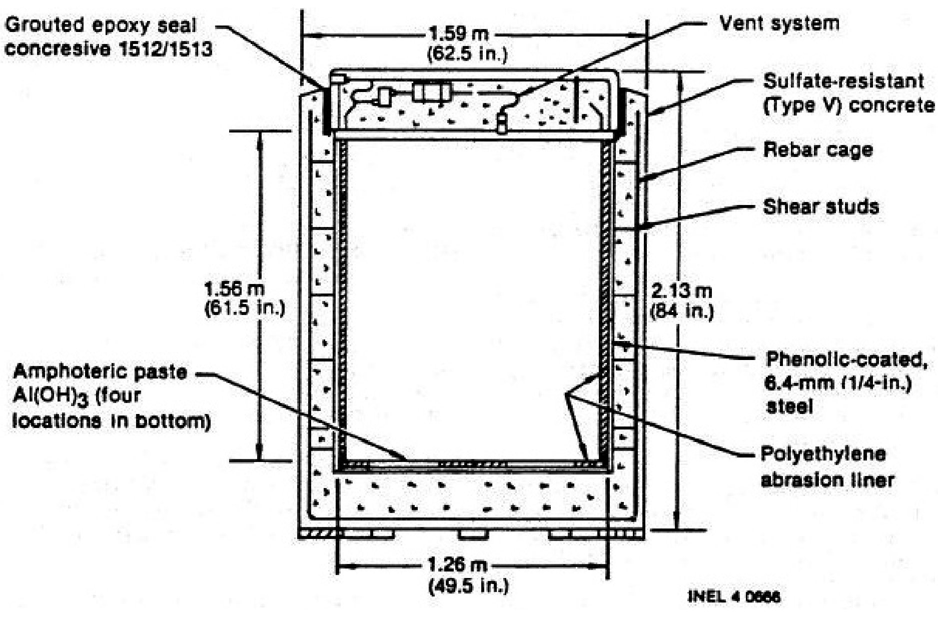


FIG. 25. Reinforced concrete high integrity container. (Courtesy of EPRI, USA.)

The radioactivity of nineteen of the zeolite vessels at TMI-2 was also greater than the criteria for disposal at a commercial LLW repository. Sixteen were buried at USDOE's Hanford disposal site, each within a custom designed overpack. The other three were used for experimental demonstration of vitrifying the contents. This was done within a hot cell. The remaining zeolite vessel activity was sufficiently low for disposal at a commercial site. These were disposed of within ferralium HICs at the Barnwell site.

5.7.3. Final disposal of waste at Chernobyl NPP

Chernobyl NPP plans and activities include a detailed approach to long term management for disposal of radioactive waste. The discussion below is taken from a 2011 report [76] (footnotes omitted) that describes the past situation and that addresses the path forward for management of these wastes [77]. (In the quoted text that follows, ChNPP stands for Chernobyl nuclear power plant.)

> "…the great bulk of the radioactive waste in Ukraine resulted from the ChNPP accident of 1986 and accumulated in the ChEZ. The ChEZ Radioactive Waste Disposal Storage (RWDS) and Radioactive Waste Interim Localization Sites (RWILS) contain 1,928,107 m^3 of radioactive waste with a total activity of 7,260 TBq (Table 3). In addition, the storage facilities of the ChNPP contain 519 TBq (22,117 m3) of solid and liquid radioactive waste resulting from the operation of the ChNPP reactors.... In addition, large quantities of radioactive waste are held in the Shelter Facility that, according to its official status, can be functionally defined as a location for surface disposal of "unorganized" radioactive waste (an unorganized radioactive waste storage facility undergoing the phase of stabilization and refurbishment) (MHU 1997). The volume of radioactive waste in the Shelter Facility is estimated to be 385,200 m^3, including 1,250 tons of lava-like fuel masses; 50 tons of dust containing over 1% of the nuclear fuel; 22,240 tons of metal structures with surface contamination equal to the equivalent dose rate of over 10 mSv h^{-1}; and 3,000–5,500 m^3 of contaminated aqueous masses identified as liquid radioactive waste. The volume of radioactive waste in the soils of the Shelter Facility is estimated to be 277,300 m^3" [76].
>
> Examples of debris and material within Unit 4 are shown in Fig. 26.

Three RWDS facilities were built shortly after the accident to manage the radioactive waste resulting from decontamination of all units and industrial areas. They are as follows:

(a) **Kompleksny RWDS:** "was created in October 1986 using unfinished structures of the radioactive waste storage facility that was part of ChNPP Reactor Unit 5, which was being built at that time…. The waste associated with the contaminated soil, metal structures, chunks of the roofing materials, and concrete from the ChNPP Reactor Unit 4 was placed into metal 1.5-m^3 containers, and those metal containers were installed into waterproof reinforced concrete tanks [see Fig. 27]. The Kompleksny RWDS was shut down in 1988. The total volume of the waste disposed at the site is 26,200 m^3 with estimated activity of 74 TBq (as of in 2000). The Kompleksny RWDS was decommissioned by filling it with sand and covering it with a 1-m-thick clay layer topped with turf grass" [76].

(b) **Pidlisny RWDS:** "This was specifically built in December of 1986 for disposal of the medium level and high level waste with the exposure dose rate ranging from 50 to 500 mSv h^{-1} resulting from the decontamination of the ChNPP Reactor Units 3 and 4. The Podlesny RWDS is a concrete structure placed on a concrete 1.5-m-thick foundation. The walls of the vaults of this disposal site are 8–9 m high reinforced with a 4–5-m-thick soil layer from the outside [see Fig. 28]. Only two vaults, Vault A-1 and Vault B-1, were actually used, but a total of eight vaults were constructed. Vault B-1 received waste placed in 1.5 m3 metal containers, and Vault A-1 received dumped non-containerized waste. The waste had not been preliminarily sorted. It contained chunks of the reactor graphite, fragments of reactor structures and other metal structures, chunks of fuel assemblies, and decontamination waste from ChNNP Reactor Units 3 and 4. The total volume of the waste disposed of at the Podlesny RWDS is estimated to be 3,960 m^3 with activity of 2.59 PBq (MESU 2010). In November of 1988, the operation of the Podlesny RWDS was stopped, and in 1990 it was decommissioned. The waste vaults were filled with 2,300–2,400 m^3 of concrete and approximately 1,300–1,500 m^3 of sand and gravel mixture" [76].

FIG. 26. Debris and material within Unit 4. (Courtesy of Chernobyl NPP, Ukraine.)

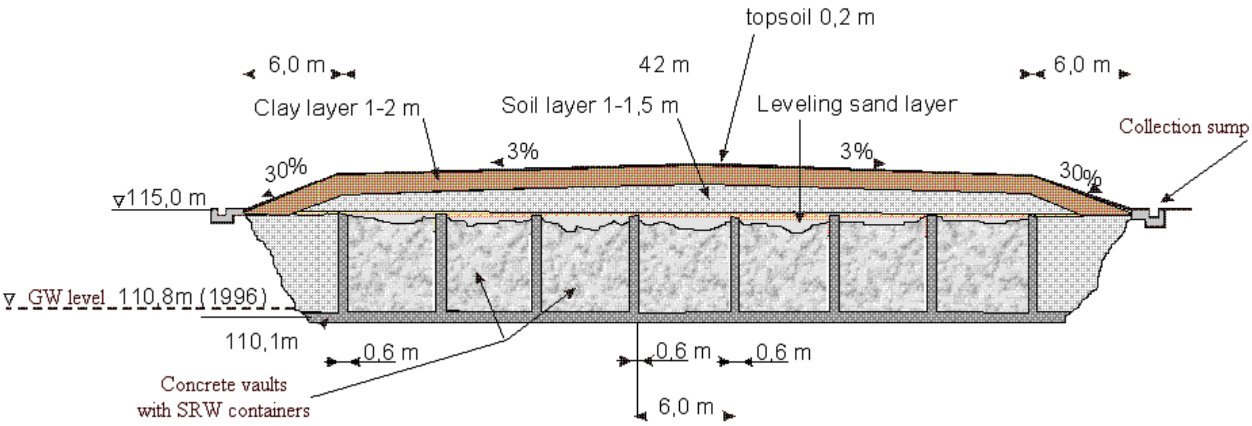


FIG. 27. Cross-section of the Kompleksny RWDS [76]. (Courtesy of Chernobyl NPP, Ukraine.)

FIG. 28. Pidlisny RWDS. (Courtesy of Chernobyl NPP, Ukraine.)

(c) **Buryakovka RWDS:** “is a subsurface disposal site located 10 km west of the city of Pripyat. It was commissioned in 1987, and it has been in operation since that time. The Buryakovka RWDS includes 30 trenches insulated with a 1-m-thick clay layer. Twenty-five trenches were completely filled with radioactive waste and decommissioned by filling them with sand topped with a 1-m-thick clay layer and covered with turf grass…. The waste at the Buryakovka RWDS is associated with contaminated machinery, metal and reinforced concrete structures, protective clothing, debris, etc. Prior to the 1990s, the criterion for waste acceptance into the Buryakovka RWDS was that the external equivalent exposure dose not exceed 50 mSv h^{-1}. On 1 January 1990, the acceptance criteria became more rigid: The maximum allowed exposure dose rate was decreased from 0.3 μSv h^{-1} to 10 mSv h^{-1}, and the content of alpha-emitting radionuclides was no longer allowed to exceed 2% of the total activity. Only waste of Chernobyl origin was accepted for disposal” [76].

A radioactive waste management strategy for Ukraine was adopted in 2009. The objective of the strategy is to complete development and ensure an effective functioning of the comprehensive radioactive waste management system. The implementation of the strategy is planned for 50 years.

Construction of the radioactive waste management complex Vektor is in progress in the Chernobyl Exclusion Zone. Design and construction of a large scale solid radioactive waste processing plant (Industrial Complex for Solid Radioactive Waste Management) were completed at the Chernobyl NPP. The plant includes several solid radioactive waste facilities combined in a single cycle supported by the production of metal barrels and reinforced concrete containers [76, 77]. Figures 29 and 30 show two of these facilities.

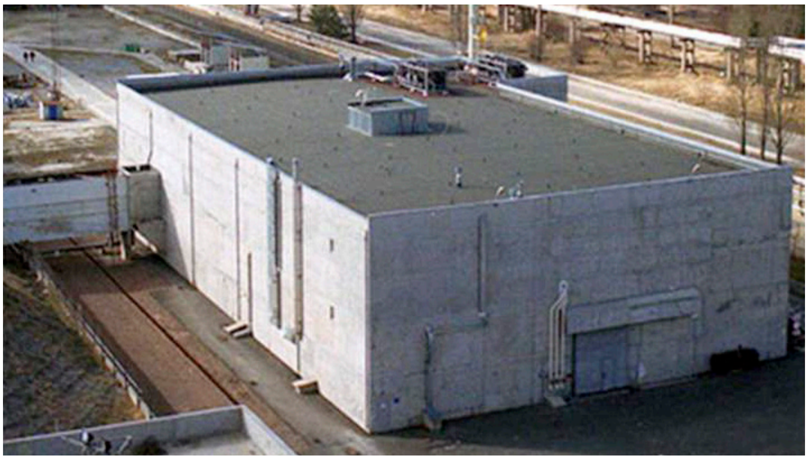


FIG. 29. Plant for liquid waste processing. (Courtesy of Chernobyl NPP, Ukraine.)

FIG. 30. Production of metal barrels and reinforced concrete containers. (Courtesy of Chernobyl NPP, Ukraine.)

The Vektor Complex site is intended for receipt, processing and/or disposal of the solid radioactive waste accumulated during the operation of the Chernobyl NPP, and the waste resulting from its decommissioning, as well as the radioactive waste associated with operation of the shelter. The Industrial Complex for Solid Radioactive Waste Management site provides for the following [76, 77]:

“• Interim storage of low-, medium-, and high-level waste (in waste packages with a service life of 30 years, followed by a period of 300 years of monitoring by the national authority after its closure) designed and built in the existing liquid and solid radioactive waste storage building at the ChNPP industrial site;
• System for retrieval of solid radioactive waste of all categories from the existing solid radioactive waste storage facility located in the solid radioactive waste storage facility building at the ChNPP site;
• Plant for sorting solid radioactive waste of all categories and conditioning low- and medium-level solid waste with total throughput of 20 m^3 of unprocessed waste per day, including the incineration system for processing solid radioactive waste at 50 kg h^{-1} and liquid radioactive waste at 10 kg h^{-1}; the grouting facility at 10 m^3 d^{-1}, the low- and medium-level legacy waste and high-level waste packaging facility at 1.5 m^3 d^{-1}; and the low and medium-level legacy waste and high-level waste interim storage facility with a capacity of 3,500 m^3 and a service life of 30 y; and
“• A specially equipped subsurface low- and medium-level short-lived solid radioactive waste storage facility with a capacity of 55,000 m^3 of waste packages and a service life of 30 y for filling up the facility followed by 300 y of national monitoring after its closure; the facility is located within the Vektor Complex” [76].

A review on the feasibility study for location and creation of new facilities for managing the contaminated material at the Chernobyl NPP was done in July 2018 as the existing infrastructure is not sufficient to manage the accumulated and foreseen waste. New facilities are therefore required to improve the existing radioactive material management system.

6. DAMAGED FUEL AND FUEL DEBRIS MANAGEMENT

6.1. INTRODUCTION

Damaged fuel and fuel debris management is the major post-accident cleanup challenge due to several factors including, but not limited to the following:

(a) Uncertainty of the location, configuration and physical characteristics of the damaged nuclear fuel and debris containing fuel materials;
(b) Loss of structural integrity of the fuel/core support structure and other vessel internals;
(c) High radiation levels from the fuel as well as fission products and corrosion products in the surroundings, and in the water and atmosphere where the materials are located;
(d) Difficulty of access to the fuel bearing materials, whether they are in air or water, in addition to high radiation levels, poor visibility and long reach distances from work areas to the furthest location of fuel material to be removed [7].

Post-accident situations present very different circumstances for fuel removal when compared to shutdown after normal operations. Fuel damage can be limited or extensive ranging from minor cladding breaches through to total core destruction. The extent of fuel and fission product release can vary markedly as has been experienced from past accidents. Fuel, FCMs, corium, hot particles and other nuclear/radioactive components may be mixed with core metals and other materials that can result in a variety of physical debris and resolidified masses. These will affect local dose rates, require very specialized handling, characterization, packaging, storage and many other activities, leading to the ultimate stabilization (disposal) of the nuclear materials.

Prior to a final resolution of its disposal, managing damaged fuel and fuel debris can be viewed with interim and final goals, including the following:

— An on-site interim stage in which the materials are captured in a safe, confined condition such that subcriticality is ensured, inadvertent release cannot occur and other safety case requirements for storage are maintained;
— A stable condition for the long term where the materials are placed in long term safe storage at an on-site or off-site location or possibly reprocessed into a safe form for disposal.

Achieving these goals requires characterization, which is essential for detailed planning before damaged fuel and fuel debris can be removed and stored (interim and long term) [7]. Ultimately, a final disposal method will need to be implemented.

6.2. ACCIDENTS RESULTING IN DAMAGE TO NUCLEAR FUEL

Several reactor accidents in which the core has been damaged are used as examples in this section. Effective fuel removal and transport off-site was accomplished at A1 NPP and TMI-2. Windscale Pile 1's fuel debris is to remain in situ until the entire facility is decommissioned. The removal of damaged fuel from Chernobyl NPP has been postponed until after the erection of the NSC. There is no information yet on fuel removal for the Fukushima Daiichi NPP. At this time, perhaps the most comprehensively documented approach addressing fuel removal issues of severely damaged fuel is the TMI-2 case, which has been reported on extensively, including in IAEA reports [1–5, 7, 18, 27]. Table 5 is a chronological tabulation of fuel damage events published and updated to include Paks and Fukushima Daiichi [7]. A few earlier events have been included for completeness.

6.3. PHYSICAL SITUATIONS OF DAMAGED FUEL AND FUEL DEBRIS

In this section, five brief damage fuel related descriptions for past accidents are presented, followed by a discussion of key challenges for damaged fuel management.

6.3.1. Windscale Pile 1

> "The nuclear properties of the various materials used to construct the piles were not fully understood at the time of design, in particular, the need to accommodate the growth in the graphite caused by neutron irradiation (Wigner growth) at the low irradiation temperatures encountered (20–153°C mean graphite temperature). Accordingly, the graphite core was designed with small gaps between the core blocks, back to front and side to side, whose size varied according to core position. The Wigner effect not only caused anisotropic growth in the graphite blocks, but also manifested itself as stored energy within the graphite crystal structure. Hence, one of the operational tasks was to limit both the growth and the stored energy by using nuclear heating to anneal the core. During the working life of both piles, nuclear heating with reduced air cooling was used to elevate the core temperature to a point where temperature excursions occurred caused by the release of stored Wigner energy. It was during one such incident in October 1957 that an uncontrollable temperature rise was experienced, which led to a fire in the reactor core. The fire was eventually extinguished by a combination of water pumped into the core and closing down the cooling airflow. The precise cause of the fire is still a source of conjecture" [36].

After the fire, a full range of damaged fuel scenarios existed — from minor cladding breaches through to fully oxidized fuel residues [78]. This range has created problems for retrieval resulting from the application of water to quench the core fire. It is conjectured that minor cladding breaches may have introduced water into the aluminium clad fuel rods, thus increasing the potential to create pyrophoric uranium hydride — a hazard for fuel removal. At the other end of the scale, the presence of uranium bearing oxidation products/dusts created a problem for retrieval and will require specially designed tooling. Additionally, the presence of isotope cartridges within the core maintains subcriticality so that care must be taken during the sequencing of fuel and isotope retrieval operations from the fire affected zone (FAZ). The accident damaged nature of Windscale Pile 1 has led to some unique technical challenges relating to criticality, the possible pyrophoricity of materials that could have been produced during and post-accident and the potential for dust explosions.

TABLE 5. FUEL DAMAGE EVENTS ARRANGED CHRONOLOGICALLY [7]

Plant (year)	Country	Primary cause	Description
National Research Experimental (1952) water cooled, heavy water moderated	Canada	Design, operator error	A reactor runaway from a combination of design flaws and operator error resulted in damage of fuel and leakage of moderator water, flooding the building — returned to service.
Windscale (1957) gas cooled graphite pile	UK	Lack of information for operators	Uncontrolled release of Wigner energy, fire and destruction of a substantial portion of air cooled core, some fission products released to the environment.
SL-1 (1961) small prototype pressurized water reactor	USA	Design	Prompt critical while shutdown with head off, reconnecting control rod-to-drive mechanism, destruction of the core, substantially contained within the building.
Chapelcross (1967) Magnox, carbon dioxide cooled, graphite moderated	UK	Design or operation	Graphite debris partially blocked a fuel channel causing a fuel element to melt and catch fire. Contamination was confined to the reactor core. The core was repaired and restarted.
Fermi 1 (1968) sodium cooled	USA	Design	‘Splitter plates’ below the core vibrated loose and blocked fuel channels, causing melting of several assemblies, contained within primary system — returned to service.
Ågesta (1968) water cooled	Sweden	Design	Spacer grid spring relaxation and flow vibration. Fifteen fuel assemblies failed. Returned to service with modified fuel.
St. Laurent (1968) gas cooled, graphite moderated	France	Procedure	Flow reducer for a control channel placed in a fuel channel. Fuel overheating and destruction of five cartridges — returned to service.
Lucens (1969) experimental gas cooled, heavy water moderated	Switzerland	Channel flow blockage	Coolant leakage, followed by moderator tank rupture, and severe damage to a single fuel assembly.
Jaslovské Bohunice, A1 (1977) gas cooled, heavy water moderated	Slovakia	Operator error, blocked fuel channel	Rupture of fuel cladding and de-cladding of fuel occurred in the upper 30–100 cm of fuel elements; 148 assemblies affected. Reactor building contaminated.
Three Mile Island (1979), pressurized water reactor	USA	Design and operator error, relief valve stuck open	Failed to keep core covered with water, destruction of the core with large fraction melted, fuel contained within systems; fission product contamination to the containment building.
Chernobyl (1986), water cooled, graphite moderated	Ukraine	Design and violation of operating procedures	Prompt critical reaction caused destruction of the reactor with substantial distribution of fuel and fission products outside the primary envelope and to the environment.
Paks (2003), pressurized water reactor	Hungary	Design, operational delay	Fuel rod damage in cleaning tank.
Fukushima Daiichi (2011), three boiling water reactors	Japan	Tsunami, design	Severe damage to the reactor cores in three of the four units.

“[The] desk top studies of the various issues relating to criticality, the possibility of uranium hydride being present and the possibility of graphite dust explosions have demonstrated successfully that such concerns have been largely unfounded. This conclusion has enabled the detailed planning for decommissioning work on pile 1 to be progressed from an initial round of non-intrusive visual inspection into the undamaged sections of pile 1 core through to intrusive inspection of the fire damaged core. The results of this work have underpinned the methodologies for removal of the remaining fuel and isotopes in the core of pile 1, because these constitute a major hazard” [36].

A fuel channel removal tool has been developed and deployed for inactive trials using a mock-up of the charge face of Pile 1 [57, 79]. The Windscale Pile 1 site is now essentially amalgamated into the larger adjacent Sellafield site under the control of the United Kingdom Nuclear Decommissioning Authority. Pile 1 is under care and maintenance while decommissioning works are carried out on the rest of the facility.

6.3.2. A1 NPP

The gas cooled heavy water moderated reactor A1 NPP was in operation at the Jaslovské Bohunice nuclear site in Slovakia from December 1972 to February 1977. Two separate accidents occurred that resulted in fuel damage.

The first accident happened in January 1976. One of the fuel assemblies was not properly installed in the active core. The pressurized CO_2 coolant, with a circulation path from the bottom of the reactor, ejected the fuel assembly into the reactor hall, resulting in its significant damage (Fig. 31). However, the consequences of the accident were such that operation of the reactor could continue after repairs were made.

The second accident occurred in February 1977. An operator failed to remove humidity absorbers inside the fuel assembly prior to placement in the reactor. The absorbers blocked the cooling of a fuel channel, resulting in the overheating and rupture of fuel cladding.

The overheating caused damage to an internal heavy water tank within the vessel and leaking of heavy water into primary containment. This affected the cladding of most of the 148 fuel assemblies in the core. The overheated and melted assembly is shown in Fig. 32. The primary circuit was heavily contaminated and, due to leakages from the steam generators, the secondary circuit and the cooling towers became contaminated. As a result, the decision was made to remove all fuel assemblies from the reactor and to place them into the long term storage pool (Fig. 33). Standard refuelling technology was used for that operation [80, 81]. Additional description of the A1 NPP damaged fuel and radioactive waste management is given in a separate annex of this publication (see Annex III).

FIG. 31. Damaged fuel assembly on the floor of reactor hall. (Courtesy of JAVYS, a.s., Slovakia.)

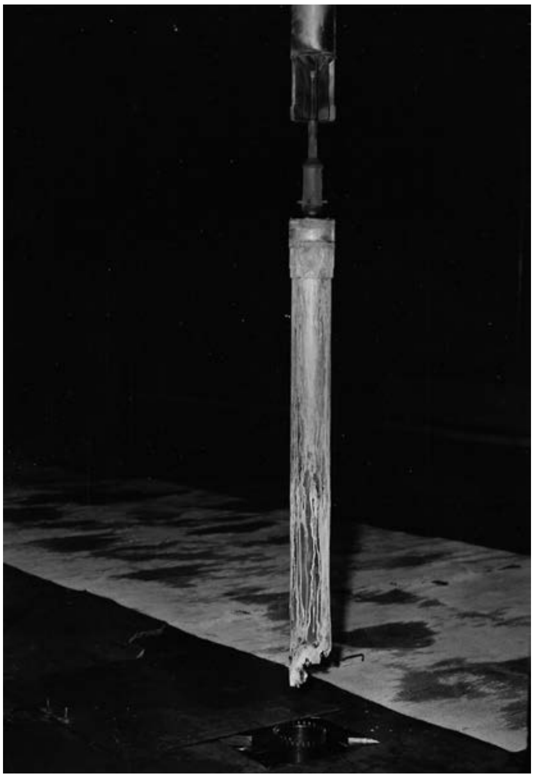


FIG. 32. Part of the overheated fuel assembly after its removal from the reactor. (Courtesy of JAVYS, a.s., Slovakia.)

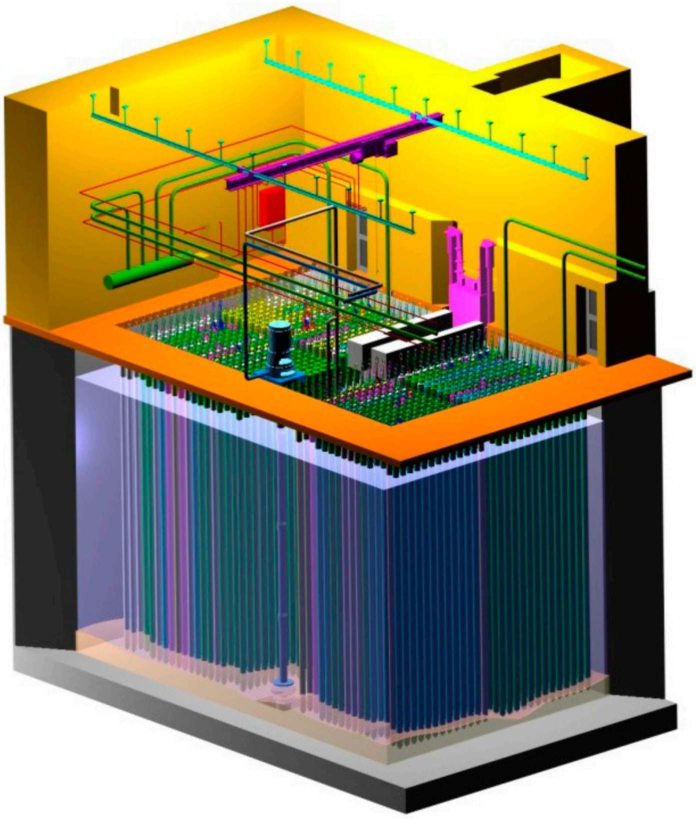


FIG. 33. A 3-D model of the A1 NPP long term storage pool. (Courtesy of AllDeco, s.r.o., Slovakia.)

6.3.3. Three Mile Island Unit 2

In 1979 at TMI-2, a reactor shutdown and stuck open pressure relief valve led to a major loss of coolant event and nuclear core overheating that destroyed most of the fuel, including melting and resolidification of a significant portion. Figures 34 and 35 show the overall core damage and an underwater photograph of damaged fuel assemblies within the core.

Removal of the fuel material began in the sixth year after the accident and continued for over four more years. Fuel and debris removal involved the use of a mining rock drill with special bits to reduce the molten mass to rubble, after which removal was manual retrieval within the water filled reactor vessel at depths of 10–20 m. The long handled removal tools were fabricated on the site as needed [82, 83].

TMI-2 is currently in care and maintenance mode with annual inspections within the containment. There are approximately 1000 kg of fuel materials remaining that cannot be retrieved without cutting open the system. This fuel is in particulate form distributed throughout the reactor primary system. The intent is to dismantle Unit 2 following Unit 1, which is planned to occur in the 2030s.

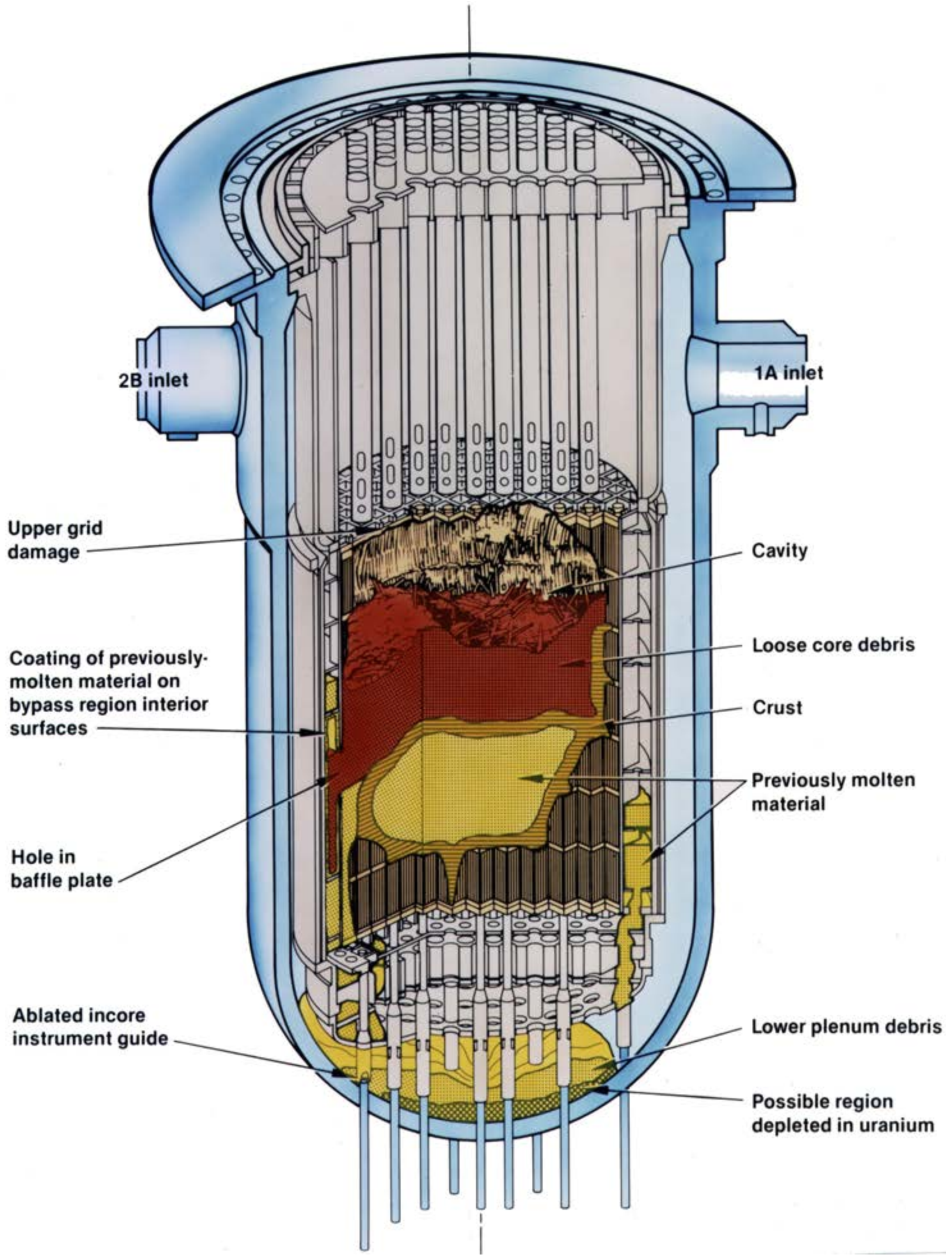


FIG. 34. TMI-2 core damage. (Adapted from Ref. [7].)

FIG. 35. Example of TMI-2 fuel damage. (Adapted from Ref. [7].)

6.3.4. Chernobyl NPP

At the Chernobyl NPP, the explosion in 1986 caused by the rapid power transient during the accident displaced the reactor charge face and ejected fuel over the surrounding area. Core meltdown rapidly ensued leaving lava-like masses of corium containing fuel, cladding and other core materials in the base of the reactor containment. Figure 36 shows the damage to the core and reactor internals from the steam explosion. While the fuel masses will remain within the damaged Unit 4 at Chernobyl NPP for the foreseeable future, early attempts to remove samples for analysis were made from the solidified lava mass known as the elephant's foot shown in Fig. 37.

Off-site analyses [84] of the Chernobyl NPP 'lava' and hot particles have now determined that high temperature (at least 2600°C) interaction between the nuclear fuel and zircaloy cladding took place in the local part of the Chernobyl reactor core before the explosion and that active chemical alteration of Chernobyl NPP lava continues.

Current plans for Chernobyl NPP Units 1–3 include a final shutdown and preservation stage through 2028, safe enclosure through 2045 and final dismantling through 2064. Investigations have indicated that the durability of some of the fuel masses has deteriorated due to the chemical interactions of the various components. This means that future plans for removal will require a thorough understanding of the fuel condition at the time of removal.

6.3.5. Fukushima Daiichi NPP

In 2011, an earthquake and a beyond design basis tsunami resulted in a complete loss of power, and the consequent meltdown of three reactors at Fukushima Daiichi. The extent of fuel damage and challenges related to its removal is much more complex than that experienced at TMI-2 due to the severity and duration of the accident and the different reactor type. The boiling water reactor vessels at Fukushima Daiichi were breached, releasing materials within the PCVs. In addition, the considerably greater amount and more complex design of metal structures of a boiling water reactor within, above and below the reactor vessels add complications compared with a pressurized water reactor.

FIG. 36. Chernobyl NPP core damage. (Courtesy of Chernobyl NPP, Ukraine.)

FIG. 37. Chernobyl NPP elephant's foot. (Courtesy of Chernobyl NPP, Ukraine.)

The overall working distances from the reactor vessel flange are some ten metres greater than in the TMI-2 case, which makes manual deployment of decommissioning tools infeasible. In the TMI-2 case, the working distances were considered to be on the limit for manually deployed tooling.

Some degree of remote technology will be deployed at Fukushima Daiichi. Planning for Fukushima Daiichi first requires removal of spent fuel assemblies from within the spent fuel pools for each unit. In the boiling water reactor design, the spent fuel pool is in the same building as the reactor containment, but it is not directly connected with the meltdown location during the accident.

The next steps involve fuel debris removal, which from the current Fukushima Daiichi long term plan have been tentatively indicated as follows:

(a) Decontamination of the reactor building using high pressure water jetting to improve accessibility to the PCV;
(b) Inspection of the lower parts of the PCV using remote closed-circuit television (CCTV) and other means;
(c) Stop leakage in the lower part of the PCV by constructing a boundary while the cooling flow for fuel debris is maintained;
(d) Water filling of the PCV, including changing the water source for cooling from the reactor building to the PCV;

(e) Inspection of the PCV for fuel debris distribution assessment and sampling;
(f) Repair of the upper part of the PCV for water filling up to the full level of the PCV for shielding purposes;
(g) Confirmation of shielding after the PCV water fill and removal of the PCV top lid;
(h) Installation of remote extending mast based machine from PCV operating floor, deploy in-core inspection/ sampling tools;
(i) Removal of fuel debris from reactor pressure vessel and PCV.

The latter steps will need to account for the very large radiation fields, and the presence of turbidity in the water that impedes visibility and compromises illumination levels.

6.4. MAINTAINING AND MONITORING SUBCRITICALITY

To maintain safe and stable conditions, it is essential to prevent any large scale energy generation from a fission reaction of the nuclear fuel in its post-accident configuration. Nuclear criticality[4] must not be allowed to occur. Great care needs to be taken to maintain control during all stages of fuel removal until it can be proven that disturbance of any remaining nuclear materials or components will not lead to criticality. During the fuel removal operations, neutron poisons/absorbers must remain present. Computer modelling can be used to support such operations. Good working practice ensures that K_{eff} continuously remains at 0.95 or below.

If the damage is limited, it may be possible to use the normal neutron activity measurement and reactivity shutdown systems designed for emergency responses and monitoring following stabilization. When the damage to the core and fuel is severe, and these instrument and system functions are lost, it is necessary to initiate fission monitoring and criticality prevention by other means as soon as possible [7]. Several methods have been used to address criticality, analysis, control and monitoring following fuel damaging accidents. The following descriptions of four of the most severe cases point these out.

6.4.1. Windscale Pile 1

The evaluation of margins to criticality, in the case of the reactor at Windscale Pile 1 following the 1957 accident, combined a variety of calculations together with inspections and measurements. The inventories of fuel and moderator in the FAZ in Windscale Pile 1 exceeded the minimum values required for criticality in an idealized maximum reactivity lattice. Past theoretical criticality assessments of Pile 1 were unrealistically pessimistic, and indicated the subcriticality margin to be small [6, 78]. Models were set-up using the reactivity modelling code called MONK [85], based on the estimated remaining fuel and isotope content assuming an intact graphite core structure. The theoretical modelling work was limited by a lack of detailed knowledge of the remaining core contents and configuration. On the basis of non-intrusive inspection exercises carried out over many years on the charge and discharge faces of the pile, an approximation was made to account for the remaining quantities of intact fuel and isotope cartridges, partially oxidized (burnt) cartridges and residual fuel bearing dusts. It later became possible to carry out core reactivity direct measurements [86], which resulted in a conclusion that there was likely to be a substantial subcriticality margin.

Updated modelling for reasons of evaluating credible seismic disturbance scenarios of the moderator, fuel and neutron absorbers showed conclusively that the measured margin of subcriticality would not be significantly changed during such events. The criticality safety assessment and the associated sensitivity studies demonstrated that Pile 1 would remain subcritical during current quiescent conditions and a seismic disturbance of the FAZ could not be expected to cause criticality [6, 78].

For purposes of understanding the margin to criticality for future core unloading, assessments were conducted using theoretical modelling and experimental determinations of the neutron multiplication factor (K_{eff}). It was demonstrated that the principal factors affecting core reactivity were not only the fissile content of the fuel, but also lithium containing isotope cartridges that still remained. These cartridges are very effective in suppressing the core reactivity due to thermal neutron absorption in ^{6}Li. The reactivity assessments demonstrated that the reactor could

[4] Criticality is "The state of a nuclear chain reacting medium when the chain reaction is just self-sustaining (or *critical*), i.e. when the *reactivity* is zero" [17].

be up to 3% supercritical if these isotope cartridges were removed before the fuel. These assessments indicated the need for a well planned method for reactor defueling and isotope removal such that the core reactivity could be progressively lowered until a status of 'criticality impossible' could be attained. That is, irrespective of the sequence adopted for fuel and isotope removal, no criticality became possible. It could be necessary to utilize neutron poisons (e.g. using boron acid solutions) during the early stages of fuel removal to improve the safety margin [6, 78].

6.4.2. Three Mile Island Unit 2

The severity of core damage combined with lack of knowledge of the actual core configuration at TMI-2 led to a management conclusion that control rods could not be relied on in any way for reactivity control and assurance of shutdown [7]. Further, there was no way of measuring the criticality margin. To ensure that the core could not become critical; a high boron concentration was maintained in the coolant.

Later knowledge showed that the control material was essentially gone from the core region as a result of melting. Thus, it was further judged imprudent to rely on analyses or models when the method assuring shutdown with boron was available.

The normal boron concentration in the TMI-2 reactor coolant system was 1000–1500 ppm. Just before the accident, a routine sample of the reactor coolant contained 1026 ppm boron. A sample taken shortly after the reactor shutdown indicated a boron concentration of only 700 ppm. This caused concern, especially when, two hours later, another sample showed a boron concentration of approximately 400 ppm. At the time, the operators believed that this was evidence of boron dilution resulting from the accident. In fact, it was because of reflux boiling in the core caused by low pressure and high temperatures. Much of the water in the coolant sample, which was taken from the loops, was condensate that contained no boron; because of boiling, there was a higher than normal boron concentration in the core region.

Immediate steps were taken to raise the boron concentration in the reactor coolant system because of the low concentration sample results and the higher than normal neutron flux readings from the source range monitors. As a result of these efforts, boron concentration increased to 1750 ppm. Because of the uncertainty regarding the extent of damage to the core, the boron concentration was then raised to over 3000 ppm; which became a requirement of the operating specifications. The requirement was later raised to 4350 ppm during operations to remove damaged fuel and fuel debris.

Boron concentration was controlled primarily by managing the sources of make-up water to the required concentration. Boron dilution events, due to possible operational error, were prevented by ensuring that only approved sources of make-up water were available for injection. Reactor coolant samples were analysed weekly to confirm this method of control. A 500 mL sample of water (at 80 mSv/h on contact) was sealed, doubly packed and flown by chartered flight to a certified laboratory. Analysis results were reported by noon on the same day.

A disadvantage of the weekly analysis method was that it could not detect changes of boron concentration in a timely manner resulting from possible equipment failures. Therefore, an in-line boronometer was installed at the sampling location. The boronometer would indicate any boron dilution events more frequently and more rapidly.

6.4.3. Chernobyl NPP

6.4.3.1. Conditions following the accident

The Chernobyl NPP shelter object conditions bear potential hazards that are much more severe than for normal facilities containing nuclear and radioactive materials. From April through June 1986, helicopters dropped thousands of tonnes of various materials into the reactor and adjacent premises that did not have a roof remaining. The amount and composition of these materials is summarized in Table 6.

The materials covered the building's debris and reactor fragments with a thick layer (in some places reaching about 10 m). It is expected that a great amount of nuclear material is located under this layer. Thus, in the reactor shaft and adjacent premises there are at least three configurations for FCMs: (1) reactor core fragments, (2) fuel dust and (3) lava-like FCMs.

The greatest hazard from the shelter object to people and the environment is the presence of radioactive materials with total activity of approximately 600 000 TBq. At the time of this publication, it is believed that about 95% of the nuclear fuel (about 200 t) remains inside the shelter object.

TABLE 6. DRY AND LIQUID MATERIALS DROPPED ONTO THE REACTOR DEBRIS AT CHERNOBYL NPP

Material	Chemical formula	Mass (t)
Boron carbide	B_4C	40
Dolomite	$MgCa(CO_3)_2$	1200
Marble aggregate, clay, sand, etc.	—	3500
Lead (pellets + ‘bars’, etc.)	Pb	6700
Trisodium phosphate (solution)	Na_3P0_4	2500
Other dust suppressive compositions (solutions)	Latex of СКС-65гп type, ethyl alcohol residue (barda), liquid glass, raw rubber СКТН, etc.	2700
	Total	640

6.4.3.2. Monitoring

There was, at the time of this publication, no access to debris in the reactor shaft. The location and condition of the fuel thrown out of the reactor, and covered with a several metre-thick layer of isolating material, is unknown. It is difficult to carry out a survey at the damaged unit owing to the high dose fields existing so far into the premises (up to hundreds of röntgen per hour). When the shelter was being constructed, a large mass of fresh concrete penetrated into the upper premises and hardened there in the form of flows and separate block masses.

Since 1988, no more than 60% of the shelter premises have been surveyed. Other premises are also not accessible either owing to high radiation fields or because of impervious barriers that arose during the explosion. The explosion also resulted in the destruction of structures and generation of lava-like FCMs. Some of the premises are inaccessible because of barriers created due to concrete injection during construction of the shelter object. Hence, a significant part of the shelter object remains an unexplored zone and continues to be one of the most serious issues.

At the beginning of the accident, there was no reliable equipment to control criticality, there were also no reliable physical barriers to stop radioactivity from spreading into the environment. Over time, improved systems have been implemented in four stages.

Buoys

Beginning in May 1986, the buoy system was created to monitor the condition of the destroyed reactor. The buoy shown in Fig. 38 was placed on the debris during June and July 1986. Each buoy device is a shell in the form of a conic frustum that was placed from above directly into the reactor debris by helicopters and later by lifting cranes. The placement of five sensors on each buoy includes the following:

(a) Heat flux sensors at the bottom surface;
(b) Air temperature sensors;
(c) An ionization chamber to measure radiation;
(d) Hot-wire airflow metres to measure airflow combined;
(e) A preamplifier for the flow metres.

These provided measurements of temperature and density of heat flow on the surface of the debris. Each buoy had a long cable connected to a control console. In total, 15 buoys with approximately 160 various detectors were installed during the system’s operation (August–November 1986).

The first accurate estimations of the reactor core condition were performed by this system. The measurements showed that radioactive decay was consistent with calculations that confirmed the absence of evidence of a self-sustaining fission reaction. The decay heat rate derived from buoy measurements over time showed that more than 90% of the fuel remained in the reactor premises. The buoy system was taken out of service when the roofs were constructed.

Shater system

In 1986, detectors and operating control devices were installed throughout the shelter. These were the basis for the creation of a comprehensive diagnostic system that covered most of the hazardous premises in the shelter object. The Shater system, commissioned in 1987, was the outcome of this effort and was operated through the year 2000.

Finish and signal systems

The finish and signal systems began routine operation in July 2000. They were designed to monitor FCM conditions by measuring the density of neutron flux and gamma radiation exposure dose rate; and to provide external sound (siren) and light (red light) alarms if changes to FCM conditions can lead to exceeding the neutron flux density limit and limit for the gamma radiation exposure dose rate. These systems perform continuous monitoring for criticality in the locations with the largest accumulations of FCMs.

Automatic information and monitoring system (AIMS)

Despite the ability of the finish and signal systems to provide nuclear and physical parameter information for FCM conditions, these systems are insufficient to comprehensively monitor the shelter object condition. A decision was made to design and construct a more enhanced system that would cover the shelter object monitoring to include other important parameters for radiation safety, conditions of building structures and seismic activity.

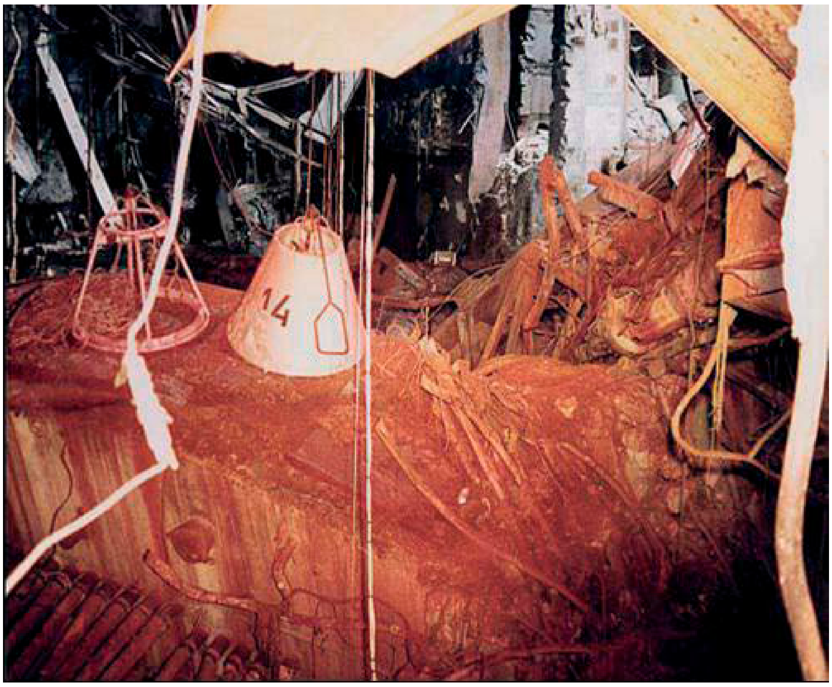


FIG. 38. The buoy system on the edge of the Elena structure. (Courtesy of Chernobyl NPP, Ukraine.)

The design and construction of AIMS was implemented within the frame of 17 tasks of the shelter implementation plan. The system provides summarized information on the shelter object main parameters necessary for timely implementation of measures to maintain and to improve the level of shelter object safety. AIMS also provides measures on mitigating the consequences of an accident at the shelter object.

AIMS offers the following types of control:

(a) Nuclear safety;
(b) Radiation;
(c) Conditions of building structures;
(d) Seismic.

AIMS is designed hierarchically and includes systems at two levels:

(a) Lower level — Primary control system receiving information directly from sensors;
(b) Upper level — Integrated monitoring system performing the functions of a centralized information system.

In general, AIMS allows operational staff to receive operational information on the processes at the shelter object and also provides information supporting operational decisions. AIMS will receive more detailed information on the shelter object condition and will maintain the records of such data to forecast, analyse and take preventive measures to avoid incidents and accidents.

6.4.3.3. Maintaining subcriticality

A system for the supply of neutron absorption solutions to the areas with FCM (onto the surface of FCM) was introduced to maintain accumulations of nuclear materials in a subcritical state. Three systems are in operation, the gadolinium nitrate solution supply system; the installation for the injection of neutron absorption solution; and the upgraded dust suppression system (MSPP, see 5.1.4.2).

Gadolinium nitrate solution supply system

A gadolinium nitrate solution supply system is a control system performing the protection function and designed to maintain the FCM in a subcritical state. The maintenance of FCM in a subcritical state is achieved by a reduction of the effective neutron multiplication factor (K_{eff}) in FCM accumulations located in Unit 4 debris, and in the area of the reactor shaft, by the supply of a neutron absorbing solution of gadolinium nitrate onto the surface of these accumulations.

The basic technical characteristics of a gadolinium nitrate solution supply system are as follows:

(a) Concentration of gadolinium nitrate solution in water is 0.1%;
(b) Volume of prepared and stored solution (two tanks in total) is 3.2 m^3;
(c) Capacity of solution feed pump (two pieces in total) is 3 m^3/h;
(d) Solution is supplied on a periodic basis once a year, as well as when a critical level of parameters measured by the system is achieved.

A gadolinium nitrate solution supply system is brought into operation remotely by staff in accordance with plans or when FCM monitored parameters exceed the critical levels established in the control systems. This system provides a supply of neutron absorbing solution only to the reactor shaft and FCM accumulations located in the area of the shaft. The neutron absorbing solution is not supplied to other nuclear hazardous accumulations (e.g. accumulations in the area of the fresh fuel suspension unit). The supply of neutron absorbing solution to those accumulations of FCM is possible partially by MSPP. When using a gadolinium nitrate solution supply system and MSPP, the neutron absorbing solution on the surface of FCM could lead to difficulties in reducing K_{eff} in the depth of the FCM accumulations.

Installation for the injection of neutron absorption solution

A neutron absorption solution injection system was designed to reduce K_{eff} in FCM accumulations located in the under reactor area. The K_{eff} value is reduced by the operational injection of a neutron absorption solution of gadolinium nitrate onto the surface of accumulations. The solution is injected upon reaching the safe operation limits that are measured by the systems controlling the condition of these FCM accumulations. The installation is placed at a separate decking. The solution of neutron absorption material is supplied through a pressure conduit laid in a casing pipe of backup well.

The installation design consists of the following main components:

(a) Frame for placement and fixation of the various elements of the system;
(b) Pneumatic–hydraulic system with two bottles for placement and storage of 40 L of 1% solution of gadolinium nitrate and a reserve of pressurized gas necessary for injecting the solution;
(c) System for automatic control of the installation during the preparation for the ready mode and within the ready mode;
(d) Temperature control system to prevent freezing of the neutron absorbing solution in the installation when the air temperature is below zero;
(e) Pressure conduit for neutron absorbing solution supply to FCM accumulations.

6.4.4. Fukushima Daiichi NPP

At the Fukushima Daiichi NPP, the normal operational instruments for monitoring criticality were destroyed by the accident. To compensate for this loss, means to detect increased nuclear reactions were introduced. The reduction of the shutdown margin to criticality is conducted by detecting the presence of short lived noble gases in the containment vessel gas control system [7]. The specific parameter measured is the concentration of the radionuclide ^{135}Xe; which is a short lived by-product of nuclear fission. In case nuclear fission increases and ^{135}Xe is generated, it would be released to the containment vessel because of the breached condition of the reactor pressure vessel and would pass to the containment vessel gas control system where it would be detected.

The temperature of the water flowing through the reactor vessel is monitored. Should an abnormal temperature rise or presence of xenon be detected, one method of control would be to inject boron. Boron-10 is a neutron absorber that prevents neutrons from reaching a concentration required to sustain a critical condition.

In the future, when fuel debris is to be removed, additional means will be needed to ensure subcriticality as the material is disturbed. Some means of detecting an increase in neutron activity needs to be incorporated into the overall scheme for debris removal.

6.5. CHARACTERIZATION OF DAMAGED FUEL AND FUEL DEBRIS

Knowledge of the damaged fuel and fuel debris condition is essential for decisions on how to remove those materials [11]. Observations and the collection of physical data are likely conducted using remote sampling, visualization and other measurement technologies. Before removal systems and equipment can be selected, designed and fabricated, the following information should be determined to the extent needed for retrieval systems and equipment:

(a) The post-accident configuration and location of the fuel bearing materials. Fuel materials may be heterogeneously mixed with other core and vessel internal components that may have also suffered physical damage. This can be further complicated in situations like at the Fukushima Daiichi NPP where vessel melt-through has occurred and non-metallic materials such as concrete can become a constituent.
(b) Dimensions of individual masses.
(c) Data about physical characteristics of the materials such as hardness, toughness, and friability for cutting and handling.
(d) Structural integrity of the fuel/core support structure, lateral retention components and other vessel internals [7].

6.5.1. Gaining access for characterization

Severe accidents create difficult challenges for gaining access; not only because the reactor core and fuel are severely damaged, but there can also be damage to components within the vessel as well. These components include control rod drive mechanisms; core support structures at the bottom, sides and above the fuel, water and flow channels; and steam separators, among others.

Gaining access to these areas and to the fuel itself can be viewed as a major step in the overall programme. The first action needs to be focused on the characterization of the extent of damage. This is done to understand the physical conditions that will be encountered when reaching the fuel and the conditions of the damaged fuel itself. The types of challenges for placement and operation of characterization equipment can be similar to debris removal; however, the scope of the equipment and methods to gain follow-up access to the fuel debris is different for the following reasons:

(a) The weight, size and force requirements for characterization devices can be much smaller than for cutting and removal of fuel debris.
(b) More precise placement in close proximity to fuel debris is needed for characterization of its physical conditions.
(c) The size of openings for characterization devices and end effectors can be smaller; of the order of a metre or less, compared with openings of several metres for removal of materials.
(d) The location of openings for characterization is not necessarily the same as for fuel debris removal. If feasible, using side penetrations might be considered even if the ultimate debris removal plan is from the top.
(e) It is preferable to not remove the reactor vessel head to insert instruments from the top because, if possible, the conditions beneath the head should first be determined.
(f) Characterization can be conducted underwater even if debris removal will be under dry conditions. A decision regarding wet versus dry debris removal may depend on the results of characterization.

At A1 NPP, taking into account the specific characteristics of the A1 NPP accident (i.e. nuclear accident without physical damage of the main reactor equipment and reactor hall structures), access for characterization of fuel did not represent a critical issue. Characterization of fuel was done mainly based on standard operational procedures such as the evaluation of measured data (burnup, residual power, cooling time), thermal and neutron physical calculations and comparison with some extra information obtained during the fuel experiments in the A1 NPP hot cell. Various technical problems needed to be analysed and technically resolved for the retrieval and further management of fuel transport to the Russian Federation (primarily damaged fuel). These activities were significantly unique and required extensive effort to be safely managed. Examples of technical issues that needed to be dealt with included:

(a) The need to cut open casks with stored fuel without contact with the fuel itself;
(b) Transfer for storage of fuel from the old casks to the new ones;
(c) Drainage and further treatment of highly contaminated cooling media from casks with damaged fuel;
(d) Size reduction and disposal of the old casks where damaged fuel was originally stored.

In the case of the damaged reactor at Windscale Pile 1 in the United Kingdom, initially it was not considered possible to make a direct entry into the damaged core of the reactor for fear of disturbance and the possibility of oxidation of pyrophoric uranium hydride acting as an initiator of further release of stored Wigner and subsequent core conflagration. On this basis, early characterization was carried out by non-contact methods, such as CCTV viewing from the reactor discharge face of some graphite samples removed for characterization purposes from regions of the core of that were remote from the damaged area. A comprehensive programme of research concluded that the earlier fears of runaway thermal excursion were unfounded and subsequently, in situ, intrusive inspection of the core became possible using a fibre optic endoscope deep within the FAZ. This characterization work facilitated the development of plans for the removal of the burnt fuel and associated fuel debris.

At TMI-2, the initial access for characterization was by inserting a video camera through a penetration in the reactor vessel head where control rods were normally inserted. This was extremely useful because it showed that the actual conditions were quite different than what was predicted by various computer simulations. The radiation level was low enough and this operation was conducted by workers. Access for in situ characterization at the Chernobyl NPP is described above in Section 6.4.3.

At the Fukushima Daiichi NPP, access to the PCV is as important as access to the reactor pressure vessel because of the damage to the lower regions of the vessel. A variety of access points are being considered for characterization from the side as well as the top. Because of radiation levels, such access will require remote technology.

6.5.2. Characterization technologies

A variety of support systems are needed for characterization such as:

(a) A control centre for remote manipulation of devices, where local manual control is not possible because of radiation levels;
(b) Transport of the instruments and devices to the access opening;
(c) Placement of characterization instruments and devices in the area for further action;
(d) Retrieval of the instruments and devices, possibly with spraying for decontamination or containers;
(e) Sample retrieval and transport to the location where they are to be analysed.

All in situ devices will need to be radiation tolerant in addition to being designed for use underwater or in air. Examples of these devices include the following:

(a) Video cameras, sonar ranging, laser imaging, and fibre optic scanning endoscopes for visualization;
(b) Probes for measuring hardness, toughness and friability, perhaps in combination with cameras;
(c) End effectors for sample retrieval with capabilities such as small hole boring, chipping, grabbing, and placing in containers;
(d) Radiation detectors (gamma and neutron) gamma cameras for singular, intense source locating.

Examples of innovative devices are illustrated in the following figures. Figure 39 shows a snake arm from Oliver Crispin Robotics limited. Figure 40(a) shows a shape changing crawler from International Research Institute for Nuclear Decommissioning (IRID), Japan, and Fig. 40(b) shows a swimming robot from IRID, Japan.

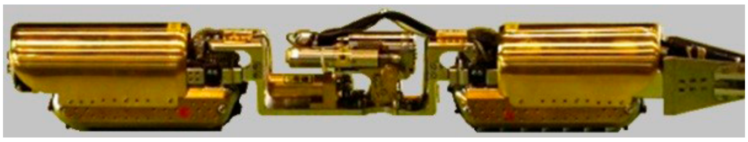
6.6. DAMAGED FUEL AND FUEL DEBRIS REMOVAL

Damaged fuel removal is one of the most challenging tasks of post-accident recovery. The conditions of radiation and radioactive contamination as well as physical damage and other conditions surrounding the facility and the fuel itself will be such that equipment and methods normally used for fuel handling and fuel management will be of little or no use for tasks to retrieve material. Both clear-cut and elaborate types of remote methods and technology will be needed.

TMI-2, the Chernobyl NPP and the Fukushima Daiichi NPP are the accidents with severe damage to the reactor vessels and enclosing buildings. All three are of different design and configuration. Of the two light water reactor types, TMI-2 was a pressurized water reactor within a large containment, whereas the Fukushima Daiichi reactors were of a boiling water reactor design, which features a much taller reactor vessel, less spacious containment and spent fuel pools that are located within the reactor building, which also encloses the containment. These differences, combined with the different magnitudes of fuel and core damage in each case, result in vastly different approaches to removal of damaged fuel. Fuel debris removal has been accomplished only at TMI-2. Fukushima Daiichi is next in line to plan for fuel debris removal.

The conclusion from these observations is that every case will be different for gaining access to and removing the damaged fuel and fuel debris [11]. For purposes of this publication, the TMI-2 experience is described because it is the only one of these three where fuel removal has been accomplished. Some examples are included to illustrate possible concepts for other cases.

FIG. 39. Snake arm. (Courtesy of Oliver Crispin Robotics limited, United Kingdom.)



(a)

(b)

FIG. 40. (a) Shape changing crawler, and (b) swimming robot. (Courtesy of IRID, Japan.)

6.6.1. TMI-2 example

During early planning, five system concepts for defueling were identified using a range of analytical and simulated projections of the range and degree of core damage. The concepts included:

(a) A telescoping tube manipulator that would be remotely operated;
(b) A rotating cylindrical platform supported from the reactor vessel flange and use of manual tools;
(c) The same concept as in (b) but with remotely operated tools;
(d) A bridge with a membrane barrier between defueling equipment and the reactor vessel internals;
(e) A lead shielded platform with access ports.

None of these were adopted, which emphasizes the need for visualization and other characterization of the actual conditions.

When the true extent of conditions within the reactor vessel became known, two additional concepts were developed. One was to remotely shred the entire core to rubble and transport with a vacuum system to a packaging station outside of containment. However, because of the development time that would be required, combined with technical and regulatory uncertainties, management decided on a manual scheme using a work platform about 1.3 m above the reactor vessel flange, shown in Fig. 41. This platform was shielded from beneath. Work was conducted through the slot shown in the figure. Rotating the platform served to position the slot above various areas within the vessel.

The hard, molten mass at the centre of the TMI-2 core was impossible to manually break up for removal. The core boring machine (a commercial mine drilling machine adapted for TMI-2 needs) was used to break up the large solid mass into small pieces and particles that could be removed manually. The machine was initially used for retrieving samples from the damaged core with hollow drill strings, and proved very useful to augment the manual defueling by boring several hundred holes through the molten material. The machine was also used for cutting the lower core support structure to gain access to the fuel debris in the bottom of the reactor vessel. Figure 42 includes a cross-section view of the core drilling machine mounted on the reactor vessel, and a photograph that shows the top of the drill below the adjacent platform.

6.6.2. Fukushima Daiichi NPP approaches

The approximate location of fuel debris inside damaged Fukushima Daiichi reactors is illustrated in Fig. 43. Estimation diagrams of fuel debris distribution are based on the results of the subsidy program project on decommissioning and contaminated water management, a joint project by IRID and the Institute of Applied Energy (IAE) with the support of TEPCO. The following three overall approaches are being considered to remove the fuel debris [87]:

(a) Removal of fuel debris from the upper part of the water-covered containment vessel;
(b) Airborne, with removal from the upper part of the containment vessel;
(c) Airborne, with removal from the side of the containment vessel.

The idea behind the first method is to fill the containment vessel with water and remove the fuel debris. The advantages of this method are cooling of the fuel debris, radiation shielding and prevention of radioactive dust from scattering. Potential issues include keeping the water within the containment vessel, seismic safety and control of criticality [87].

FIG. 41. TMI-2 work platform. (Courtesy of the NRC.)

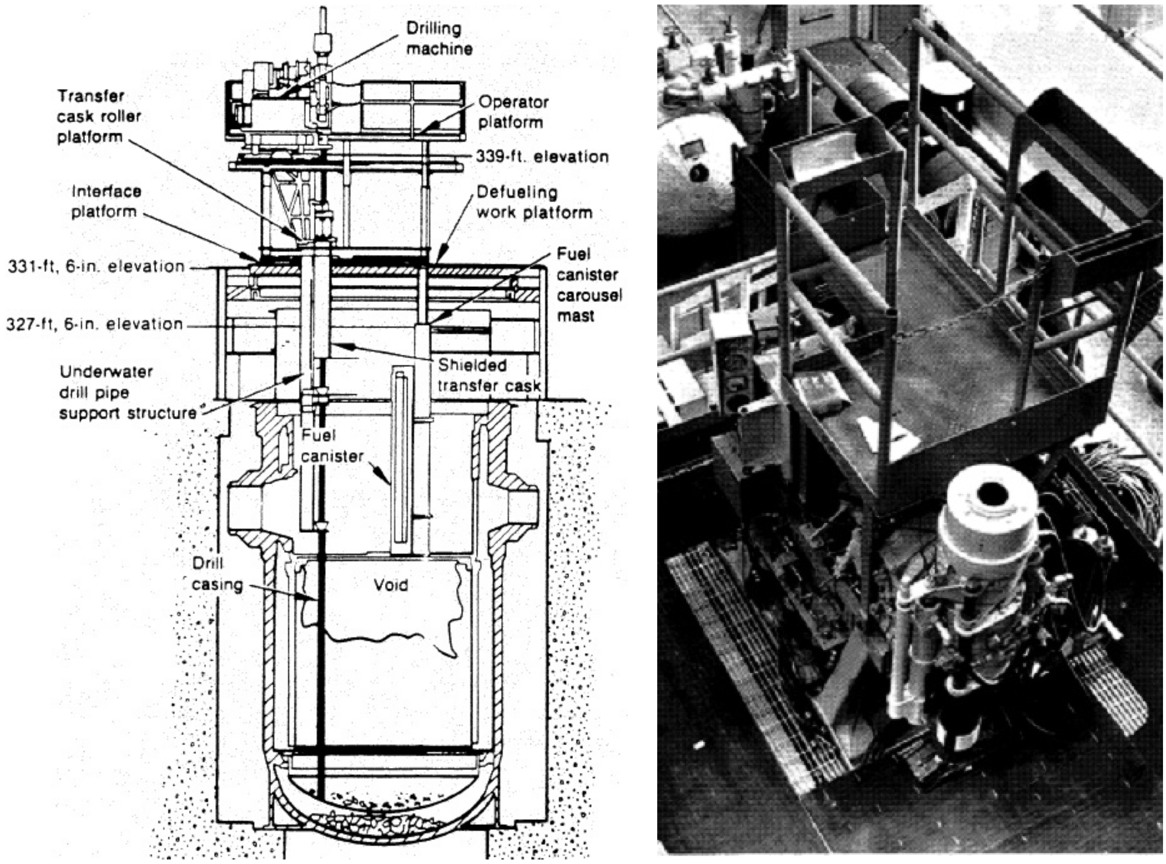


FIG. 42. TMI-2 core drilling machine. (Adapted from Ref. [7].)

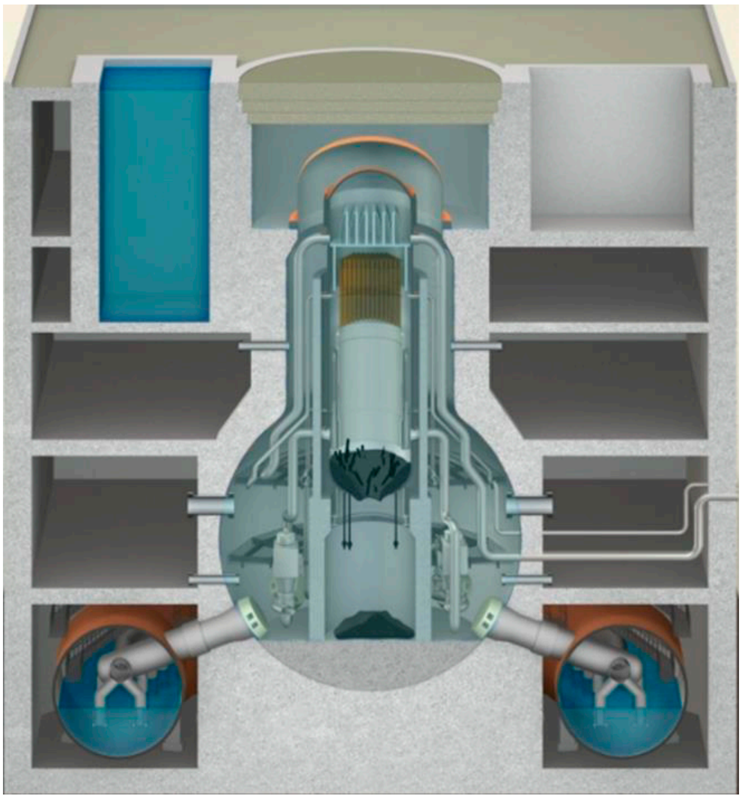


FIG. 43. Illustration of approximate location of fuel debris inside damaged Fukushima Daiichi NPP reactor. (Courtesy of IRID, IAEA and TEPCO.)

Two airborne methods include removal of debris either from the top or the side (on the first floor of the reactor building). One of these options may be implemented if filling the containment vessel with water proves to be impossible. Potential issues are the scattering of radioactive dust and lack of radiation shielding. Regardless of the final method chosen, many of the functions shown in Fig. 44 will be needed and many of them need to be remotely operated.

There are many unknowns and uncertainties related to the physical and radiological situation in the reactor buildings in general and especially inside the PCVs. Additional information and overview of this topic are provided in Refs [88, 89].

6.6.3. Containerization

To prevent spread of contamination, damaged fuel assemblies and fuel debris would usually be placed in containers as they are removed. Figure 45 shows the TMI-2 fuel debris canister that was used for this purpose. The canisters were positioned underwater beneath the platform and filled manually. As each was filled, it was placed in a shielded transfer container and removed to interim storage within the on-site spent fuel pool. In addition to the debris canister, two other designs were created with the same physical envelope. One was in the recirculation loop for filtration to capture small fuel particles from water within the reactor vessel. The third design was called a 'knock out' canister. It was for gravitational separation of small pieces of fuel material while using a water suction device to vacuum materials into the debris canisters.

6.6.4. Other examples of equipment

Since the TMI-2 accident, considerable advancement has been made with remote-controlled systems that could be adapted for fuel removal. One example is shown in Fig. 46. One is a telescoping mast, the second is a rotating work platform with manipulators and the third is a large power manipulator.

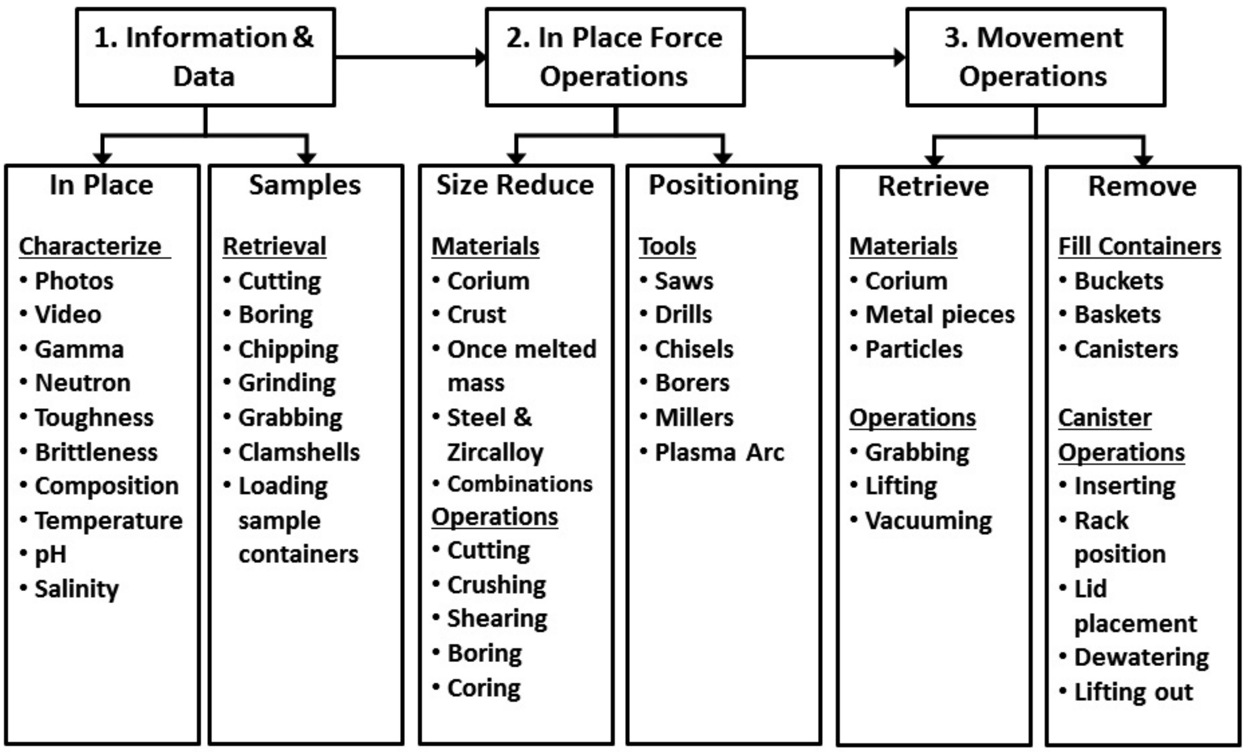


FIG. 44. Functions required for fuel debris removal.

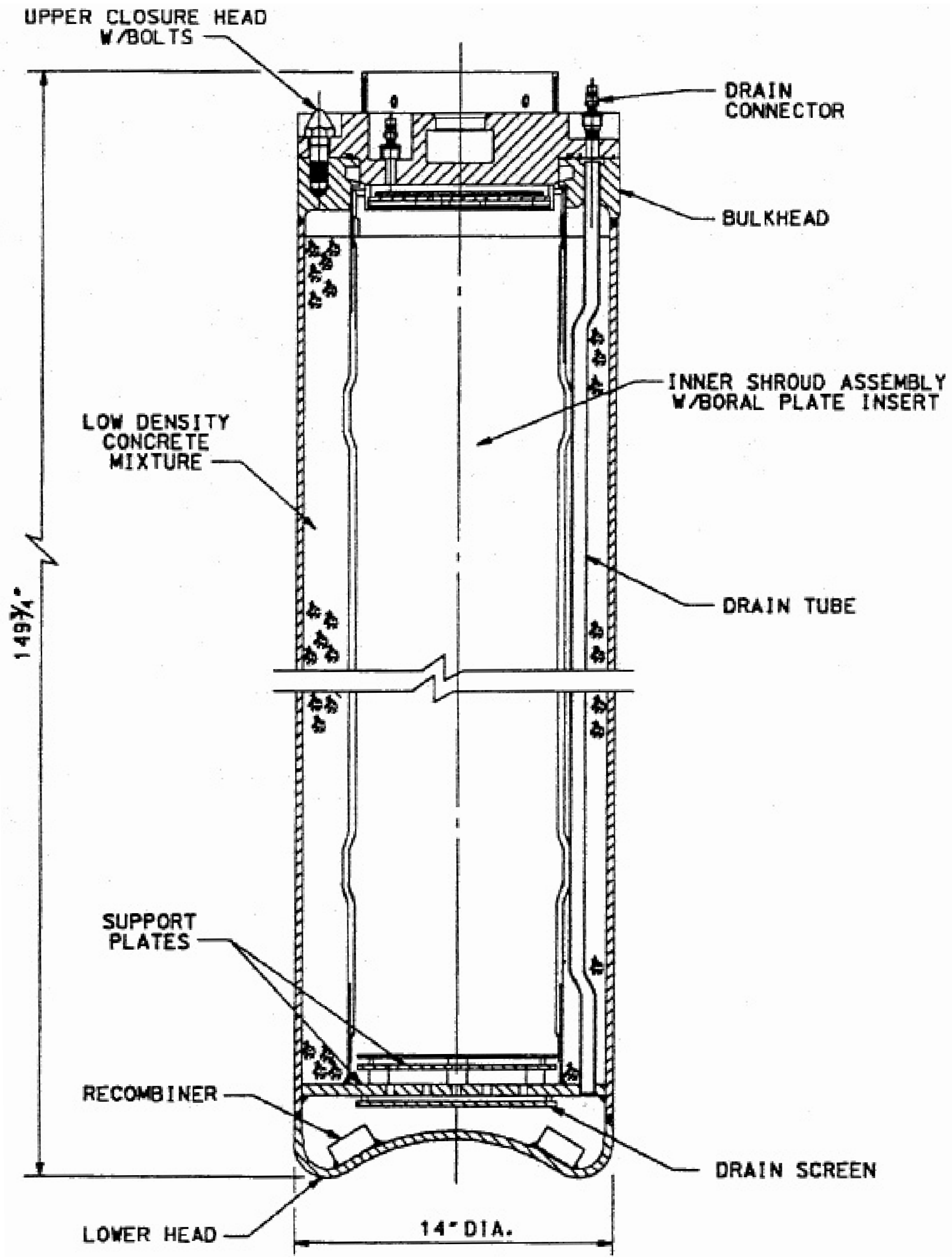


FIG. 45. TMI-2 fuel debris canister. (Courtesy of EPRI, USA.)

Applying any device, however well developed, to damaged fuel removal will be a complex process involving design, fabrication, assembly, integrated system testing and mock-up testing. A few considerations include:

(a) Distances from work areas to the furthest location of fuel material to be removed.
(b) Whether the materials are to be removed underwater or in air. This will greatly affect equipment, tools, and selection and design.
(c) Radiation tolerance of materials and electronic circuits.
(d) Components such as platforms, cranes, lifting equipment, boring, shearing, sawing, arc cutting, grasping end effectors and removal containers.
(e) Additional support equipment for decontamination, maintenance and repair that also may have to take place in highly radioactive areas.

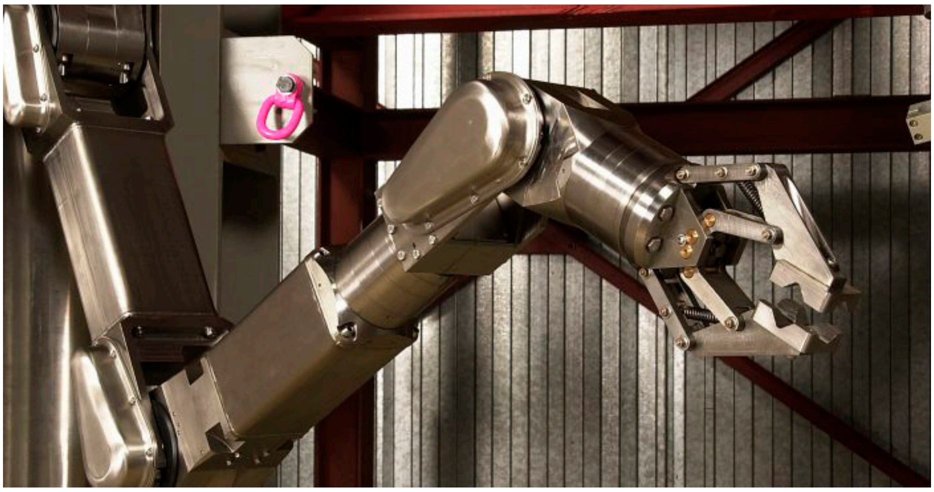


FIG. 46. Remote manipulator system Maestro. (Courtesy of Cybernétix, France.)

6.7. STORAGE OF NUCLEAR FUEL AND FUEL DEBRIS

Storage of damaged nuclear fuel and fuel debris is to some extent addressed in several IAEA publications, such as Refs [1, 18, 90, 91]. On-site or off-site storage are options to be considered based on the availability of storage capacity, internal or external transport routes and other technical and non-technical factors, however in the majority of post-accident cases the on-site option prevails.

The duration of the interim storage period may not be known and, owing to political uncertainties regarding the repository, it is desirable to reserve the option for the fuel treatment or conditioning steps that may occur prior to the geological disposal. Every effort should be made to condition damaged nuclear fuel and fuel debris at the source to avoid spread of activity or contaminated materials.

The most recent example of the on-site storage of nuclear fuel after an accident is in Fukushima Daiichi Unit 4 nuclear fuel storage in the common spent fuel area near Unit 4. The removal of nuclear fuel from the Unit 4 pool commenced on 18 November 2013 and was completed on 22 December 2014. Some 1535 new and spent fuel assemblies were transferred, the majority of them for storage in the common spent fuel area, see Fig. 47. A total of 180 new fuel assemblies were transferred to the Unit 6 fuel pool [41, 92].

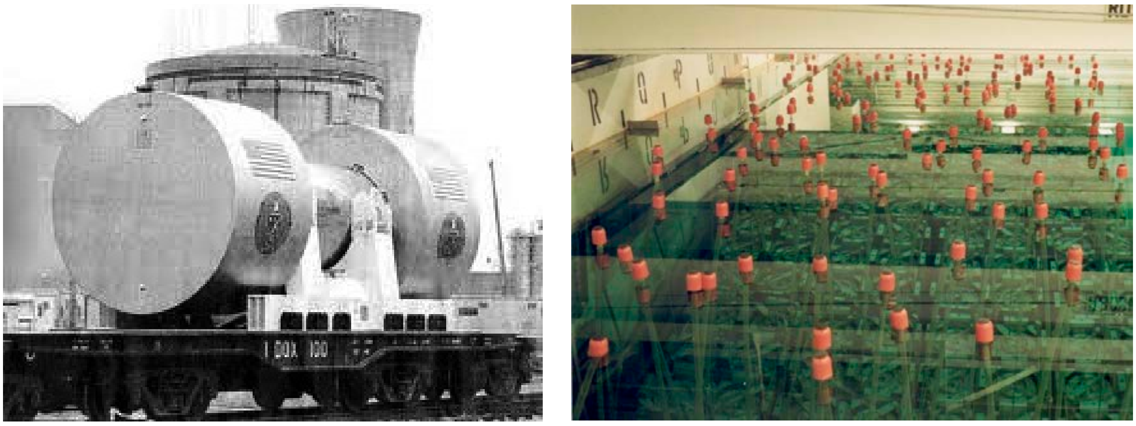
6.8. TRANSPORT PRIOR TO REPROCESSING OR TOWARDS FINAL DISPOSAL

An example of transport prior to reprocessing or towards final disposal is the removal of 440 fuel assemblies from the A1 NPP, mainly from pre-accident operation of the reactor, from Slovakia to the Russian Federation (country of origin). The first batches of non-damaged fuel were sent between 1983 and 1990. For retrieval and transport of the 132 most damaged fuel assemblies, it was necessary to develop and fabricate special drainage and encapsulation equipment, as well as other equipment such as transport containers.

The fuel was retrieved from the long term storage facility, repacked into new cases and transported in special containers between 1996 and 1999. By removing fuel from the plant, conditions were established for the A1 NPP decommissioning implementation activities [80, 81].

At TMI-2, fuel debris canisters were transported to a long term storage pool at a government site (see Fig. 48). The canisters were transported by rail (as shown in Fig. 48(a)) to the pool storage shown on the right in Fig. 48(b). Later, the debris canisters were removed from the pool, vacuum dried and placed in long term dry storage, shown in Fig. 49.

FIG. 47. Fukushima Daiichi common spent fuel area.



(a) (b)

FIG. 48. (a) Transport, and (b) storage of TMI-2 fuel debris canisters. (Courtesy of USDOE, USA.)

FIG. 49. Placement of TMI-2 fuel debris canisters into long term dry storage. (Courtesy of USDOE, USA.)

6.9. PROCESSING DAMAGED FUEL AND DEBRIS IN LIEU OF STORAGE AND DISPOSAL

Other than storage and direct disposal of damaged fuel and fuel debris, in some circumstances processing the fuel may be a solution, either for more efficient disposal, or for possible reuse. There are three possible approaches: aqueous reprocessing, electrochemical dissolution and vitrification as described below. All three options require complex physical chemistry in a highly radioactive environment within medium to large facilities [7].

Two cases in which aqueous processing has been conducted include the following:

(a) The damaged fuel from the A1 NPP that was returned to the Russian Federation where it was reprocessed.
(b) At Paks in 2003, a limited number of fuel rods were damaged when they overheated in a cleaning tank external to the plant's reactors. In 2014, the fuel was transported to the Mayak nuclear facility by rail for reprocessing [93, 94].

6.9.1. Aqueous reprocessing

Aqueous reprocessing of spent fuel has been conducted for decades. It represents a kind of hydrometallurgy treatment using aqueous solutions that dissolve the metal, at times also using electrolytic cells to separate them (e.g. zinc production, copper refining) [95].

This type of reprocessing capacity was developed in France (La Hague), Japan (Rokkasho), the Russian Federation (Mayak) and the United Kingdom (Sellafield). The Thermal Oxide Reprocessing Plant (THORP) in Sellafield (Windscale Pile 1) is an example of a facility that uses aqueous reprocessing. Figure 50 is from 1994 and shows the first fuel being lowered into the THORP receipt and storage building from the rail bay. Fuel is stored in the receipt and storage building prior to being transferred to the head end building for shearing. THORP started operation in August 1997 and combines all the facilities necessary for reprocessing spent oxide fuel under one roof. Operations are divided into the main areas:

(a) Fuel receipt and storage;
(b) Head end plant operations where spent fuel is chopped up and dissolved in nitric acid;
(c) Chemical separation where uranium, plutonium and waste products are separated out [7].

The Nuclear Decommissioning Authority confirms that a decision, in 2012, was taken to close THORP by 2018.

6.9.2. Electrorefining

Electrorefining, also called pyroprocessing, is an electrometallurgical treatment for spent nuclear fuel that uses molten salt to recover the uranium and other actinides. Descriptions of examples of this process can be found in Ref. [96]. Two examples are described briefly below.

The Central Research Institute of Electrical Power in Japan has reported on the electrorefining of uranium–plutonium–zirconium alloy fuel from a metallic fast breeder reactor using the process depicted in Fig. 51 and described in Ref. [97]. This was conducted in a hot cell with an argon atmosphere. The electrorefiner steel vessel used a molten salt electrolyte consisting of a liquid lithium–chloride–potassium eutectic mixture. The cathode consisted of liquid cadmium contained within a ceramic crucible.

Electrorefining is examined by the Central Research Institute of Electrical Power and Japan Atomic Energy Agency as a part of the R&D programme of the Fukushima Daiichi NPP damaged fuel and fuel debris management options [98].

An example of development and demonstration of electrochemical processing applied to a specific fuel type (used fuel from Experimental Breeder Reactor II) conducted at the Idaho National Laboratory is described in Ref. [99]. Some of the equipment used in the process is shown in Fig. 52 and an example of the uranium product collected during advanced refiner testing is illustrated in Fig. 53. Further R&D in this area is needed to achieve use at the industrial level.

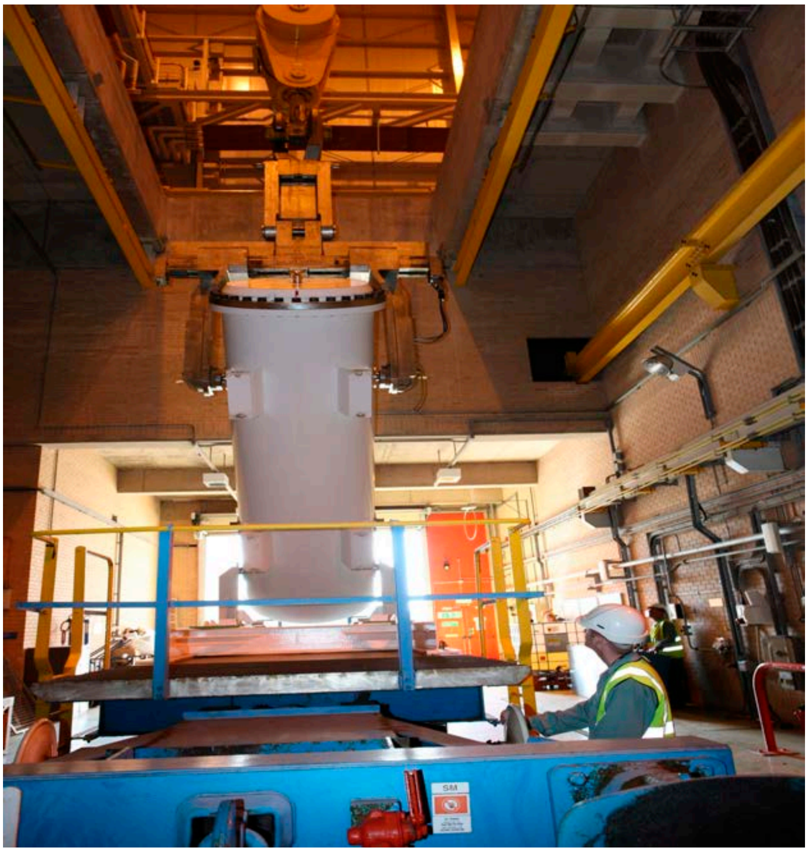


FIG. 50. First fuel being lowered into the THORP receipt and storage building from the rail bay in 1994. (Courtesy of Sellafield Ltd, United Kingdom.)

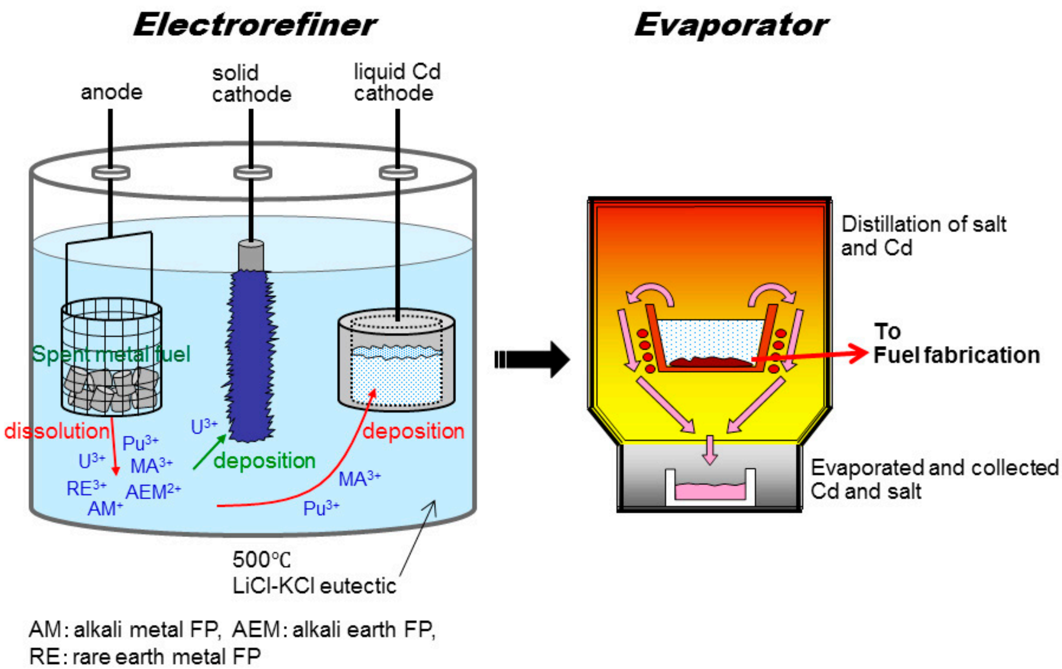


FIG. 51. Principle of separation by electrorefining. (Courtesy of Central Research Institute of Electrical Power, Japan.)

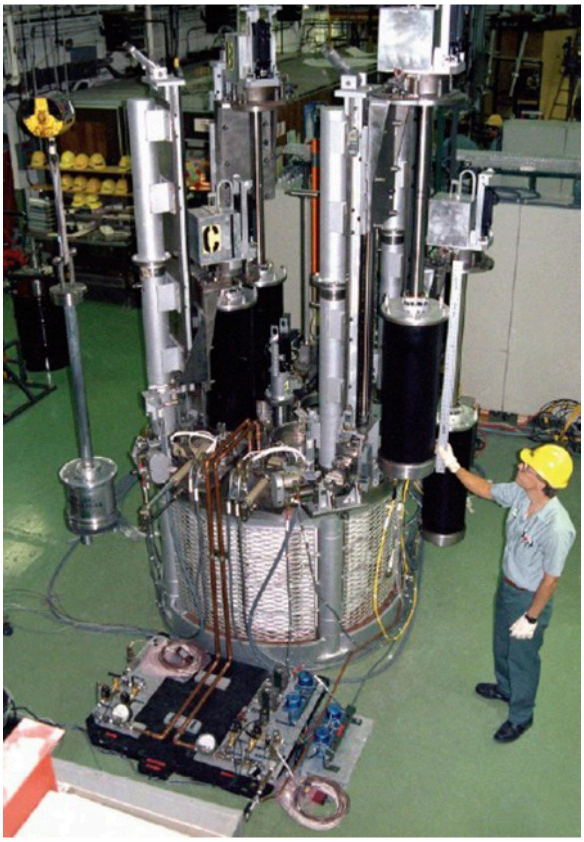


FIG. 52. Electro refining equipment. (Courtesy of USDOE, USA.)

FIG. 53. Uranium product collected during advanced refiner testing. (Courtesy of USDOE, USA.)

6.9.3. Vitrification

"During the TMI-2 cleanup projects, the contents of three of the spent fuel canisters were vitrified in a hot cell. It was conducted as an experiment to understand the technical feasibility. However, the eventual decision was to opt for long term storage and ultimate direct disposal as described in the previous subsection. Since that time, vitrification has advanced considerably for stabilization for disposal of high activity fission products and reprocessing waste, but not for fuel" [7].

Figure 54 illustrates the magnitude and complexity of such facilities.

7. CONSIDERATIONS AND RELEVANT ASPECTS TOWARDS FINAL DECOMMISSIONING

7.1. DECOMMISSIONING CONTRASTS

For any NPP, regardless of whether it has suffered an accident, the following applies:

"The purpose of decommissioning, including associated site remediation, is to establish the facility end-of-life disposition and, ultimately, to remediate the site to a safe and acceptable state for the long term. The goals for decommissioning are to reduce risk to on-site personnel and the public and to protect the environment from the distribution of radioactive substances via natural pathways, such as wind and water streams" [7].

7.1.1. Release from nuclear material controls

Under non-accident circumstances, elimination of the need for nuclear material controls is a primary condition sought for a facility that has reached the end of its operating life and is beginning the process of a planned, permanent shutdown. This is usually achieved by removal of nuclear fuel from the reactor and reactor storage pools to be stored or shipped for reprocessing or disposal elsewhere.

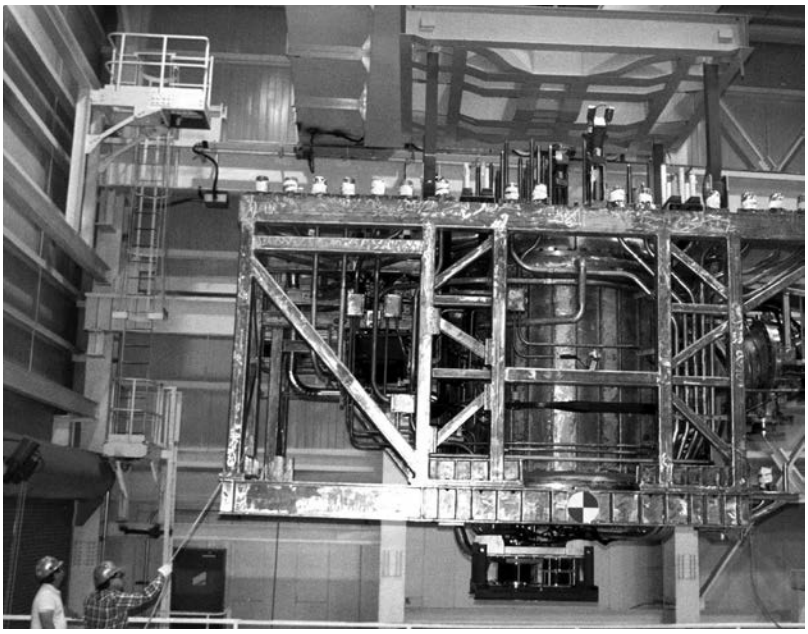


FIG. 54. Defence waste processing facility melter assembly prior to installation. (Courtesy of USDOE, USA.)

In contrast, release from nuclear material controls for an accident damaged facility is much more complex. Depending on the severity, this may not be possible for a considerable amount of time, until fuel and fuel debris can be removed such that the residual can be classified as contamination and not as nuclear material. In the case of TMI-2, for example, this required ten years. Removal of severely damaged fuel will require considerable technological adaptation and development of special tools and equipment, both directly and remotely operated.

7.1.2. Release from nuclear regulatory control

Following permanent shutdown for non-accident situations, the next goal is to achieve conditions that release a licensed facility from all nuclear regulatory controls. This is typically achieved by making the nuclear power production facilities completely harmless through immediate dismantling, deferred dismantling (long term safe storage), entombment or some combination of the three.

For an accident damaged plant, it is unrealistic to predict the ultimate end state conditions of a site early in the cleanup activities. A final end state decision will need to consider many factors. Some examples are dose rates to decommissioning workers; the types, amounts, and conditioning of wastes; and disposal of processed water. Decisions will be needed for the final disposal of waste, including disposal of spent fuel and nuclear fuel debris. These uncertainties and related decisions are major challenges for the owners, regulators and leaders within the State of the accident facility.

7.2. LONG TERM SAFE STORAGE

Because the conditions for each accident are unique, there is no standard for determining readiness for decommissioning following an accident. For the Fukushima Daiichi NPP, it is too early to completely specify the preconditions for decommissioning. None of the other three NPPs that experienced the most severe fuel damage have yet to achieve the final end state for complete decommissioning. All three are in safe storage mode. A brief summary of each is here; additional details are at the end of Section 7.2.

(a) The damaged Windscale Pile 1 unit is currently in a care and maintenance condition with a plan to place it in safe storage in the next several years with final decommissioning to occur around 2050.
(b) At the TMI site, the undamaged Unit 1 is operating normally and decommissioning is planned to start within the next 20 years. The TMI-2 plant is in a safe storage mode with a plan to complete dismantling and site remediation as part of combined decommissioning with Unit 1.
(c) Chernobyl NPP Unit 4 is currently in the process of being placed in a condition of safe storage with the time of the final decommissioning also projected for around 2050.

Based on these experiences, establishing a safe storage condition has become a normal interim phase for plants that have had major fuel damaging accidents. The purpose here is to provide a general idea of the considerations that are likely to form the basis of planning leading to safe storage.

7.2.1. Establishing a cleanup end state prior to safe storage

As completion of nuclear fuel and fuel debris removal approaches, a safety case is needed to address the following materials and fluids that may remain on the site for extended periods measured in years:

(a) Nuclear residues, particles and other radioactive materials remaining within the facilities;
(b) Spent fuel in storage;
(c) Fuel debris in storage;
(d) Solid radioactive waste in storage;
(e) Processed water in storage.

Decisions for the safety case are part of the drivers for the overall cleanup end state to be established for safe storage. In turn, a systematic evaluation is required to specify the conditions to be achieved for all SSCs at the cleanup end state. Methods for developing these specifications can be found in Ref. [100]. Although that publication was written for non-accident plants, the process is the same. Historically, establishing the safe storage condition for TMI-2 was one major predecessor for the methods in that publication.

Some examples of subjects that will need to be addressed to arrive at specified conditions include the following [7]:

(a) Requirements for the regulatory and technical inspections, care and maintenance of any on-site stored wastes and fuel bearing materials;
(b) Periodic inspections with procedures that specify what is to be inspected, the frequency, criteria for evaluation of conditions and the walk-through path, including roof inspections;
(c) Ability to purge closed areas prior to entries for inspection;
(d) Prevention of serious spread of contamination by airflow pathways; filtered exhaust ventilation may be necessary;
(e) Preventing or minimizing in-leakage of storm water and snowmelt; removal and treatment of any such in-leakage;
(f) Ageing management of passive systems and damaged structures;
(g) Any SSCs required for the continued safety of the NPP need to be maintained in a fit-for-service condition;
(h) Structural integrity against earthquakes and other natural hazards;
(i) Fire detection and response if there are combustibles remaining;
(j) Prevention of intrusion by vermin, birds and other wildlife;
(k) Security of the site;
(l) Archive of records along with periodic inspection and repair reports.

The Chernobyl NPP can be considered as an excluding case. Units 1–3 were shut down after normal operation and will be placed into safe enclosure, while Unit 4 will be transformed into an ecologically safe system [101]. Further details are provided in Annex I.

7.2.2. Operational systems and equipment during safe storage

During preparation for safe storage, it is important to ensure long term reliability for the SSCs that maintain conditions as required. Backups and/or alternatives for installed or new systems and components may be needed for several functions. Installation of completely new systems may be more effective than attempts to use existing systems in a significantly different operating mode.

Systems needed to remain operational will vary for each situation. Here are some examples:

(a) Electricity to support lighting and systems remaining in operation or needed in an emergency.
(b) Communications and data systems as links for personnel within the facility for their safety and to support their activities.
(c) Liquid systems to pump water from sumps that may accumulate from storm water in-leakage or other sources.
(d) Ventilation for rooms that remain in use and for purging closed spaces prior to entry. There may be a need to operate ventilation fans at speeds significantly lower than their original design basis, thus requiring a change.
(e) Fire protection systems may be reconfigured to comply with recommendations resulting from a fire hazard analysis. An objective of the fire hazard analysis for deactivated buildings is to determine what systems may no longer be required for the deactivated condition because of the reduction of combustibles, elimination of initiation sources and non-occupancy by personnel.

7.2.3. Inspection activities during safe storage

Inspections during safe storage will vary depending on many factors. The systems remaining operational, as indicated above, will need related surveillances. Other inspections will generally be for the purpose of detecting changing conditions such as water in-leakage, degrading structures and animal intrusion.

The following presents an example:

"The only activities currently conducted at TMI-2 are a few maintenance routines and preventive maintenance for some systems. Routine maintenance includes checking and changing high efficiency particular filters for the air being exhausted from the containment. This flow is passive to ensure no differential pressure conditions develop within the environment. A preventive maintenance procedure verifies that radiation conditions have not changed; the procedure includes a once per year containment walk down and survey.

"The control room is operational to the extent needed for monitoring conditions and the few systems in operation. This includes electrical systems and control room ventilation. Preventive maintenance is performed on the motor control centres and ventilation fans and motors. A fire detection system is in place however, there is no active fire suppression system. This is justified by the elimination of combustibles and [minimizing] ignition sources. If a fire is detected, the fire brigade from the adjacent Unit 1 would respond. The domestic water system is partially operational and is maintained to correct occasional leaks" [7].

7.2.4. Safe storage examples[5]

7.2.4.1. Windscale Pile 1

The Windscale Pile 1 reactor is passively safe and has been in safe enclosure for some decades (subject to routine review). This deferred period will allow the decay of radioactive isotopes. It is assumed that new technologies will become available for more safe and efficient decommissioning. Financial assurance is required to allow the project to commence uninterrupted.

"The passively safe condition is based on a balanced risk review across the Sellafield site, and the reactor is approved to remain in its current condition for a significant period of time subject to routine review. Ongoing justification is needed for continuing the operation of the facility under the deferred period, now referred to as 'surveillance and maintenance', as opposed to the previously used term 'care and maintenance'. The use of this terminology signifies recognition that Pile 1 is an operational facility that will be adequately maintained in its present form within an asset care programme to replace worn out or obsolete equipment where necessary" [7].

7.2.4.2. Three Mile Island Unit 2

The configuration of the TMI-2 facilities was determined using a post-accident cleanup end state specification process. Table 7 summarizes the estimated amount of fuel debris remaining throughout the plant [102]. During the post-accident cleanup, access to remove this material could not be gained because it would have involved cutting large components and pipes in high radiation areas [7].

"The decision to place TMI-2 in safe enclosure status was based on four major issues:

(1) Because TMI-1 will continue to operate and be decommissioned for at least 30 years beyond 1990, it would be efficient to remove both facilities as part of a single project. If the licence is renewed, this would be at least 50 years later.
(2) This delay would allow substantial decay of ^{137}Cs and ^{60}Co for a range of 30–50 years; the remaining amount of ^{137}Cs would be 29–50% and ^{60}Co would be 1.9–13%.
(3) Increased financial assurance by allowing a collection of funding over the 30–50 year time frame for the estimated $869 million (2009 reference year) to decommission TMI-2.
(4) It is presumed that over this time period, there will be technology development that will make decontamination and demolition safer and more efficient" [7].

[5] This section is based on Ref. [7].

TABLE 7. LOCATION OF RESIDUAL FISSILE MATERIAL AT TMI-2 [102]

Building/location	Residual fissile material (kg)
Auxiliary and fuel handling buildings	< 17
Reactor building (excluding the reactor coolant system)	< 75
Reactor coolant system (excluding the reactor vessel)	< 133
Reactor vessel	< 900
Total	< 1125

7.3. ENTOMBMENT

Entombment is discussed in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [103] as quoted here:

> "Entombment, in which all or part of the facility is encased in a structurally long lived material, is not considered a decommissioning strategy and is not an option in the case of planned permanent shutdown. It may be considered a solution only under exceptional circumstances (e.g. following a severe accident)".

The following extract from Ref. [74] also discusses entombment:

> "A general description of an ISD project encompasses an entombed facility; in some cases limited to the below-grade portion of a facility. The envelope of the project may extend beyond the outer walls. The entombed portions of the facility are of robust construction, generally of reinforced concrete exterior that provides a migration barrier between internal contamination and the environment; with significant internal void spaces backfilled or grouted. The scope of entombment may include ancillary equipment and structures and may contain radioactive and hazardous materials and contamination within the facility and waste imported from outside the facility."

Entombment is a permanent decommissioning end state. The detailed physical completion conditions (the end state) of the decommissioned facility is project specific and needs to be in conformance with regulatory approval processes. The final condition is passive, meaning there are no requirements for ongoing operational systems or equipment within the decommissioned facility. The key to entombment is the performance assessment conducted with the use of pathway modelling to demonstrate long term safety to the environment and to public health [74].

The entombed plant would be appropriately monitored and maintained until it can be shown that the risk for leaving as is, permanently, is acceptable. A decision to entomb a facility involves several considerations; the most important include the following:

(a) Permanent institutional control needs to be maintained;
(b) The results of a dose pathways analysis for future exposures to individuals via air, surface water, groundwater and food ingestion need to be within acceptable limits;
(c) The risk to workers for implementation is reasonable.

As practised by the USDOE, entombed facilities are ones in which the significantly contaminated portions of the facility are below ground or for which a permanent cap is emplaced following the entombment. One result is that the surface area above the entombment can only be used as open space. Thus, entombment differs from greenfield and brownfield end states in that excavation of the cover area is permanently forbidden and should not be used for construction of any type of facility. A comprehensive analysis of entombment is discussed in Ref. [104].

7.3.1. Experience to date

To date, entombment has been used on a smaller scale for research test reactors, and on a larger scale for former nuclear materials production reactors and large fuel processing facilities for non-NPP materials in the United States. In 2011, the USDOE completed the entombment of two former nuclear materials production reactors at the Savannah River Site [105]. All fuel was removed, and the below grade sections of the facility were filled with cementitious grout. The above grade structures, which are built of reinforced concrete, were left in place. To restrict rainwater intrusion, high strength inter-crystalline grout caps were installed on the flat roofs above the production areas [106]. Another practical example already implemented is the entombment of Experimental Breeder Reactor II, which was put in place at the USDOE Idaho site from mid-2015 [107].

The Russian experience is presented in Ref. [108]. The intention of "Rosatom" State Atomic Energy Corporation is to entomb five shutdown uranium graphite reactors. The joint-stock company established the Pilot and Demonstration Centre for Decommissioning of Uranium-Graphite Nuclear Reactors to implement this task.

7.3.2. Implementation

Compared with the immediate or deferred dismantling of contaminated facilities, entombment offers some advantages. First, the need to remove extremely large contaminated or activated pieces of equipment can be avoided. Also, decontamination or applying fixatives to prevent spread of contamination can be significantly reduced or avoided completely. The need to ship and dispose of large amounts of radioactively contaminated and/or activated materials to waste disposal facilities can be greatly reduced. And, the potential for worker exposure to high radiation can be much less. The end results may be significant cost savings and reduced exposure. These savings are likely to be more significant for a large process facility than for NPPs.

In contrast, an entombment project requires large amounts of grout, use of construction equipment, and industrial protective safety equipment [109]. Grout delivery and placement requires demanding critical timing to ensure availability when needed. This can be a huge logistical challenge compared with a project that is mostly dismantling and demolition.

7.4. RELEASE OF SITE

International decommissioning standards recognize a range of alternatives for release of site once decommissioning actions are complete [110]. While the stated goals of all decommissioning standards are to release the licensed site from regulatory controls, they also recognize that there may be the need to place restrictions on the future use of the site. For accident sites, such restricted use may be the only reasonable alternative for the ultimate end state. Figure 55 illustrates some of the possibilities.

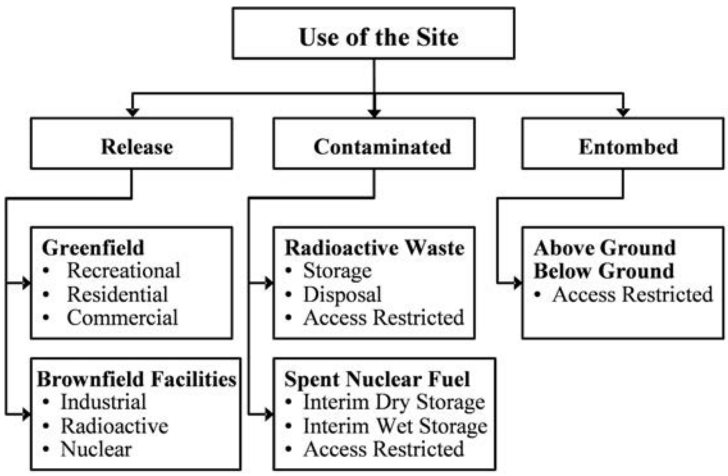


FIG. 55. Options to be considered for use of the site.

Alternatives range from release of the land for any purpose, including residential, reuse for limited purposes (such as industrial) or retaining regulatory control over its access and use, shown in the figure as 'use of the site' under contaminated or entombed. Decisions for these latter two are likely where the cost, resources, and/or risk to workers are too great to establish greenfield or brownfield conditions. For a site covering a large area, it may also be possible to employ an end state that combines these options with different end use restrictions being applied to different parts of the site.

Examples of such decisions include:

(a) A1 NPP treatment facilities: The ultimate goal of A1 NPP dismantling activities is the treatment and disposal of all operational (legacy, historical) and decommissioning radioactive waste, and dismantling of unused equipment and parts of buildings and constructions. After 2033, A1 NPP buildings that are reusable for the operation of waste treatment facilities will be refurbished and reclassified as a part of a new nuclear facility currently planned for the treatment and conditioning of radioactive waste.
(b) TMI-2: Although not planned at this time, because of the ideal location in the Susquehanna River, it could be a site for a future power plant, nuclear or otherwise. However, it is also possible for greenfield use, such as a park.
(c) Chernobyl NPP: All radioactive wastes generated as a consequence of the post-accident emergency, and later planned decommissioning and environmental remediation activities, are to be conditioned, stored and disposed of in an exclusion zone, close to the Chernobyl NPP site. Several national radioactive waste management centres recently being operated are to be enhanced and new, focused ones will be built in the future [77].

Alternatives for the end state and future use of the site need to be discussed with stakeholders before the final decommissioning strategy is selected and implemented. This approach gives the responsible parties the best possibility of designing a decommissioning plan to support likely future use of a site.

Environmental impact assessment of decommissioning alternatives needs to be performed to support decision making. The environmental impact assessment considers and evaluates a variety of impacts that may share commonalities with the nuclear regulatory control framework, many of the evaluated impacts are beyond and separate from this framework. Table 8 shows some of the environmental impacts typically assessed. A further overview of environmental impact assessment considerations is given in Ref. [110].

8. CONCLUSIONS

There are many useful lessons to be learned and much knowledge to be gained from the post-emergency decommissioning activities following a nuclear accident.

Since 1952, there have been 14 reactor accidents involving damaged nuclear fuel. Beginning with Windscale Pile 1 in 1957, four of these were at NPPs that resulted in partial or even complete destruction of the reactor fuel. Two of these (the Chernobyl NPP and the Fukushima Daiichi NPP) also resulted in significant environmental contamination, and one involved three reactors (Fukushima Daiichi). Only at TMI-2 were the physical consequences limited to the power plant site. In addition, the Jaslovské Bohunice A1 NPP suffered damage to its fuel assemblies, but the consequences at the facility were not as severe as for the other four.

Each accident is unique with specific post-accident decommissioning issues and challenges and requires a specific R&D programme. Approaches, methods and technologies to stabilize and perform cleanup activities following accidents were discussed in this publication. At Fukushima Daiichi, based on the conditions and progress to date, it is clear that technologies will be an important factor.

In the overall picture, as presented in this publication, there are three major phases (see also the footnote in Section 2 on the quiescent phase considered in the United Kingdom) of post-accident activities, each of which can require several years to accomplish the major goals. These three phases are summarized in the following:

(a) Stabilization: During stabilization activities, much of existing decommissioning experience and techniques are, to various extents, applicable to the post-accident situations although in significantly more severe radiological and physical conditions. Physical and radiological characterization, decontamination, dismantling and reconstruction, water processing, and waste management are a few of such areas.

(b) Damaged fuel and fuel debris management: This is the most difficult challenge taking into account the broad range of issues to be tackled, such as transport, physical characterization in very high radiation fields, the retrieval of damaged fuel and fuel debris from difficult to access areas, storage and disposal of these wastes in appropriate facilities [11].

(c) Towards final decommissioning: A final decommissioning end state has yet to be accomplished for any of the five cases (Windscale Pile 1, A1 NPP, TMI-2, the Chernobyl and the Fukushima Daiichi NPPs). Safe storage (care and maintenance) has been established for Windscale Pile 1 and TMI-2. The Chernobyl NPP is moving towards an ecologically safe system and the A1 NPP at the Jaslovské Bohunice site is proceeding towards brownfield dismantling. These provide useful experience with many technical variations that might be implemented and completed in the future. The ultimate decommissioning end states of Bohunice and TMI-2 are defined. Final decisions concerning Windscale Pile 1, the Chernobyl NPP and Fukushima Daiichi may be flexible and are to be left to future generations.

TABLE 8. ENVIRONMENTAL RESOURCE IMPACTS TYPICAL IN THE UNITED STATES OF AMERICA

Impact to be assessed	For each alternative being considered, assess the following
Human health and safety	For nuclear facilities, and especially for post-accident conditions, the major impact of concern is radiological impact on current and future workers (e.g. farmers, fishermen and industrial workers), the general public and visitors to the area of the site. These impacts affect not only the immediate area, but also a greater area, through a variety of exposure pathways. Other health and safety impacts can be from occupational risks, hazardous chemicals and material, and potentially harmful emissions.
Land use	Proposed use of the space at the site in comparison with current use. Identify if additional land use will be required, or what amount would be converted to green or brownfield use.
Geology and soils	Assess disturbance of geology and soils at the site for: Damage or loss of a geological feature; Loss of access to a mineral resource; Loss of agricultural land and/or degradation of topsoil quality; Ground stability related to removal of structures, landslips, etc.; Loss of recharge to an aquifer; Change in groundwater flow to surface waters.
Noise	Noise level and comparison with the surroundings, effects on humans and wildlife.
Greenhouse gas impacts and climate change	Greenhouse gases include, for example, carbon dioxide, carbon monoxide, methane and nitrous oxide. Typically for assessments of decommissioning impacts, the quantity of these gases released to the environment from the emissions of the vehicles used in the proposed project or other action are estimated and assessed for impact on air quality. This can be done by comparison of the existing air quality or by how much these gases may add to the usual similar traffic of impacted areas.
Air quality	Assess how various pollutants may impact human health and the environment with regard to toxicology, public health, health sciences and epidemiology. Compare with regulatory standards for ambient conditions and emissions.
Water resources	Risks from diversion and contamination to controlled waters. Water bodies include surface water (such as streams, rivers, ponds, lakes, canals, drainage channels, coastal waters), groundwater and aquifers.
Biological resources	Displacement of terrestrial wildlife due to the removal of vegetation, and changes or elimination of habitats. Aquatic biota and migratory or other bird species are also addressed.
Transportation safety	Effect of changes in automobile and truck traffic on fatalities from accidents.
Traffic	Effect on the traffic and traffic control infrastructure as a result of changes to automobile and truck use in the local and regional area.

There are many differences between normal decommissioning and post-accident situations for each subject matter addressed in this publication. All three phases will include activities that are akin to new build projects as a result of the many structural, system and component modifications and additions needed to cope with the conditions. These activities may encompass the need for planning, engineering, design, procurement, fabrication, installation and proof testing. There are also significant differences and complications compared with typical nuclear facility decommissioning because of the more severe spread of radioactive contamination, the high intensity of direct radiation and other conditions encountered without precedent.

The importance of effective management of large amounts of radioactive waste produced during decommissioning activities should be emphasized. The strategy and comprehensive plans for the management of radioactive waste will need to be developed to cover a variety of relevant technical and non-technical issues. As advised for Fukushima Daiichi [41], an integrated plan for decommissioning and radioactive waste management needs to be prepared founded on minimizing impacts to human health and on the protection of the environment.

As Fukushima Daiichi decommissioning proceeds, the technological successes need to be documented as lessons learned. This might not be just a recording of factual data, but an evaluation and analysis that will be useful to serve future needs. These data should describe each challenge, the options considered and focus on the decisions made, outlining the pros and cons of the various options and including an evaluation of the errors and failures along the way to success. This was the approach taken for TMI-2 where the technical history has proven to be useful not only for work at Fukushima Daiichi, but also at Paks and will be, to some extent, useful for the Chernobyl NPP in the future.

The decision concerning the possible ultimate decommissioning options and end state for a nuclear facility after an accident will be very challenging and will need to be considered by all stakeholders involved. Technical considerations (pathway analyses and acceptance criteria, remediation criteria, intrusion prevention and long term monitoring) as well as non-technical aspects (availability of human resources or potential reuse and redevelopment of the site in the future) need to be taken into account.

An important issue that needs special attention is further management of damaged fuel and fuel debris after its retrieval from the reactor unit. There is some experience available on dissolution of fuel and possible separate long term storage and permanent disposal of fuel that will not be reprocessed.

Valuable and unique examples of R&D outcomes, such as innovative characterization techniques, special robots and other remote handling systems or advanced decontamination and dismantling procedures may give inspiration for nuclear decommissioning worldwide. Innovative characterization techniques have been used at Fukushima Daiichi, such as cosmic-ray muon radiography applied to locate the fuel and fuel debris in the damaged reactors [111, 112]. However, the effective use of adapted, commercially available technologies should also be considered in-line with the actual needs to be addressed during the implementation of post-accident decommissioning.

The last aspects to be mentioned are uncertainties and unknowns that may affect planned implementation of decommissioning of nuclear facilities after an accident. Uncertainties and unknowns are caused by the uncontrollable generation and spread of radioactive substances (often of unpredictable physical, chemical and radiological nature), and by the damage of SSCs as a result of the accidental release of high pressures and temperatures previously confined within safe barriers. The IAEA guidance on managing the unexpected in decommissioning is provided in Ref. [36].

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Annex I

CHERNOBYL NUCLEAR POWER PLANT TRANSFORMATION INTO AN ECOLOGICALLY SAFE SYSTEM

I–1. CHERNOBYL NUCLEAR POWER PLANT UNIT 4 (SHELTER OBJECT) TRANSFORMATION

On 26 April 1986, the worst accident in the history of nuclear energy occurred at the Chernobyl Nuclear Power Plant (NPP) Unit 4 in the then Soviet Union (see Fig. I–1). Today, this is in the territory of Ukraine. As a result of the accident, the reactor was completely destroyed, contaminating a vast area of about 200 000 km^2 and producing large amounts of high level radioactive waste on the premises of the destroyed unit and at the Chernobyl NPP site. Construction of the protective shelter above the damaged unit started in May 1986 under conditions of severe radiation levels.

By 30 November 1986, the State Commission accepted that Chernobyl NPP Unit 4 should be contained and maintained under a state of preservation under a structure that adopted a double name: shelter or sarcophagus (see Fig. I–2).

The shelter or sarcophagus contains the remnants of the destroyed Unit 4 as a result of the beyond design basis accident that resulted in the loss of all functional properties of the power unit. Priority measures were undertaken to reduce the accident's consequences and work was performed to ensure nuclear and radiation safety.

Therefore, within a very short time — just six months — the destroyed Unit 4 was transformed into the shelter object, a one of a kind structure. This allowed mitigation of the negative impact of the destroyed unit on the environment, personnel and the general public within a short time and, while unprecedented in its scale, the human-made source of the hazard was localized.

In parallel with the construction of the shelter, work of a large scope was performed to decontaminate the area around the shelter, the roofs of adjacent facilities and inside the Chernobyl NPP premises. This allowed the undamaged units to be put back into operational mode. Further actions towards shelter transformation were performed according to plan, with participation of the international community. Transformation of the shelter into an ecologically safe system was achieved through three basic implementation stages as approved (on 12 March 2001) by an interdepartmental commission for comprehensive solutions to Chernobyl NPP problems.

FIG. I–1. Chernobyl NPP Unit 4 destroyed as a result of the accident. (Courtesy of Chernobyl NPP, Ukraine.)

FIG. I–2. A general view of the shelter or sarcophagus. (Courtesy of Chernobyl NPP, Ukraine.)

I–2. STAGE 1: STABILIZATION

Measures were carried out during this stage (1998–2008) to improve durability, reliability and efficiency of existing and newly created structures and systems (construction, control, dust suppression, emergency) needed for the maintenance or improvement of the existing safety level at the facility.

The objective of stabilization was to reduce the risk of structural collapse of the shelter prior to construction of the new safety confinement (NSC) planned for completion in the longer term. Implementation of stabilization measures allowed improvement of the safety level of the shelter until 2023. In the future, the problem of unstable shelter structures will be resolved by dismantling or reinforcing the inside of the NSC.

I–3. STAGE 2: CREATION OF PROTECTIVE BARRIERS

The construction of the NSC (2008–2016) was to achieve the following objectives:

(a) Ensure personnel, public and environmental protection from nuclear and radiation hazards associated with the existing shelter;
(b) Create the necessary conditions for performing practical activities aimed at transforming the shelter into an ecologically safe system, including retrieval of residues of nuclear fuel and fuel containing material (FCM), radioactive waste management and dealing with unstable shelter constructions requiring dismantling or reinforcement.

The NSC is a protective structure that includes technological equipment for retrieval of fuel containing material from Unit 4 and equipment for radioactive waste management. Other systems are intended for transforming this unit into an ecologically safe system providing safety and environmental assurance for both operating personnel and the public.

The design, construction and commissioning of the NSC was performed by the International Consortium, NOVARKA, consisting of two French companies: VINCI Construction Grands Projects, and Bouygues Travaux Publics. Installation of NSC into the final position was completed in December 2016 as illustrated in Fig. I–3.

According to the design, the NSC includes the following:

FIG. I–3. NSC in the final position over the Unit 4. (Courtesy of Chernobyl NPP, Ukraine.)

— A main building, consisting of an arch structure that spans in the north–south direction 257.44 m, 108.39 m high, 150 m long. Additionally, there are foundations, eastern and western end walls, and necessary support systems.
— A technological building, including decontamination, fragmentation and packaging facilities, sanitary locks, workshops and other technological facilities.
— Auxiliary facilities.

An integrated management system was created for the NSC to ensure nuclear, radiation and industrial safety and its effective operation with a minimum quantity of operating personnel. It consists of the following systems:

(a) Radiation safety monitoring system.
(b) Seismicity monitoring system.
(c) Building construction condition monitoring system.
(d) Operations support systems: Ventilation system, water supply system, sewerage system (including liquid radioactive waste management), power supply system.
(e) Technological systems for radioactive waste and FCM management.

Fire safety and security systems are created and communication and television networks are provided. Crane equipment was installed to ensure dismantling of Unit 4 unstable structures. The main functions of the NSC are the following:

(1) Limit the radiation impact on the population, personnel and the environment within the established boundaries, both during normal operation of the shelter object and in the case of disturbances to normal operations (i.e. emergencies and accidents, including accidents during the dismantling of unstable structures, FCM and radioactive waste management operations);
(2) Limit the spread of ionizing radiation and radioactive substances present inside the shelter object;
(3) Provide technological support, including the creation of conditions for dismantling the unstable structures, the future retrieval of FCM and radioactive waste, accumulated water removal, and ensuring the implementation of control and maintenance measures in the shelter and at its industrial site;
(4) Monitoring of all shelter condition parameters and technological process controls;
(5) Physical protection, including the prevention of unauthorized access to FCM and radioactive waste and maintenance of safeguards for nuclear materials.

NSC construction is intended to allow the following to be achieved by providing an increase in the durability of the containment system:

(a) To improve the radiation safety level; the integrity of NSC construction restricts radiation impact on the population, personnel and the environment within the planned period of operation of 100 years.
(b) To reduce the probability of accidental collapse during the dismantling of unstable structures.
(c) To reduce the impact of emergency collapse from the load-bearing structures and monitoring systems inside the NSC.
(d) To improve the nuclear safety of the shelter by elimination of atmospheric moisture penetration into FCM, hence significantly reducing the risk of a self-sustaining chain reaction.

I–4. STAGE 3: TRANSFORMATION INTO THE ECOLOGICALLY SAFE SYSTEM

Within the third stage, according to the strategy, it is planned to remove FCM from the shelter, and to transform them into a controlled state by providing storage within the protective barriers and/or disposal in geological radioactive waste disposal facilities. All FCM will be sorted according to activity level, compacted and transferred in safe condition before storage (high level and nuclear hazardous radioactive material). FCM accounting should be in accordance with current legislation (see the tentative schedule in Fig. I–4).

After FCM retrieval from the shelter object, it will be possible to implement the final stage of the life cycle (i.e. decommissioning). At the decommissioning stage of the shelter, long term risks to people and to the environment will be eliminated. The decommissioning strategy will be chosen in accordance with the available technical and financial resources available for transformation into an ecologically safe end state.

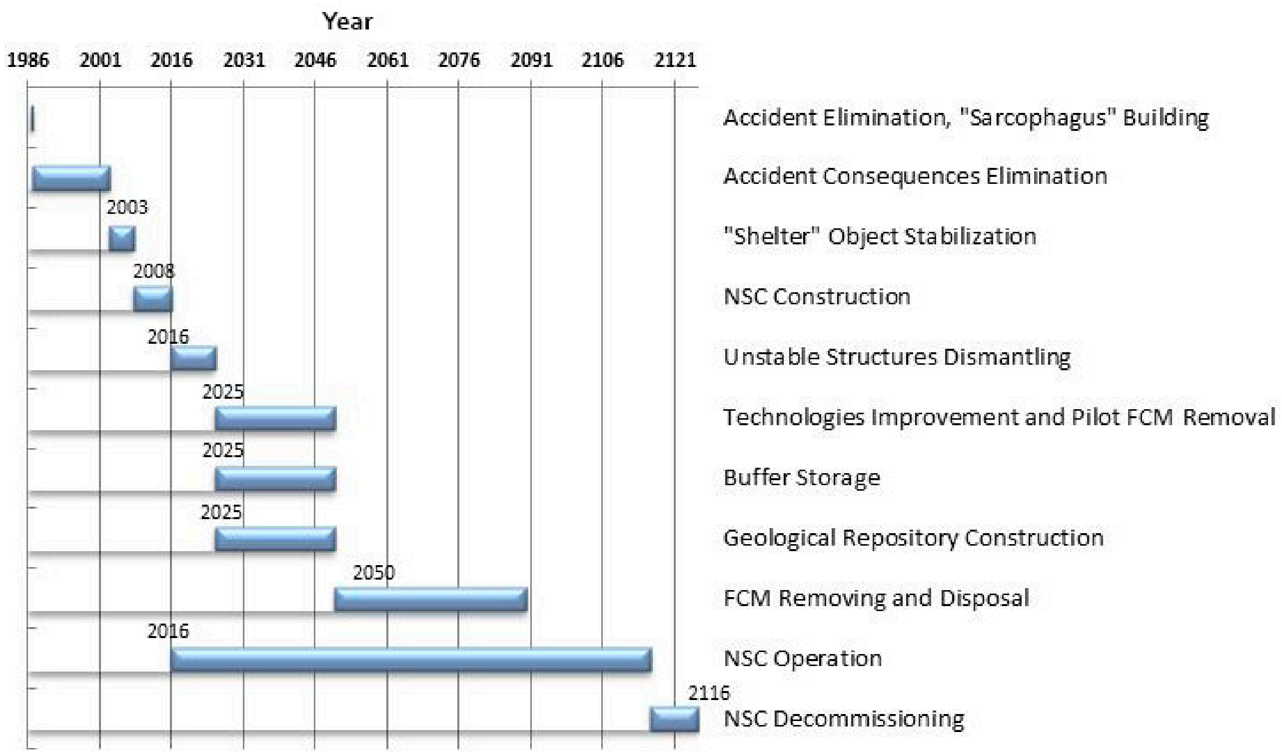

FIG. I–4. Tentative schedule of implementation strategies for shelter object transformations.

Annex II

THE UNITED STATES DEPARTMENT OF ENERGY'S ROLE AT TMI-2

Acknowledgement

The information in this annex was prepared by Willis W. Bixby who was the Manager of the Department of Energy's TMI-2 Site throughout the cleanup.

II–1. BACKGROUND FOR THE GOVERNMENT INVOLVEMENT AT TMI-2

Soon after the accident it was foreseen that TMl-2 would provide valuable information to validate the accident and core performance computer models used by the industry and the nuclear regulatory agency. The extent of damage to the reactor core and the subsequent release of fission products to the reactor containment and elsewhere in the plant was the most extensive experienced in any known light water reactor power system up until that time. Understanding the progression of events would provide a test of the models. However, as it turned out, the infusion of technical solutions proved to be extremely beneficial to the General Public Utilities Corporation (GPU), the owner/operator of TMI. This was especially important where GPU did not have the authority and technical resources for many of the investigations and developments that would be needed.

What follows in this annex are key institutional and policy lessons learned during the support activities of the United States Government for TMI-2 cleanup. Many of the technical experiences described in the main sections of this publication also relied on this involvement.

II–2. GOVERNMENT SUPPORT

Support from the highest levels of the government up to and including the White House (that is, the Office of the President) was essential to advancing the overall cost sharing agreement proposed by the Governor of Pennsylvania, which was the cornerstone for funding the TMl-2 cleanup. This support included recognition of the importance of the first of its kind information to be gained from the cleanup as well as an understanding of the causes and extent of the accident. The Government support advanced by the United States Department of Energy (USDOE) was reflected in specific Congressional budget justification documents throughout the ten year cleanup period. In addition, the United States Government support for taking ownership of the damaged fuel, debris and the wastes not routinely encountered during normal reactor operation, reduced the risk to the public had these materials been left on the site for an extended period without a disposal path. Sustaining support for the cleanup required continual education of both the legislative and executive branches of the Government throughout the cleanup. It also required clear financial accounting to ensure that the government was not 'bailing out' GPU but was, in fact, receiving value for the United States taxpayer. This required that the USDOE and its support contractor develop and negotiate scopes of work for the funding that was provided, via contract, to GPU. This was a key role for the USDOE on-site technical integration office.

II–3. USDOE ON-SITE R&D OFFICE: TECHNICAL INTEGRATION OFFICE

In addition to cost and schedule control for USDOE funded activities, effective integration of new technologies for the cleanup required very close coordination with the GPU. The on-site technical integration office allowed the USDOE and its contractor Edgerton, Germeshausen, and Grier, Inc, (EG&G), to interface with GPU and the Nuclear Regulatory Commission on a day to day basis. This allowed technical and regulatory issues and operating constraints to be addressed without long delays. It provided a central point to control and prioritize interactions with the various USDOE sites supplying technology and services to the cleanup while providing GPU with a local point of contact to access USDOE laboratories for selected technical issues. An on-site organization, with access to the USDOE R&D resources around the State, made USDOE and EG&G an integral part of the overall TMl-2 recovery team.

While the technical integration office had many functions, for the purpose of this publication, the following were of special importance:

(a) Assisted in the scheduling, monitoring and reporting of on-site R&D activities;
(b) Served as the focal point for systematic collection of R&D information;
(c) Provided the single point of contact for national laboratory information requests;
(d) Developed the fuel transportation system including cask procurement, equipment handling, and receipt and storage;
(e) Arranged for analysis and results reporting for samples;
(f) Provided support to GPU for any work performed at USDOE laboratories.

II–4. INTEGRATING USDOE OFF-SITE OPERATIONS WITH ON-SITE DECISION MAKING

Applying a systems approach to key cleanup challenges was important to reduce overall costs. Using TMI water samples very soon after the accident allowed Oak Ridge National Laboratory (ORNL) and Sandia National Laboratory (SNL) to develop a flowsheet to remove caesium and strontium in two processing systems rather than building multiple systems. This systems engineering approach to waste disposal helped drive operational decisions in the cleanup. For example, the curie loading on the submerged demineralizer system (SDS) vessels was increased based on (i) the ORNL and SNL analysis of the SDS flowsheet and the capacity of the resins to hold more radioactivity, and (ii) a recognition that the higher loaded resin material could be safely transported and accepted by USDOE for R&D. Similarly, after the double containment shipping requirement for the fuel debris was established, the site worked closely with SNL and the cask supplier on a test programme to support the structural analysis for the shipping cask. In addition, the decision to use railroads versus highway trucks to ship the core to Idaho was the result of a systems analysis.

II–5. APPLIED R&D

R&D was focused on securing the information needed to make informed engineering decisions for subsequent cleanup steps. While the ‘quick look’ and the lead screw analysis would contribute to a general understanding of core damage progression, they were also very important in understanding the constraints associated with removal of the reactor vessel head, upper plenum and fuel debris. The analysis allowed the engineers to make informed decisions about plenum damage and/or deformities that may have impacted plenum removal. The focus on getting to and removing the damaged core was the overriding driver for the schedule. Outputs from R&D developmental activities, such as the core boring machine, supported the recovery schedule. Robotic adaptation and development was supported if there was an immediate application, and it supported the TMl-2 recovery schedule.

To the extent possible, key decisions were based on information derived from visual observation or physical examination. This was particularly true when it came to the extent of damage in the core. Use of the core boring machine provided key information on the extent of relocation of the core material and helped guide subsequent core sample, characterization and removal efforts. The core bore machine also turned out to be a key method for breaking the core into pieces that could be manually removed from the defueling platform.

II–6. CONSTRAINED BUDGETS

Because the USDOE was limited in its appropriation by Congress and the White House, it was forced to be selective about tasks that it contracted for with GPU. This contractual approach required the USDOE (via its implementing contractor, EG&G) to agree upon tasks and negotiate costs and deliverables for the tasks. This required GPU to plan the major tasks so there was a clear understanding about the scope, schedule and cost to complete the selected activities. Because of the constrained funding, there were times when the USDOE would only fund a portion of a task with the remainder costs borne by GPU.

USDOE's focus was on gathering information to understand the accident's progression, which resulted in a constrained budget. Therefore, the majority of the USDOE's subsequent funding for the cleanup was focused on tasks to remove and dispose of the damaged core and could not contribute to general support and infrastructure projects.

II–7. STAKEHOLDER INTERACTIONS

Providing a framework for local communities to independently assess information being supplied by GPU and the government was important to re-establishing trust. After the early stages of the accident, a major effort was put forth by all parties engaged in the cleanup to communicate the progress and challenges at the site. Face to face engagements with the stakeholders helped gain support for key evolutions during the cleanup and helped re-establish trust. The site employed the latest social media tools available at the time of the accident: press releases, tours and public meetings. These were essential to increasing the transparency of activities.

Routine status reports were provided to public and local officials so they could independently understand the extent of progress. Allowing citizens to monitor potential off-site radiation releases as part of the Community Radiation Monitoring Program (CRMP) helped establish a sense of independence regarding GPU reported releases. Over time, as the results from the CRMP reaffirmed what GPU was reporting and, along with GPU cleanup performance, helped increase trust in the GPU.

The planned shipment of the damaged fuel from TMI to Idaho raised safety concerns in the local community around the plant and communities along the route. Meetings with stakeholders in communities along the rail route helped educate them on the integrity of the shipping packages and the depth of emergency response capability, should it be needed. After several safe shipments to Idaho, subsequent shipments were viewed as routine and not a cause for alarm.

II–8. HISTORICAL RECORD

A key element of the USDOE programme was the establishment of a databank to capture and document the decisions and progress of the cleanup. The information was formally documented by four organizations commonly referred to as the GEND group (GPU Nuclear Corporation, the Electric Power Research Institute, the Nuclear Regulatory Commission, and the United States Department of Energy) through reports, periodic updates, annual reports and supporting technical reports. In addition, towards the end of the cleanup, the Electric Power Research Institute funded a comprehensive technical history of the TMI cleanup[1] and three companion topical reports on waste management, data acquisition and analysis and decontamination. These reports cover the entire cleanup and document the challenges and considerations that led to key evolutions during the cleanup. Most importantly, these reports addressed what went wrong as well as successes.

II–9. WASTE MANAGEMENT

A major lesson learned at TMI was the need to develop a comprehensive waste management strategy as part of the need to plan other major evolutions. Commercial pathways existed for conventional low level waste as defined by NRC regulations, but the disposal of the fuel, core debris and waste materials posed institutional and technical challenges that needed to be resolved to allow the cleanup to proceed. Because TMl-2 was located on an island in the Susquehanna River, it did not have the space to accumulate large quantities of waste material. This necessitated off-site disposal of the material as it was generated at either commercial LLW disposal sites or government facilities for waste not routinely encountered in commercial operations.

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[1] ELECTRIC POWER RESEARCH INSTITUTE, The Cleanup at Three Mile Island Unit 2: A Technical History: 1979–1990, EPRI, Palo Alto, CA (1990).

The disposal and shipping approach was important as it provided the requirements for packaging the waste. For wastes and fuel debris that could not be disposed at commercial facilities, the availability of government managed alternatives was essential to the success of the cleanup.

II–10. INDEPENDENT EXTERNAL REVIEW

Independent advisory groups played key roles in shaping the direction throughout cleanup. In the early stages of the accident, the Industry Advisory Group came from industry and government organizations to help establish damaged core status, core cooling modes and approaches to bring the plant to safe shutdown. The Industry Advisory Group was a forerunner of the USDOE sponsored Technical Assistance Group, established to assist in the design of the SDS. The success of the Technical Assistance Group led to the subsequent creation of the Technical Assistance and Advisory Group (TAAG). This group was established to assist GPU and the USDOE in the ongoing review of defueling and data acquisition. The TAAG was chaired by a retired Director from the Bettis Laboratory, a USDOE National Laboratory that supported the development and advancement of government and commercial reactors.

The TAAG group included senior managers from the commercial entities that operated US Nuclear Navy shipyards, the architect–engineer for TMl-2, utility service organizations and national laboratories. In addition to their strong management background, the group brought experience in radiological controls, refuelling, water processing, radioactive operations and maintenance, and ‘one-of-a-kind’ practical problem solving. This group, with their diverse backgrounds and experience, helped provide the site team (GPU, the Nuclear Regulatory Commission and the United States Department of Energy) with alternative insights and approaches throughout the course of the cleanup. One of the most important examples of this was the quick look that emphasized gaining visual information on the actual damaged core as soon as possible to guide subsequent defueling evolutions.

Annex III

A1 NUCLEAR POWER PLANT DAMAGED FUEL AND RADIOACTIVE WASTE MANAGEMENT

III–1. INTRODUCTION

The A1 nuclear power plant (NPP) was the first NPP in the former Czechoslovak Socialist Republic. It was a gas cooled (CO_2) and heavy water moderated reactor (Fig. III−1), and operated from 25 December 1972 to 22 February 1977. The A1 NPP's average power output was approximately 150 MW(e). There were 148 fuel assemblies and 40 control rods in the core. The fuel assemblies had diameters of 102 mm or 114 mm and were made up of fuel elements of natural uranium with magnesium/beryllium cladding. The length of their active part (in the core) was 4 m. The reactor vessel has a diameter of 5.1 m and a height of 20.1 m.

The primary circuit consists of six loops, each with a turbo compressor and a steam generator. The moderator circuit consists of three loops, each with a circulation pump and a cooler. At the end of February 1977, an operational accident occurred. As a result of the accident, fuel cladding fractured and fuel damage occurred in the upper part of fuel elements ranging from 30 cm to 100 cm in length. The primary circuit (coolant) was contaminated by fission products. Some auxiliary circuits and facilities were also contaminated.

III–2. PHASE I: STABILIZATION AND CHARACTERIZATION

After reactor shutdown, the first task was to begin the cooling of the fuel elements in the reactor. This was carried out by coolant circulation in the primary circuit. During this period, particular attention was given to the control of the radiation situation, including monitoring of the contamination and the environment. Some fuel elements were unloaded from the reactor for visual inspection in a hot cell to obtain more detailed information about the extent of the cladding damage.

All fuel assemblies were removed later from the reactor into the facility for interim fuel storage, which is also located in the reactor hall. Fuel removal was carried out with a refuelling machine, used for fuel reloading during normal operation of the reactor. Spent fuel assemblies were placed into the carbon steel casings (penals) filled with cooling medium dowtherm (a mixture of diphenyl and diphenyloxide) or an aqueous solution of potassium chromate or bichromate (Chrompik) [III–1].

FIG. III−1. A1 NPP reactor hall. (Courtesy of JAVYS, a.s., Slovakia.)

III–3. PHASE II: CLEANUP OPERATIONS

After fuel removal, the coolant cleanup was carried out directly in the primary circuit by means of suitable filters. The heavy water (moderator) was discharged from the auxiliary moderator circuit into storage tanks. Total decontamination will be carried out later as a part of the decommissioning process.

Three basic methods were developed and used for decontamination — chemical decontamination, chemical decontamination combined with ultrasound, and electrochemical decontamination. Decontamination procedures based on the use of efficient foams have been developed for decontamination of large floors and wall surfaces in rooms. Gaseous wastes were treated as during normal operation. For example, air was collected by ventilation systems and cleaned up by means of special filter systems. The dominant radionuclides in the gaseous wastes were radioisotopes of the noble gases: ^{131}I, ^{141}Ce, ^{144}Ce, ^{103}Ru, ^{106}Ru, ^{140}Ba, ^{140}La, ^{134}Cs, ^{137}Cs, ^{95}Zr and ^{95}Nb.

Liquid radioactive waste, generated during the stabilization period, was processed as part of the waste management system. Liquid wastes were collected in two tanks, each with a capacity of 100 m^3. From there, they were pumped into a gravitation tank and flowed down into a mixer-settler. After the addition of necessary chemicals to initiate coagulation, the sludge was separated from the solution in a coagulation tank. The cleaned up water passed through a filter bed into an evaporator.

Concentrated liquid wastes were stored in a number of storage tanks. The spent beds of the sand filters and the ion exchange filters were also discharged into the storage tank. About 300 m^3 of radioactive liquid concentrate was produced by the treatment of primary liquid wastes during the stabilization period. The average volume activity of the concentrates was 1–10 GBq/m^3 with a salt content up to 10 kg/m^3 and pH levels from 8 to 9. The dominant radionuclides in the liquid wastes were ^{141}Ce, ^{103}Ru, ^{106}Ru, ^{140}La, ^{134}Cs, ^{137}Cs, ^{95}Zr, ^{95}Nb, ^{131}I and ^{60}Co. Transuranium elements were also found. Radioactive solid wastes were taken to the solid waste storage facility on the A1 NPP site. Soft wastes were pressed before storage [III–1].

III–4. PHASE III: CONDITIONING, STORAGE AND TRANSPORT OF DAMAGED FUEL

A classification for the stored fuel was prepared, before the fuel was transported to a reprocessing plant. The classification was based on the operational history of the fuel elements, a review of fuel elements in the hot cell and control measurements in the fuel assembly casings in spent fuel storage.

On the basis of above classification, the technology for treatment prior to transport was determined. The technological operation allowed the transfer of fuel assemblies from the storage casings into the special transport casings, before being sealed. The sealing had to be carried out to high standards to prevent leakage of fission products from the transported fuel element. Because of the requirement for hermetic sealing, it was necessary to remove moisture from the fuel elements to prevent hydrogen formation. For this reason, it was decided to transfer fuel assemblies from the dowtherm and chromate solutions into dry casings two months prior to hermetic sealing. After two months of dry storage, the formation of hydrogen decreased sufficiently. The fuel assemblies were then transferred into a sealed transport casing. A helium leak test was later carried out.

In total, 18 sealed casings with fuel assemblies were placed into a transport cask. When the transport cask was filled, it was placed into a special wagon (wagon container). The transport cask was then sealed and the sealing was checked by means of helium. It was then filled up to a pressure of 0.3 Pa with a gaseous mixture containing 4–4.8% of O_2, maximum of 2% of H_2 and the rest of N_2.

For cases with damaged fuel assemblies an alternative procedure was designed. The procedure was based on the following steps:

(a) Withdrawal of the casing together with the fuel assembly from the storage pool;
(b) Transfer of the casing with the fuel assembly to a drainage and cutting stand;
(c) Discharge of the chromate solution from the casing and shearing of the casing to the transport length;
(d) Transfer of the fragmented storage casing (containing the fuel assembly) to the sealed (transport) casing;
(e) Sealing of the transport casing and checking of the sealing;
(f) Transfer of the transport casing into a wagon container.

The fuel in the sealed casings was transported in special railway wagons and wagon containers, which met strict radioactive material transport requirements. Out of the total amount of 572 fuel assemblies, 440 fuel assemblies (stored in dowtherm) were retrieved and transported to the Russian Federation in the period 1983–1990. For retrieval and transport of the remaining 132 fuel assemblies (stored in Chrompik, the so-called 'non-manipulable fuel'), it was necessary to develop and manufacture special drainage and encapsulation technology and additional equipment including transport containers. The fuel was retrieved from the long term storage facility and transported by special containers to the Russian Federation. This was implemented in the period 1996–1999. By removing fuel from the plant, prerequisite conditions were established for further implementation of the A1 NPP decommissioning activities [III–2, III–3].

III–5. A1 NPP SPECIFIC LIQUID RADIOACTIVE WASTE

A Chrompik cooling medium was used during long term storage of spent fuel and later an organic coolant, dowtherm (organic liquid mixture of biphenyl and biphenyl oxide) replaced the Chrompik. The cooling media represented liquid radioactive waste with fission products (^{134}Cs, ^{137}Cs, ^{90}Sr) and certain amounts of alpha nuclides (^{238}Pu, ^{239}Pu, ^{241}Am and others). Their volumetrically concentrated beta activity was in the range of 4–6 MBq/L.

There was sludge in the Chrompik, which contained mainly corrosion products of the fuel element cladding, chromium III, uranium in the form of oxides, metallic uranium and corresponding amounts of plutonium (depending on the stage of fuel element burnup).

Part of the Chrompik with activity up to 1 GBq/L has been processed (in view of its radioactivity level and chemical composition) by vitrification (see model of A1 NPP vitrification facility in Fig. III–2), Chrompik with activity of 100 GBq/L will be processed in a modified vitrification line. Dowtherm with an activity up to 200 MBq/L was incinerated. If the activity was higher than this, it was decontaminated by adsorption resins and then incinerated. Spent adsorbents were solidified into geopolymer and incorporated into the containers.

A certain volume of dowtherm and sludge remains in casks that contained spent fuel. After retrieval of residues from the casks, it will be characterized and a suitable matrix for the conditioning of these residues will be determined. Several matrices are being assessed for sludge solidification such as cement and geopolymer.

Treatment of contaminated storage casks follows after the retrieval of stored media. Altogether, 350 casks plus 100 pieces of the upper parts of casks are to be characterized, remotely cut into pieces of length of about 350 mm, transported to the decontamination unit and decontaminated in an ultrasonic bath. Decontaminated fragments will be placed in 200 L drums and measured on gamma and alpha scanners. The volume of drums will be reduced by using a high force compactor and the product will be placed into a fibre-concrete container to be disposed in the national repository for radioactive waste.

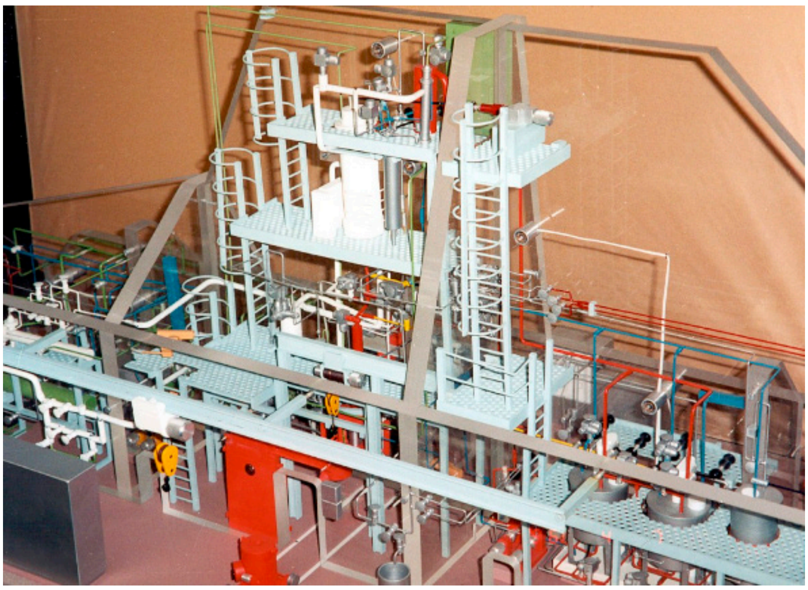

FIG. III–2. A1 NPP vitrification facility. (Courtesy of JAVYS, a.s., Slovakia.)

III–6. SECONDARY RADIOACTIVE WASTE FROM DECONTAMINATION

The substantially higher concentrations of active components in decontamination solutions, and high content of iron in solutions after decontamination, were extraordinarily demanding when treating liquid wastes from decontamination. Used decontamination solutions were collected separately from other liquid wastes. Two procedures had been tested for fixation of used decontamination solutions — cementation and fixation into a geopolymer matrix. Special procedures for solidification have been developed and approved.

Spent decontamination solutions are neutralized in drums. Sediment is precipitated during the adjustment of pH, settled, and purified water is repumped to a special collection system (after the accomplishment of limits on pH, activity, surfactants, salinity). This water, together with other waste waters, is treated by evaporation. A significant amount of water is then released into the environment after it has been purified. Secondary condensate is used for the preparation of cement grout for grouting of solid or solidified radioactive waste in fibre reinforced concrete containers.

The separated sludge (content of dry residue circa 10–15% weight) is treated by in-drum cementation or fixed into a geopolymer matrix. The cemented drums are placed in containers and, after grouting with cement, the final product is disposed of in the national repository for radioactive waste in Mochovce (see Fig. III–3).

Special liquids, such as sludges from the secondary waste, will be treated either by the in-drum cementation used for the sludges from external storage tanks or by the in-drum solidification into a geopolymer matrix.

III–7. SOLID RADIOACTIVE WASTE

The decommissioning of nuclear power plants that were shutdown as a result of an operating accident often generates large amounts of solid waste. Solid waste can include metallic waste from equipment, concrete, decontamination agents and protective clothing.

The major challenges in managing solid radioactive waste are not only the volume and their heterogeneity, but in the case of A1 NPP, the fact that certain parts of solid waste contain alpha nuclides [III–1]. In addition to the utilization of standard methods for the treatment of solid waste (for example, combustion and pressing), considerable interest is being shown in using melting as a method for metallic radioactive waste treatment resulting in volume reduction and a waste form that is more suitable for long term storage and disposal.

An assessment of alternatives in the treatment of metallic waste can be found in the statement of the environmental impact assessment study on a facility for the melting of metallic radioactive waste [III–4]. The study takes into account the reclassification of the radioactive waste in Slovakia and the establishment of the category of very LLW with the planned construction of a very LLW repository. This study has analysed the feasibility

FIG. III–3. National repository of radioactive waste in Mochovce. (Courtesy of JAVYS, a.s., Slovakia.)

for development and construction of a facility on the Bohunice site aimed at melting low contaminated metals from the decommissioning of nuclear facilities in Slovakia. The new melting facility has been built and is to be commissioned in 2019.

REFERENCES TO ANNEX III

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Annex IV

CONSIDERATIONS ON POST-ACCIDENT USE OF THE INTERNATIONAL STRUCTURE FOR DECOMMISSIONING COSTING

The purpose of the international structure for decommissioning costing (ISDC) is to facilitate communication, to promote uniformity and to provide a common platform in presenting the decommissioning costs [IV–1]. Clear definitions of ISDC items support the common understanding of cost items (i.e. what is behind the cost). Experience shows that implementation of ISDC is effective in comparing the costs for decommissioning of various nuclear power plants (NPPs) even when comparing the costs developed originally in different cost structures and under different assumptions and boundary conditions for decommissioning.

ISDC is the list of typical decommissioning activities which may be identified in any decommissioning project, any nuclear facility and any condition for decommissioning, including the facilities decommissioned after nuclear accidents. ISDC decommissioning activities are organized in a hierarchical structure with the first and second levels being aggregations of typical activities identified at the third level [IV–1].

At Level 1, the following eleven principal activities are identified:

01 – Pre-decommissioning actions;
02 – Facility shutdown activities;
03 – Additional activities for safe enclosure or entombment;
04 – Dismantling activities within the controlled area;
05 – Waste processing, storage and disposal;
06 – Site infrastructure and operation;
07 – Conventional dismantling, demolition and site restoration;
08 – Project management, engineering and support;
09 – Research and development;
10 – Fuel and nuclear material;
11 – Miscellaneous expenditures.

Activities at Level 2 represent a subdivision of the Level 1 activities; grouping of activities at Level 2 is based on the similar nature of the grouped activities. Activities at Level 3 provide a further subdivision of activities to the level of typical decommissioning activities, which may be identified in any decommissioning project. Level 3 is the reference level linked to the cost items developed in cost structures other than ISDC. Level 3 activities represent the basic building blocks for developing the overall cost estimate.

Cost estimators may add additional hierarchical levels to the cost structure in order to distinguish costs relating to specific parts of the plant or to specific systems, or in order to distinguish costs according to specific phases of a decommissioning project. ISDC is the structure open to cost estimators down to the third level. The third level is the reference level for presenting decommissioning cost data.

ISDC gives a list of typical decommissioning activities, which may be identified in any decommissioning project, and also offers a new costing approach that is based on using the ISDC as the cost calculation structure. Specific cost structures may be developed for various types of nuclear installations.

The ISDC costing methodology has specific features allowing the calculation of costs in important cost categories (e.g. labour cost, investment cost, expenses and contingency). An important advantage of the ISDC is the possibility of its use as a checklist for activities in any decommissioning project.

The basic ISDC cost data format is presented in Ref. [IV–1] as a matrix of vertically numbered ISDC items (typical decommissioning activities) and, horizontally listed cost categories. The matrix provides also the platform for subsequent cost data processing.

Considering the main features of the ISDC listed above, the ISDC may be also used effectively in evaluating the cost for the decommissioning of nuclear installations after accidents. The following section presents the generic definitions of ISDC items at Level 1, and the assumed impact of accidents on the ISDC principal activities.

IV–1. GENERAL OUTLINE OF IMPACT OF ACCIDENTS ON DECOMMISSIONING COSTING ACCORDING TO THE ISDC PRINCIPAL ACTIVITIES

This section presents the basic characteristics of ISDC principal activities 01–11 (ISDC Level 1) as they are defined in Ref. [IV–1], with an assumed impact of an accident on the ISDC principal activities and the consequences for costing.

In general, the accidents in nuclear installations may have the following impacts, which may affect decommissioning costs:

(a) Impact on the radiological situation within the installation: Dose rates, level of contamination, specific composition of nuclide vectors.
(b) Non-standard and specific types and quantities of radioactive waste generated during the accident and during post-accident clean out.
(c) Impact on the environment.
(d) Impact on safety in systems and structures.
(e) Damaged fuel and fuel debris management.
(f) Physical changes of structures affected or damaged because of the accident.
(g) Specific procedures/equipment for characterization, decontamination, dismantling and other typical decommissioning activities.
(h) Decommissioning of post-accident emergency/safety systems and structure (new temporary structures, massive shieldings, new roads).
(i) Clean out activities/projects.
(j) Extended preparation, management, support, R&D activities, procurement and other activities.

IV–1.1. ISDC 01: Pre-decommissioning actions

Principal activity, ISDC 01, concerns the activities that are needed prior to licensing (approval) of a decommissioning project, including contracting activities if the general contractor or a multicontractor model is implemented. The activities are graded, starting from the very preliminary costing feasibility studies (in some countries at the commissioning of the facility) up to the level of detailed decommissioning documentation for licensing and planning. Most of these activities are specific engineering, characterization, planning and management activities, performed by the owner’s personnel and by contracting to companies specialized in the preparation of decommissioning documentation.

The assumed impacts of accidents on ISDC 01 activities are as follows:

(a) Extended planning due to additional decommissioning activities, phases, duration, more complex decommissioning schedule;
(b) Extended facility characterization due to the non-standard situation in the facility and environment;
(c) Extended safety assessment due to a specific safety situation;
(d) Extended waste management planning due to additional types and quantities of radioactive waste;
(e) Extended authorization of decommissioning and related extended management group.

The planning of decommissioning after an accident requires a greater workforce, additional documents, new specific procedures, additional instrumentation and equipment, and specific licensing. Decommissioning costs may be increased approximately by factors of 2–5, or even more, depending on the accident. The following is an overview of possible impacts on specific ISDC costing categories:

(a) 01.0100 Decommissioning planning: Development of a specific decommissioning strategy. Planning should reflect the actual non-standard situation in the installation. Planning of decommissioning activities is more extensive. Planning may be done in several phases according to the available information on the installation.

(b) 01.0200 Facility characterization: Several thoroughly prepared characterization campaigns are needed in addition to specific instrumentation and procedures, quantity and types of samples, methods for the definition of nuclide vectors, and an extended facility inventory database is needed.
(c) 01.0300 Safety, security and environmental studies: The impact of the accident may have had a serious impact on the safety of systems, structures, the site and the environment. Much more safety assessment is needed in order to evaluate the actual situation in systems and structures based on data from the characterization to predict the evolution of safety, to propose immediate and long term measures to ensure safety and to propose modified decommissioning activities to maintain safety during their implementation.
(d) 01.0400 Waste management planning: Planning should respect additional non-standard types and quantities of radioactive waste generated during the accident and during the post-accident clean out. Decommissioning waste will also have additional features in comparison with standard decommissioning waste. Additional waste management techniques and waste characterization will be needed.
(e) 01.0500 Authorization: Licensing procedures and licensing documents will be extended. Licensing may be performed in several phases in relation to the planned decommissioning phases, which should be licensed individually.
(f) 01.0600 Preparing management group and contracting: The size of the group and extent of duties will be broader in comparison with standard decommissioning.

IV−1.2. ISDC 02: Facility shutdown activities

ISDC 02 concerns activities during the transition period after the shutdown and until the licence for decommissioning is obtained. The main purpose of these activities is to prepare the facility for decommissioning, using experienced operations personnel and, in specific cases, specialized services. The standard situation, at the start of decommissioning, is that there are no historical/legacy wastes (including operational wastes), the systems are without any operations fluids and the primary systems are decontaminated using the operational procedures (which may be modified for harder decontamination) and existing personnel. This situation is hard to achieve in old facilities or facilities after accidents. In these cases, the assumptions and boundary conditions for a decommissioning project should define the starting position of the decommissioning project in relation to ISDC 02 activities.

The removal of all operational waste is included in ISDC 02; using existing operational procedures and personnel. Further treatment, conditioning, transport and disposal are included in ISDC 05. The retrieval of historical/legacy waste is also included in ISDC 05.

The assumed impacts of accidents on ISDC 02 activities are as follows:

(a) The facility shutdown period may be more complex and longer by a factor of approximately 2–4 with many specific additional activities especially related to ensuring safety. Activities related to management of spent fuel may play an important role.
(b) Accidents with an impact on nuclear fuel will have a serious impact on spent/damaged fuel management.
(c) Additional and/or redundant specific activities and equipment are needed to maintain safety.
(d) The generation of large quantities of operational waste with specific radionuclide compositions; in the case of accidents with an impact on fuel, alpha radionuclides are also present.

Facility shutdown may be normally funded from the operator’s fund and/or from the decommissioning fund. The cost in the case of standard decommissioning may represent approximately 0–15% of the total decommissioning costs, depending on national legislation. Accidents affect the structure and extent of facility shutdown activities; the cost may be a large percent of the total decommissioning cost. ISDC 02 is one of the most affected ISDC principal activities because of accidents.

Accidents affecting the fuel in the reactor core and/or consequently in the cooling pools in reactor buildings may have an especially serious impact on the decommissioning costs.

A long term shutdown period for the facility (decades) may be needed to implement post-accident clean out in order to establish the safe status of the facility before the start of decommissioning. This increases decommissioning costs. The following gives an overview of the possible impacts on specific ISDC costing categories:

(a) 02.0100 Plant shutdown and inspection: Facility shutdown and inspection activities are more complex. Spent fuel management may require specific ad hoc emergency solutions and consequent solutions for stabilization of the situation; the same for waste management. Safety is the priority that may also require parallel redundant solutions. Stabilization of the site, the affected on-site and off-site environment and the underground waters requires additional efforts.
(b) 02.0200 Drainage and drying of systems: Additional activities/instrumentation are expected in accident cases.
(c) 02.0300 Decontamination of closed systems for dose reduction: New specific procedures and equipment are needed to solve the situation.
(d) 02.0400 Radiological inventory characterization to support detailed planning: Additional characterization campaigns and instrumentation are needed to assess the situation after the accident, during the performance of emergency solutions and after realization of safety measures in order to maintain safety and to prepare actual data for decommissioning planning and for safety assessment.
(e) 02.0500 Removal of system fluids, operational waste and redundant materials: Removal of system fluids and operational waste is more complex due to the modification of the system fluids and the increased quantity and modified properties of operational waste, especially from emergency activities. Owing to the long duration of shutdown periods, some types of waste change their physical/chemical properties and become historical waste with additional consequences for management.

IV−1.3. ISDC 03: Additional activities for safe enclosure or entombment

ISDC 03 concerns the activities that are implemented in the decommissioning scenarios with deferred dismantling. These activities are needed for preparation of the safe enclosure for the facility in order to ensure the long term stability and safety during the period of safe enclosure. This principal activity is not implemented in decommissioning scenarios with immediate dismantling. If there is partial decommissioning of selected systems and buildings during the phase of preparation of safe enclosure, the activities are allocated to ISDC 04–11.

In the specific decommissioning scenarios of entombment, ISDC 03 also involves the activities for achieving the final state of entombment. Other activities before achieving the final state are allocated to ISDC 04–11.

Assumed impacts of accidents on activities of ISDC 03 are as follows:

(a) Complex radiological situation in systems and on premises;
(b) Additional radioactive waste and materials present on premises;
(c) Affected site and environment.

Preparation for safe enclosure normally expects systems and structures to be as they are after normal shutdown activities (i.e. systems are empty, dried, primary circuit decontaminated and the site is clean). The situation may be significantly different after an accident. If safe enclosure is the chosen option for the decommissioning strategy, many additional activities will need to be performed in comparison with installation shutdown after normal operations. The following gives an overview of the possible impact on specific costing categories:

(a) 03.0100 Preparation for safe enclosure: Additional activities are needed to prepare the systems and premises within the safe enclosure due to the complex radiological situation, additional radioactive waste and materials generated during clean out activities.
(b) 03.0200 Site boundary reconfiguration, isolating and securing structures: Additional activities are needed due to the contaminated areas of the site, contaminated underground water and affected areas outside of the site boundaries.
(c) 03.0300 Facility entombment: Not relevant.

IV–1.4. ISDC 04: Dismantling activities within the controlled area

ISDC 04 includes the activities for removing the contaminated and activated systems and structures from the controlled area and identified contaminated items at the site which are outside of the controlled area. Prior to dismantling, there are procurement activities, preparation activities and pre-dismantling decontamination activities for ensuring safe dismantling. Dismantling is organized according the types of facilities and according to the main components and materials to be removed. Removal of contamination includes also decontamination of building surfaces, removal of embedded elements within the premises and removal of contaminated systems and structures (such as the underground pipes) and soils outside of premises of the facility. No waste management is included in ISDC 04. At the end the final radioactivity survey, buildings are released. The buildings are ready for conventional demolition in ISDC 07, if this is part of the decommissioning strategy.

The assumed impacts of accidents on activities of ISDC 04 are as follows:

(a) The controlled zone may be significantly extended in comparison with the previous controlled area before the nuclear accident;
(b) Complex radiological situations in systems and premises may arise;
(c) Additional radioactive waste and materials present in premises;
(d) Site and environment affected; additional contaminated systems and structures.

For standard dismantling activities, as defined in ISDC 04, the systems, structures and the site are in the conditions described in ISDC 03 (i.e. residual contamination and activation at the end of operation in systems and structures and a clean site). After accidents, the systems and structures remain affected, more remote-controlled operations are needed; more parallel supporting activities such as preparation, characterization and finishing activities are required. Instrumentation for decommissioning activities is more complex. The following is an overview of possible impacts on specific costing categories:

(a) 04.0100 Procurement of equipment for decontamination and dismantling: Additional equipment and instrumentation is required. Some of this may be very specific. Installation, testing, licensing is more complex. Modification/adaptation of standard instrumentation may be needed.
(b) 04.0200 Preparations and support for dismantling: Additional activities and temporary systems are needed. More complex ongoing material characterization and additional instrumentation is needed during decontamination and dismantling activities.
(c) 04.0300 Pre-dismantling decontamination: Additional systems and new specific procedures are required. The extent of decontamination may be much higher in systems and also within the premises. The handling of larger quantities of waste generated is required.
(d) 04.0400 Removal of materials requiring specific procedures: Additional activities and instrumentation may be needed due to additional contamination of these materials, which are normally clean, or only partially and slightly contaminated.
(e) 04.0500 Dismantling of main process structures, systems and components: More remote-controlled activities, additional specific preparation and finishing activities, and handling of complex radioactive waste generated prior to processing.
(f) 04.0600 Dismantling of other systems and components: More remote-controlled activities; ongoing characterization, additional preparation and finishing activities; and handling of complex radioactive waste generated prior to processing.
(g) 04.0700 Removal of contamination from building structures: Structures affected by accidents require more effective decontamination techniques, the extent of activities is higher, including additional handling of waste generated prior to processing.
(h) 04.0800 Removal of contamination from areas outside buildings: Additional activities are needed owing to affected areas on-site and affected underground pipes and structure, which are normally clean.
(i) 04.0900 Final radioactivity survey for release of buildings: The survey of buildings is more complex due to increased contamination of the buildings.

IV–1.5. ISDC 05: Waste processing, storage and disposal

ISDC 05 includes all the activities for management of historical/legacy waste and for decommissioning waste generated in activities of ISDC 04 (primary and secondary radioactive waste) and conventional and hazardous waste generated in ISDC 07. At the beginning of the ISDC, there are activities for establishing operational support and decommissioning of the waste management system operated within the decommissioning project. The assumptions and boundary conditions for the decommissioning project should define the extent and types of waste to be handled including the management of the operational waste from ISDC 02, if this is the case. Management of waste types, not covered by the waste management system operated within the decommissioning project, are considered as external services. Waste management systems shared with other decommissioning projects are also an option.

Management of waste in ISDC 05 is organized at the second level according to type of waste as defined in the latest IAEA waste classification [IV–2]. Characterization of any kind, waste retrieval (if this is the case) and processing, final conditioning, storing, transport, disposal and containers for each type of waste are defined at the third level. The end state of all types of waste is the disposal at repositories for radioactive waste for relevant types of waste, repositories for hazardous waste, repositories for conventional waste and free release or conditional release of reusable materials. Specific treatment activities (sorting, fragmentation, decontamination, super compaction, incineration, conditioning, any characterization methods) are identified at the fourth level; the extent and additional numbering is open for users.

The assumed impacts of accidents on ISDC 05 activities are as follows:

(a) Additional, specific historical/legacy radioactive waste generated during the emergency and post-accident clean out activities, which remain on the site;
(b) Historical/legacy wastes that are stored under specific conditions; retrieval may be very complex after longer storage periods;
(c) Additional decommissioning radioactive waste;
(d) More complex waste management techniques required including storage and disposal.

In standard dismantling activities as defined in ISDC 04, the volumes, types and radiological properties of waste are well predicted. In the case of accidents, the volumes, types and radiological properties of waste are very case specific. A waste inventory should be identified after the implementation of post-accident cleanup activities.

Additional historical/legacy waste remaining on the site after finishing emergency and post-accident cleanup activities may change its physical or chemical properties after longer storage periods. As a consequence, additional specific characterization is needed before retrieval of this waste. Complex techniques are needed for retrieval, mostly remote-controlled. Design, construction, testing, licensing and related safety assessment require large efforts.

The costs for management of historical/legacy waste may be equivalent or may exceed the costs for management of decommissioning wastes. Management of historical/legacy waste may require additional waste management techniques. Management of decommissioning waste may be more complex. The following gives an overview of possible impacts on specific costing categories:

(a) 05.0100 Waste management system: Additional waste management techniques, instrumentation, temporary premises and storage capacities are needed; operational support for the extended waste management system is more complex.
(b) 05.0200–05.0600 Management of historical/legacy waste: Additional complex characterization required before retrieval and during all steps of further processing of waste; additional, mostly remote-controlled, techniques for retrieval of waste; additional specific waste management techniques; long term waste storage may be required.
(c) 05.0700–05.1200 Management of decommissioning waste: Additional volumes of decommissioning waste generated due to affected systems and structures.
(d) 05.1300 Management of decommissioning waste generated outside of the controlled area: Additional volumes of waste due to additional activities related to decommissioning activities.

IV−1.6. ISDC 06: Site infrastructure and operation

ISDC 06 concerns the activities for site security and surveillance, site operation and maintenance, operation of support systems and radiation and environmental safety monitoring. All these are the activities for ensuring safety at the site and operability of auxiliary systems needed for supporting the decommissioning activities.

The demand for these activities may be very different for individual phases of the project especially in the case of deferred dismantling. During the main decommissioning phases, the requirements are reduced as decommissioning proceeds. Proper adjustment of these activities is important.

The assumed impacts of accidents on ISDC 06 activities are as follows:

(a) Additional specific systems for security and surveillance;
(b) Extended site operation and maintenance;
(c) Extended operation of support systems;
(d) Extended operation of radiation and environmental safety monitoring.

Standard ISDC 06 activities are normally the reduced and/or modified activities from the operational period. In accident cases, additional activities are needed in all groups of activities at ISDC Level 2 in comparison with standard decommissioning. This results in additional costs for additional personnel, equipment and the operation of systems.

The duration of dismantling activities in a decommissioning project is prolonged in accident cases. This results in additional costs due to the extended duration because costs in ISDC 06 are mostly period dependent. The duration of shutdown activities (the transition period) may be significantly prolonged in accident cases. If these activities are funded from the decommissioning fund, the costs may significantly increase. The following gives an overview of possible impacts on specific costing categories:

(a) 06.0100 Site security and surveillance: Extended personnel and instrumentation, due to higher risks on the site.
(b) 06.0200 Site operation and maintenance: Additional activities and instrumentation due to additional barriers in systems and structures.
(c) 06.0300 Operation of support systems: Additional auxiliary systems (some of them may be redundant due to higher risks on the site).
(d) 06.0400 Radiation and environmental safety monitoring: Extended systems due to a higher radiological inventory, on-site risks and risks of affecting the environment.

IV−1.7. ISDC 07: Conventional dismantling, demolition and site restoration

ISDC 07 concerns conventional dismantling of systems in premises outside of the controlled area and demolition of structures, both for buildings originally located within the controlled area (after their declassification in ISDC 04) and for buildings outside the controlled area according the scope of the decommissioning project (level of demolition may be different). Some of the buildings can be refurbished for their further use or can be considered as the site assets of a decommissioning project. Activities include also the site cleanup, landscaping and the final survey of the site. Management of conventional and hazardous waste from dismantling and demolition is included in ISDC 05.

In some cases, the site is released with defined restrictions that may require additional costs for the period of restricted use of the site or its parts. The activities in ISDC 07, especially the conventional demolition, can include costly items, so a clear definition of the end state of buildings and site is required for assumptions and boundary conditions for the decommissioning project. As an example, the end state can differ, such as no demolition of buildings, demolition to the level of one meter deep or complete demolition of concrete structures to the base plate levels.

The assumed impacts of accidents on ISDC 07 activities are as follows:

(a) Additional inventory from previous measures and activities during the decommissioning;
(b) Additional activities related to the level of demolition;
(c) More complex procedures for site release.

In standard decommissioning, ISDC 07 activities refer to clean buildings and site. Also, site release means the implementation of standard procedures. In accident cases, the demolition strategy may be changed. As an example, some buildings should be demolished to the ground plate to be sure that there is no residual contamination. There may some parts of the site affected during activities for removal of contamination from site or underground pipes and structures. The following gives an overview of possible impacts on specific costing categories:

(a) 07.0100 Procurement of equipment for conventional dismantling and demolition: Not significantly affected.
(b) 07.0200 Dismantling of systems and building components outside of controlled area: Not significantly affected.
(c) 07.0300 Demolition of buildings and structures: May be affected significantly in a case where the demolition level is changed to the level of the ground plate.
(d) 07.0400 Final cleanup and landscaping: May be affected in a case where additional activities were performed during the post-accident period.
(e) 07.0500 Final radioactivity survey of site: Extended activities and procedures for site release due to increased radiological inventory on-site as the consequence of accidents.
(f) 07.0600 Perpetuity funding/surveillance for limited or restricted release of property: This ISDC item is normally not relevant in standard decommissioning projects; however, it may be the subject of the specific decommissioning project; additional costs are considered in these cases.

IV−1.8. ISDC 08: − Project management, engineering and support

ISDC 08 includes all types of activities for the management of decommissioning activities, engineering, technical, safety and other relevant support, during all phases of the decommissioning project. Support activities prior to the start of decommissioning activities are included, such as mobilization of personnel and establishment of the infrastructure for decommissioning and subsequent demobilization activities after completion of the main decommissioning activities.

Where a prime contractor is appointed to oversee the overall project, or where contractors perform selected decommissioning activities, the cost of certain activities can be differentiated between owner costs and contractor costs. The conditions for performing those activities can be different on the sides of the owner and contractor. These activities should be evaluated separately. For these purposes, two identical segments, for the licensee and contractors are available in ISDC 08.

Another specific aspect for activities is the grading of activities in ISDC 08 according to the phases of the decommissioning project and also within individual phases. Proper adjustment of ISDC 08 activities for individual phases of a decommissioning project is important.

The assumed impacts of accidents on ISDC 08 activities are as follows:

(a) Extended scope of the decommissioning project;
(b) Additional types of decommissioning activities;
(c) Specific external services;
(d) Specific contractor(s) scheme.

Standard ISDC 08 activities are normally the transformation of activities from the operational period. In accident cases, additional activities are needed in all groups of activities at ISDC Level 2 in comparison with standard decommissioning. This results in additional costs mostly for additional personnel and external services. The duration of dismantling activities in a decommissioning project is prolonged in accident cases. This results in additional costs due to the extended duration because cost types in ISDC 08 are mostly period dependent. The duration of shutdown activities (the transition period) may be significantly prolonged in an accident case. If these activities are funded from a decommissioning fund, costs may be significantly increased. The following gives an overview of possible impacts on specific costing categories:

(a) 08.0100, 08.0600 Mobilization and preparatory work: Extended activities, due to the extended scope of the project.
(b) 08.0200, 08.0700 Project management: Extended scope of the project, additional specific activities to be managed, managing of other types of risks in the decommissioning process, and need for additional external services and consultancy.
(c) 08.0300, 08.0800 Support services: Support for extended decommissioning activities; specific training; support for specific decommissioning activities.
(d) 08.0400, 08.0800 Health and safety: Extended activities and instrumentation due to additional types of risks in decommissioning.
(e) 08.0500, 08.0900 Demobilization: Extended activities, due to the extended scope of the project.

IV−1.9. ISDC 09: Research and development

ISDC 09 concerns all activities with the character of research and development, specific for the decommissioning projects, where the information due to background of the project is not sufficient or not available at the time. Normally, research and development is contracted to specialized institutions and companies on national and/or international levels. Simulation of complicated work on models may be performed by the owner's personnel or contracted to specialized institutions and companies.

The assumed impacts of accidents on ISDC 09 activities are as follows:

(a) Limited instrumentation and/or equipment for case specific decommissioning activities available on the market;
(b) Limited knowledge of case specific decommissioning procedures.

ISDC 09 activities are needed normally to a very limited extent in standard decommissioning, where enough knowledge has already been accumulated. In decommissioning projects caused by accidents, additional instrumentation and procedures are needed to meet the specific conditions of selected individual decommissioning activities. Due to increased risks, the preparation of decommissioning activities may include their extensive simulation prior to implementation.

The following gives an overview of possible impacts on specific costing categories:

(a) 09.0100 Research and development of equipment, techniques and procedures: Requirements for R&D for specific instrumentation for characterization, decontamination, dismantling, waste management and other R&D activities.
(b) 09.0200 Simulation of complicated works: Requirements for physical mock-ups and training, test or demonstration programmes, computer simulations, visualizations and 3-D modelling and other activities.

IV−1.10. ISDC 10: Fuel and nuclear material

ISDC 10 concerns the activities defined within the decommissioning project for spent fuel and for nuclear materials after defueling reactors (ISDC 02). The extent of activities may be simple for a NPP after a standard shutdown — the spent fuel is transported from the cooling system in the reactor building into the external storage facility for long term storage of the spent fuel. It is assumed that a long term spent fuel store is available. Decommissioning of large spent fuel storage facilities, such as the stores for NPPs, is normally organized as a separate decommissioning project.

The situation may be different for NPPs or research reactors where the external storage facility is not available owing to the type of the spent fuel, damage of the spent fuel or other reasons. In these cases, the buffer storage for spent fuel and/or nuclear materials (fuel debris) should be considered within the decommissioning project (construction, licensing, operation, decommissioning) as well as the transfer of spent fuel and/or nuclear materials away from this buffer storage. Special programmes for the repatriation of highly enriched fuel from research reactors may be another example.

The assumed impact of accidents on ISDC 10 activities includes damaged spent fuel and fuel debris requiring specific actions (characterization, retrieval, handling, transport, storage).

In standard decommissioning, the ISDC 10 activities are implemented normally to a very limited extent for transportation of the spent fuel out of the reactor building to the interim spent fuel storage. In the case of damaged spent fuel and fuel debris, additional activities, equipment, instrumentation and premises will be needed to condition the spent fuel to meet the criteria for storage in standard interim spent fuel storages. This additional equipment and instrumentation may be placed in the reactor building or in external specific buildings, especially in case of research reactors. In critical cases, additional solutions should also be developed. The removal of the spent fuel from the facility to be decommissioned is one of the priorities to enable the decommissioning.

The following gives an overview of possible impacts on specific costing categories:

(a) 10.0100 Removal of fuel or nuclear materials from the facility to be decommissioned: Requirements for dedicated equipment and instrumentation for preparation of the damaged spent fuel to external storage or to the dedicated buffer storage.
(b) 10.0200 Dedicated buffer storage for fuel and/or nuclear materials: May be needed especially in the case of research reactors. In the case of accidents with the damaged fuel, dedicated buffer storages may be constructed within the reactor buildings.
(c) 10.0300 Decommissioning of buffer storage: Will be implemented in the case of dedicated buffer storage.

IV–1.11. ISDC 11: Miscellaneous expenditures

ISDC 11 concerns cost items which are directly related to a decommissioning project (i.e. are within the scope of the project) but cannot be allocated directly to any of the categories, ISDC 01–10. Examples of such items are the transition plans that compensate the shutdown of the facility or the consequences of decommissioning, pension schemes or requalification projects for personnel who leave the nuclear facility to be decommissioned, payments to authorities and various specific external services or payments with no direct allocation to ISDC 01–10. Taxes and insurances are addressed in ISDC 11. In some decommissioning projects, assets may be identified in relation to the sale of reusable equipment or materials as a result of the activities in ISDC 02, ISDC 04 or ISDC 07. Reuse of the site may play an important role in some cases. All assets during the decommissioning project are allocated to ISDC 11.

The assumed impacts of accidents on ISDC 11 activities are as follows:

(a) Additional specific projects for the transition period;
(b) Unplanned release of operating personnel.

In a standard decommissioning project, ISDC 11 activities are implemented to the extent required for a transition project from operations to decommissioning (e.g. for compensating the impact of shutdown on the site and regional infrastructure or as compensation for released operating personnel). Taxes and insurance are consequently modified from the operational period according to individual phases of decommissioning. Asset recovery depends on the types of reusable equipment and/or materials.

For decommissioning following an accident, additional transition period projects may be expected. Taxes do not differ in principle from standard decommissioning. Insurance depends on the remaining risks and asset recovery is hampered by the accident. The following gives an overview of the possible impact on specific costing categories:

(a) 11.0100 Owner costs: Additional specific transition period projects may be required;
(b) 11.0200 Taxes: Limited impact on taxes;
(c) 11.0300 Insurance: Remaining on-site risks may require additional insurance in comparison with standard decommissioning;
(d) 11.0400 Asset recovery: Limited asset recovery.

IV–2. CONCLUSIONS

The ISDC of nuclear installations has proven to be an effective tool for the presentation and benchmarking of costs for decommissioning projects. As a format that includes all typical decommissioning activities, it can be used also for the presentation of costs for decommissioning projects for a nuclear installation after an accident.

There are several cost comparison/benchmarking formats. One of them is presented in Fig. IV–1 below. The benchmarking format presents cost in US $ (million) of decommissioning projects of various NPPs; as an example, NPP 'A to K'. This style of presentation (segmented bars) shows the total costs for an individual decommissioning project, and simultaneously also shows the distribution of costs for ISDC Level 1 items. In this way, in addition to showing the differences in total costs, the differences for cost categories at ISDC Level 1 can also be visualized.

Similar presentations with further details can be provided also for an individual decommissioning project; in this case the main bars correspond to ISDC Level 1 items and the bars are segmented according to the ISDC Level 2 items. Typical distributions of costs in this ISDC format may be developed for decommissioning projects for NPPs after planned shutdown with various types of reactors. When such ISDC presentation formats are supported by assumptions and boundary conditions structured according to the ISDC items, a clear understanding of cost items for various decommissioning projects may be achieved.

In the case of a decommissioning project after an accident, typical ISDC cost spectra may be developed based on the type of accident and its impact on decommissioning. This style of presentation of costs for a decommissioning project, along with the assumptions and conditions for decommissioning in the ISDC format and detailed cost data in ISDC format Level 1, Level 2 and Level 3, appears to be a powerful tool for the comparison of post-accident, decommissioning activity costs. In the case of an NPP with decommissioning after a planned shutdown, it has been proven that the ISDC format is the only way to compare the decommissioning costs. It would be useful to prepare and to realize an international project with these objectives.

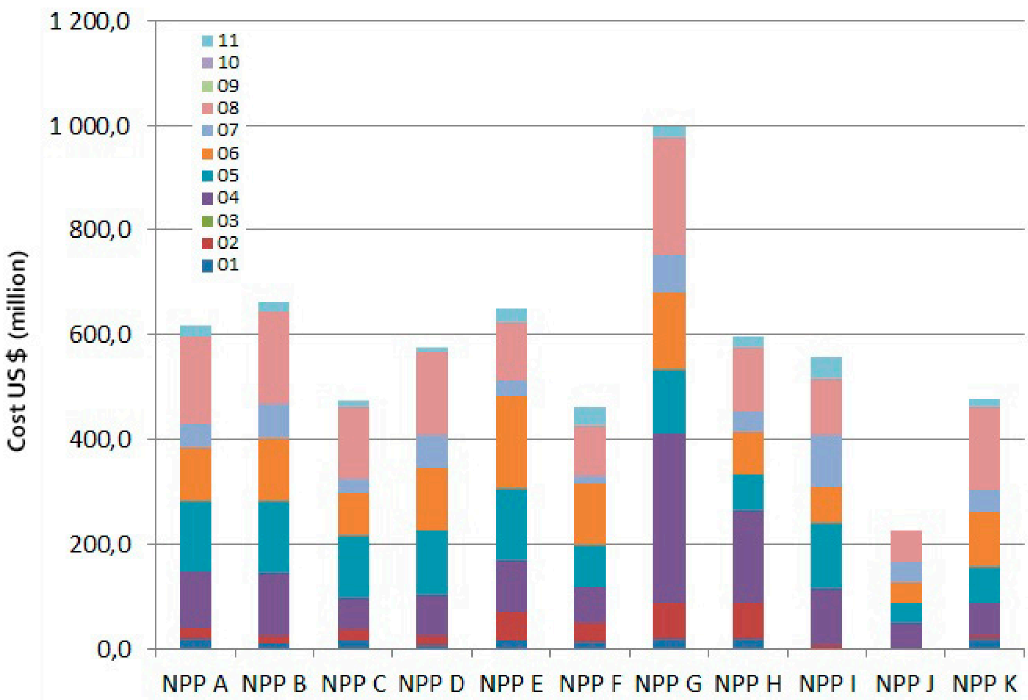

FIG. IV–1. Example of the ISDC cost benchmarking format at the ISDC Level 1. 01: Pre-decommissioning actions; 02: Facility shutdown activities; 03: Additional activities for safe enclosure or entombment; 04: Dismantling activities within the controlled area; 05: Waste processing, storage and disposal; 06: Site infrastructure and operation; 07: Conventional dismantling, demolition and site restoration; 08: Project management, engineering and support; 09: Research and development; 10: Fuel and nuclear material; 11: Miscellaneous expenditures.

REFERENCES TO ANNEX IV

[IV–1] OECD NUCLEAR ENERGY AGENCY, International Structure for Decommissioning Costing (ISDC) of Nuclear Installations, Radioactive Waste Management, OECD Publishing, Paris (2012).

[IV–2] INTERNATIONAL ATOMIC ENERGY AGENCY, Classification of Radioactive Waste, IAEA Safety Standard Series No. GSG-1, IAEA, Vienna (2009).

GLOSSARY

The symbol '' denotes a definition that differs from that provided in the IAEA Safety Glossary*[1].

accident. Any unintended event, including operating errors, equipment failures and other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection and safety.

characterization. Determination of the nature and activity of radionuclides present in a specified place.

controlled area. A defined area in which specific protection measures and safety provisions are or could be required for controlling exposures or preventing the spread of contamination in normal working conditions, and preventing or limiting the extent of potential exposures.

corium*. Melted mixture of components that can consist of nuclear fuel, fission products, control rods, structural materials from the affected parts of the reactor, products of their chemical reaction with air, water and steam, and, if the reactor vessel has been breached, concrete from the structure of the reactor space.

decommissioning. Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility.[2]

decommissioning plan. A document containing detailed information on the proposed decommissioning of a facility.

decontamination. The complete or partial removal of contamination by a deliberate physical, chemical or biological process.

demolition*. In the context of this publication, demolition is the overall process of removing the facility and usually but not exclusively relates to civil structures.

de-sludging*. The process of removing sediments by draining and cleaning (a tank, a pool).

dismantling*. The disassembly and removal of any structure, system and component during decommissioning.

disposal. Emplacement of waste in an appropriate facility without the intention of retrieval.

entombment*. Entombment is the strategy by which radioactive contaminants are encased in a structurally long lived material until radioactivity decays to a level permitting the unrestricted release of the facility, or release with restrictions imposed by the regulatory body.

fuel debris*. Any fuel rod or assembly material that cannot be retrieved as part of a fuel assembly.

mixed waste. Radioactive waste that also contains non-radioactive, toxic or hazardous substances.

nuclear legacy. Nuclear legacy facilities are buildings and/or sites that are left over from the origins of a country's nuclear activities (be they scientific, defence, or power related), and have not yet undergone complete decommissioning or remediation efforts to terminate the site license. Examples include military facilities, early prototype facilities, old front end fuel cycle facilities, and other facilities concerned with radioactive materials handling. These facilities typically feature varying states of disrepair due to a lack of consistent

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2018 Edition, IAEA, Vienna (in preparation).

[2] Except for a repository or for certain nuclear facilities used for the disposal of residues from the mining and processing of radioactive material, which are 'closed' and not 'decommissioned'.

maintenance, a lack of proper inventory recording, and a lack of a proper regulatory framework (or one that does not stand up to modern safety standards). These facilities are often characterized by varying degrees of radionuclide contamination, the presence of disused buildings and infrastructure awaiting decommissioning, and surrounding land that has been affected by past operation. US legal definition to encompass non-reactor facilities: A legacy site is a facility that is decommissioning and has an owner who cannot complete the decommissioning work for technical or financial reasons .

recovery. It is generally understood to be the process of return of affected facility and site to a state of normality after a disaster.

remediation. Any measures that may be carried out to reduce the radiation exposure from existing contamination of land areas through actions applied to the contamination itself (the source) or to the exposure pathways to humans.

risk management*. The process of identifying, assessing and controlling risks arising from operational factors and making decisions that balance risk cost with mission benefits.

robot*. An industrial robot is an automatically controlled, reprogrammable, multipurpose manipulator, programmable in three or more axes which may be either fixed in place or mobile for use in industrial automation applications. The term 'robot' is currently used for systems that have 'motion' and 'intelligence' rather than being 'multipurpose'. Teleoperated robot is a robot that can be remotely operated by a human operator.

safe enclosure*. Safe enclosure (sometimes called safe storage, safe store or deferred dismantling) is the strategy in which parts of a facility containing radioactive contaminants are either processed or placed in such a condition that they can be safely stored and maintained until they can subsequently be decontaminated and/or dismantled to levels that permit the facility to be released for unrestricted use or with restrictions imposed by the regulatory body.

spent fuel. Nuclear fuel removed from a reactor following irradiation that is no longer usable in its present form because of depletion of fissile material, poison buildup or radiation damage.

stabilization*. Activities implemented during and subsequent to the emergency response that are needed prior to beginning the intensive post-accident cleanup.

stakeholder*. Interested party; concerned party.

storage. The holding of radioactive sources, radioactive material, spent fuel or radioactive waste in a facility that provides for their/its containment, with the intention of retrieval.

ABBREVIATIONS

ALARA	as low as reasonably achievable
EPRI	Electric Power Research Institute
FAZ	fire affected zone
FCM	fuel containing material
HICs	high integrity containers
INES	International Nuclear Events Scale
IRID	International Research Institute for Nuclear Decommissioning
ISDC	international structure for decommissioning costing
LLW	low level waste
MCF	moveable cementation facility
MSPP	modernized stationary dust suppression installation
NEA	Nuclear Energy Agency (part of OECD)
NPP	nuclear power plant
NRC	United States Nuclear Regulatory Commission
NSC	new safe confinement
OECD	Organisation for Economic Co-operation and Development
PCV	primary containment vessel
RWDS	radioactive waste disposal storage
SPP	stationary dust suppression installation
SSCs	structures, systems and components
TEPCO	Tokyo Electric Power Company
TMI	Three Mile Island
USDOE	United States Department of Energy
VÚJE	Nuclear Power Plant Research Institute Trnava

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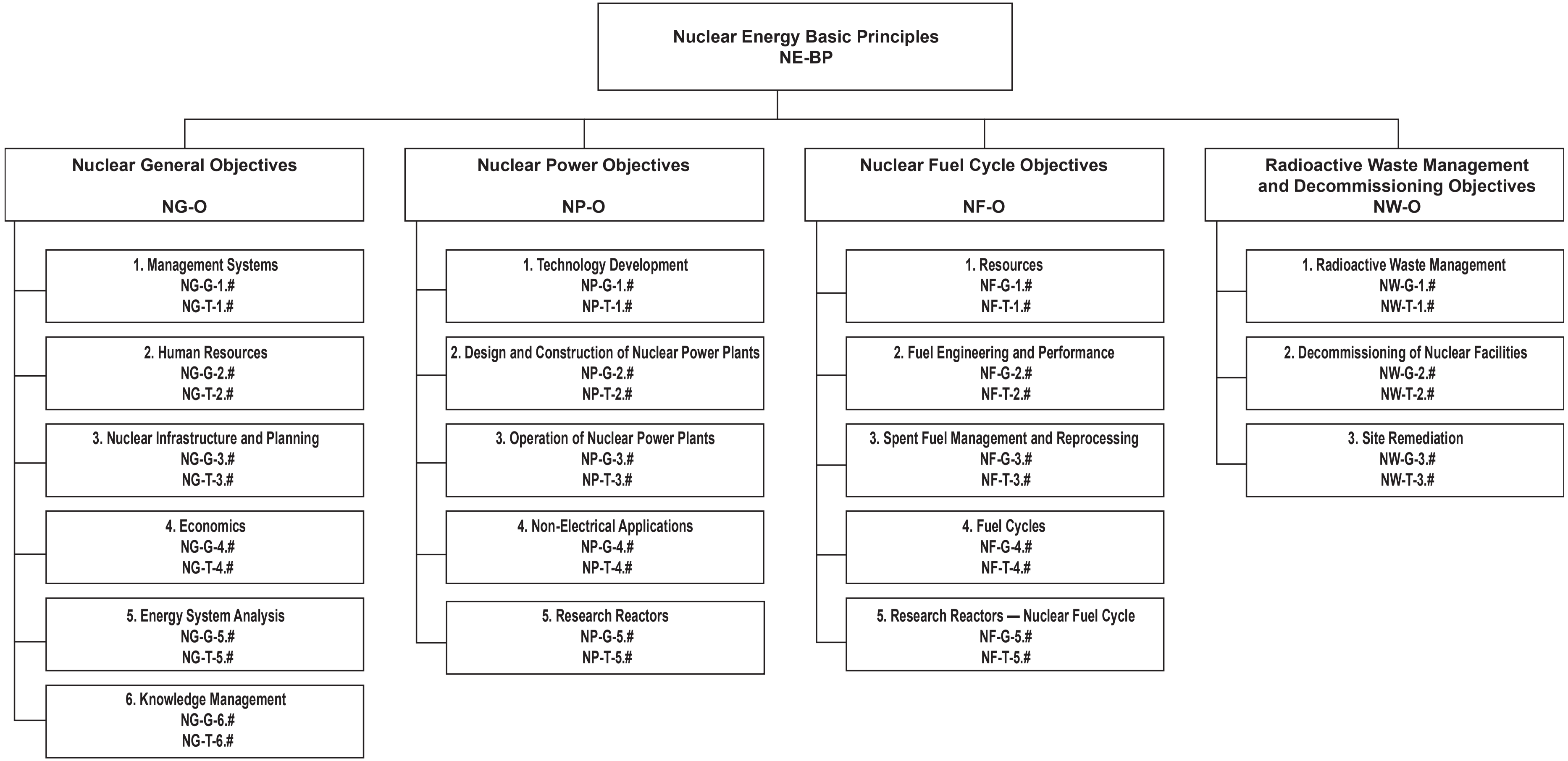

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