Nuclear Safety
2021

Long-Term Management and Actions for a Severe Accident in a Nuclear Power Plant

Status Report



OECD
BETTER POLICIES FOR BETTER LIVES

NEA
NUCLEAR ENERGY AGENCY

Nuclear Safety

Long-Term Management and Actions for a Severe Accident in a Nuclear Power Plant

Status Report

© OECD 2021
NEA No. 7506

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 37 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, Colombia, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Latvia, Lithuania, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

This work is published under the responsibility of the Secretary-General of the OECD. The opinions expressed and arguments employed herein do not necessarily reflect the official views of the member countries of the OECD or its Nuclear Energy Agency.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 34 countries: Argentina, Australia, Austria, Belgium, Bulgaria, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Korea, Romania, Russia, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission and the International Atomic Energy Agency also take part in the work of the Agency.

The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes;

– to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management and decommissioning, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

This document, as well as any [statistical] data and map included herein, are without prejudice to the status of or sovereignty over any territory, to the delimitation of international frontiers and boundaries and to the name of any territory, city or area.

Corrigenda to OECD publications may be found online at: *www.oecd.org/about/publishing/corrigenda.htm*.

© OECD 2021

You can copy, download or print OECD content for your own use, and you can include excerpts from OECD publications, databases and multimedia products in your own documents, presentations, blogs, websites and teaching materials, provided that suitable acknowledgement of the OECD as source and copyright owner is given. All requests for public or commercial use and translation rights should be submitted to *neapub@oecd-nea.org*. Requests for permission to photocopy portions of this material for public or commercial use shall be addressed directly to the Copyright Clearance Center (CCC) at *info@copyright.com* or the Centre français d'exploitation du droit de copie (CFC) *contact@cfcopies.com*.

Cover photos: Three Mile Island NPP, Chernobyl NPP (hujgcemj70, Creative Commons) and Fukushima Daiichi NPP (IAEA).

Table of contents

List of figures

List of tables

List of authors

Lead authors

D. Jacquemain	Institut de Radioprotection et de Sûreté Nucléaire (IRSN), France
S. Basu	Nuclear Regulatory Commission (NRC), United States
M. Salay	NRC, United States
T. Lind	Paul Scherrer Institute (PSI), Switzerland
P. Vokac	ÚJV Řež, Czech Republic
J.-C. De La Rosa Blul	European Commission, Joint Research Centre, the Netherlands
S. Bechta	Royal Institute of Technology (KTH), Sweden
A. Kubarev	KTH, Sweden
P. Ellison	GE Power & Water, United States
Y. Yamamoto	Tokyo Electric Power Company (TEPCO), Japan
K. Takeda	TEPCO, Japan
T. Niisoe	Nuclear Regulation Authority (NRA), Japan
V. Krasnov	Institute for Safety Problems of NPP (ISPNPP), Ukraine

Persons who contributed to the work group

A. Viktorov	Canadian Nuclear Safety Commission (CSNC), Canada
N. Mesmous	CSNC, Canada
J.-H. Song	Korean Atomic Energy Research Institute (KAERI), Korea
H.-Y. Kim	KAERI, Korea
K.I. Ahn	KAERI, Korea
M. Kim	International Atomic Energy Agency (IAEA), Austria
C. Pilleux	Electricité De France (EDF), France
A. Dubreuil-Chambardel	EDF, France
A. Hotta	NRA, Japan
R. Kojo	NRA, Japan
T. Matsuo	TEPCO, Japan
W. Giannotti	Nuclear and Industrial Engineering (NINE), Italy
K. Funaki	NEA
K. Hida	Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF), Japan

Organisations that provided information on long-term management status (cf. questionnaire in Appendix A) and for the action identification and ranking table (AIRT) exercise (Section 3.5)

W. Barten	Federal Nuclear Safety Inspectorate, Switzerland
M. Richner	Kernkraftwerk Beznau (KKB), Switzerland
J.-U. Klügel	Kernkraftwerk Gösgen (KKG), Switzerland
A. Dubreuil-Chambardel	EDF, France
T. Nemec	Slovenian Nuclear Safety Administration (SNSA), Slovenia
E. Takasuo	Radiation and Nuclear Safety Authority (STUK), Finland
H. Tuomisto	Fortum, Finland
P. Isaksson	Swedish Radiation Safety Authority (SSM), Sweden
T. van Rompuy	BelV, Belgium
M. Sonnenkalb	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany
F. Robledo	Consejo de Seguridad Nuclear, Spain

Persons who provided a complete external review of the report

Y. Maruyama	Japan Atomic Energy Agency (JAEA), Japan
D. Marksberry	NRC, United States
N. Okuzumi	International Research Institute for Nuclear Decommissioning (IRID), Japan

Persons who provided a review on specific parts of the report

J. Byrne	General Public Utilities, United States, review for RWMC group
H. Chatri	Canadian Nuclear Safety Commission (CNSC), Canada, review for WGRISK group

Secretariat

M. Kissane	NEA
N. Sandberg	NEA

This publication is dedicated to Martin Kissane who passed away in August 2020. He is sorely missed.

List of abbreviations and acronyms

AIRT	Action identification ranking table
ALARA	As low as reasonably achievable
ALPS	Advanced Liquid Processing System
BWR	Boiling water reactor
CRPPH	Committee on Radiation Protection and Public Health (NEA)
DOE	Department of Energy (United States)
EC	European Commission
EPRI	Electric Power Research Institute
FCM	Fuel-containing materials
IAEA	International Atomic Energy Agency
INEL	Idaho National Engineering Laboratory
IVC	In-vessel cooling
IVMR	In-vessel melt retention
ISLOCA	Interfacing system loss-of-coolant accident
LFCM	Lava-type fuel-containing materials
LT	Long-term
LTM	Long-term management
LTMA	Long-term management and actions
MAAP	Modular Accident Analysis Programme – Integral severe accident code developed by Fauske & Associates Inc.
MCCI	Molten corium-concrete interaction
MELCOR	Integral Severe Accident Code developed by Sandia National Laboratory for the US Nuclear Regulatory Commission
NDA	Nuclear Decommissioning Authorities (United Kingdom)
NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation (Japan)
NEA	Nuclear Energy Agency
NPP	Nuclear power plant
NRA	Nuclear Regulation Authority (Japan)
OECD	Organisation for Economic Co-operation and Development

PCV	Primary containment vessel
PDS	Plant damaged state
PEIS	Programmatic Environmental Impact Statement
PIRT	Phenomena identification ranking table
PMORPH	Primary Containment Vessel Internal Survey Equipment
PRA	Probabilistic risk assessment
PWR	Pressurised water reactor
RC	Release category
RCS	Reactor coolant system
R&D	Research and development
RPV	Reactor pressure vessel
SA	Severe accident
SAMG	Severe accident management guideline
SARRY	Simplified active water retrieve and recovery system
SDS	Submerged demineraliser system
SED	Safety and environmental detriment
SFP	Spent fuel pool
TCOFF	Thermodynamic Characterisation of Fuel Debris and Fission Products Based on Scenario Analysis of Severe Accident Progression at Fukushima Daiichi Nuclear Power Station (NEA)
TEPCO	Tokyo Electric Power Company, Japan
TMI	Three Mile Island
NRC	Nuclear Regulatory Commission (United States)
WENRA	Western European Nuclear Regulatory Association

Executive summary

This status report deals with the long-term management (LTM) of a nuclear power plant (NPP) after a severe accident (SA), and has been produced with the objective to i) review the existing regulations and guidance, practices, technical bases and issues considered in member countries of the OECD Nuclear Energy Agency (NEA) regarding LTM of an NPP; ii) exhaustively identify, describe and discuss the main challenges and issues to be tackled; and iii) propose recommendations and areas for future investigation to improve LTM of an NPP.

In the report, LTM refers to accident management actions implemented after a plant has reached a stabilised and controlled state following a reactor or spent fuel pool (SFP) severe accident and up to and including fuel and debris retrieval from the damaged plant, temporary on-site storage of the fuel and debris, and eventual transportation to off-site permanent storage. The main LTM actions aim at: i) evaluating the plant damaged state (PDS) from a physical and radiological standpoint; ii) maintaining a stabilised and controlled state of the damaged plant from a safety perspective; iii) implementing provisions against further failures; iv) cleaning up and decontaminating; v) managing accident wastes (conditioning, treatment, packaging and storage); vi) preparing and achieving fuel and debris retrieval; and vii) protecting plant personnel from exposure. Off-site long-term management and actions are not discussed, and neither are radiation protection, waste disposal and decommissioning aspects.

Based on a questionnaire that was circulated among NEA member countries it was concluded that most of these countries do not have specific regulations for the long-term phase of an SA, rather, it is commonly considered to be covered by the existing regulation and severe accident management guidelines.

Information from the three major accidents at Three Mile Island (TMI), Chernobyl and Fukushima Daiichi has been gathered and analysed in order to provide information and insights specifically for LTM. Whereas TMI-2 LTM can be considered as completed when referring to the above LTM definition, Chernobyl and Fukushima Daiichi LTM still face tremendous challenges to complete LTM with important learnings yet to come. The three accidents have shown that different challenges and issues may arise for LTM depending on accident nature and consequences with different “entry states” to LTM. For the three accidents, unique national regulatory and licensing requirements were developed and they have also required the development of complex technical means (e.g. unique systems, equipment and instrumentation), actions and organisations. The main challenge in implementing LTM in all three accidents is that it had to be conducted with limited knowledge of LTM entry state, i.e. the status of the core and the plant at the entry to the long-term management phase, of risks to evolve to a new unstable situation, and of risks related to long-term management and actions (LTMA) implementation. Looking more specifically at damaged fuel diagnostics and retrieval, the three accidents have resulted in distinctive damaged fuel distributions and characteristics, even among the three damaged reactors at Fukushima Daiichi. At TMI-2, the fuel retrieval strategy had to be revised after investigations in the reactor pressure vessel. The fuel retrieval was performed successfully with a specific defuelling platform where operations were exercised beforehand to limit workers’ exposure. At Chernobyl, investigations have shown that ageing through leaching and interactions with atmospheric gas and water may affect the integrity of the damaged fuel with possible fuel dusting in the long term. At Fukushima Daiichi, uncertainties still remain regarding the fuel distribution in the three damaged reactors, corium composition and its behaviour concerning leaching and ageing effects. The best strategies for damaged fuel retrieval have not yet been established for Chernobyl and Fukushima Daiichi.

Next, approaches to long-term management have been discussed from a general perspective. Firstly, the group developed the LTM definition and scope and identified main long-term controlled state functions and their necessary monitoring for a safe LTM. Then, an approach has been developed to identify and categorise challenges, issues and risks for LTM covering a large diversity of accident scenarios. In a first step, possible LTM "entry states", which depend on the accident progression up to that point, are systematically classified. LTM challenges and issues have been identified for each generic entry state. As a second step, a structured approach aimed at identifying and categorising main issues and risks for LTM was developed to guide LTM. Within this approach, a risk-informed, plant-specific classification method of the events, different to the one using generic entry states, is used.

Finally, an action identification and ranking table (AIRT) was developed to identify knowledge, challenges, open issues and technological gaps related to main LTM actions, such as maintaining a coolable configuration and confinement integrity, managing water wastes, solid wastes and effluents, site clean-up and decontamination, intact and damaged fuel removal from reactors and SFPs and their disposal.

Based on the material and discussions presented in this report, recommendations are given in the following areas:

- Knowledge development or consolidation for:
  - calculation tools and methods for analysis of reactor and SFP severe accidents to enhance capabilities to predict the stabilised state and the corresponding PDS;
  - status of components, equipment, systems, including passive ones, and structures after an SA with emphasis on those that contribute to maintaining a stabilised state on the long-term (LT);
  - LT phenomena that can affect LTM (e.g. corrosion-erosion reactions, fuel "dusting" and dispersion);
  - methods or expert systems for risk assessment for LTM and LTMA optimisation.
- Provision development for:
  - monitoring of the PDS and its evolution;
  - upgrading equipment, components, systems and structures for LTM;
  - developing harmonised practices and technical means to limit workers' occupational exposure in LTM.

Common between the three accidents reviewed are the serious challenges associated with handling of contaminated and leaking cooling water. Therefore, it is also recommended to develop provisions for the optimisation of management of cooling waters to facilitate LTM:

- during the emergency phase, closed-loop cooling should be implemented as early as possible;
- strategies for flooding and cooling the corium should as far as feasible avoid transfer of contaminated waters outside the confinement;
- use of water with controlled chemistry should be further studied with respect to limitation of risk for re-criticality, fission products remobilisation, corrosion, clogging, and for facilitation of water management in the long term.

Chapter 1. Introduction

1.1. Background

Following the accident at the Fukushima Daiichi nuclear power plant (NPP), one of the imperatives for the nuclear safety, science and industry communities is to reassess the safety of existing NPPs, notably to evaluate the sufficiency of technical means and administrative measures addressing the management of accidents for the design basis and beyond and including long-term phases.

Up to now, international actions primarily addressed lessons learnt from the Fukushima Daiichi accident for the management of short-term phases (emergency operating procedures and severe accident management guidelines [SAMGs] domains) and for emergency preparedness, with exceptions mentioned in Section 1.2, but the long-term management and actions (LTMA) have not been examined in detail.

In recognition of international interest in additional valuable information that could be gained from the Fukushima Daiichi accident LTMA, NEA Committee on the Safety of Nuclear Installations (CSNI) recommended at the end of 2014 to launch an action i) that addresses a review of the experience gained from management of long-term phases in Three Mile Island unit 2 (TMI-2), Chernobyl and Fukushima Daiichi accidents and of technical and organisational measures and practices in NEA member countries, ii) that would provide recommendations for enhancement for the management of long-term phases of a severe accident (SA) and identify related safety research knowledge gaps. It was felt that both enhancements in terms of technical means and organisation could be proposed to optimise implementation of LTMA.

This report describes the outcomes of this action that was led by a working group of the NEA Working Group on Accident Management and Analysis (WGAMA) with the final objective of providing guidance for the international community to enhance safety related to the long-term management of a severe accident and engage well focused research in the field.

1.2. Objectives and scope

The objectives of the Long-Term Management of Actions for a Severe Accident in a Nuclear Power Plant (LTMNPP) Working Group were to:

- review the experience gained for LTMA from the TMI-2, Chernobyl and Fukushima Daiichi accidents;
- review envisaged, planned or existing regulations, guidance and practices in NEA member countries for LTMA for a severe accident in an NPP;
- suggest possible approaches for LTMA following a severe accident including methods for risk ranking and issues identification;
- identify the main challenges and open issues faced during the long-term phase of a severe accident;
- derive recommendations to better address them in the future, including, for instance, orientation onto where research and development (R&D) efforts should be allocated, or improvement of methods to assess LTMA;
- derive recommendations to enhance LTMA.

The report is primarily focusing on on-site LTMA following an SA and is not covering issues related to off-site long-term management and actions. Specific working groups and actions are dealing with this issue. Also, radiation protection, waste disposal and decommissioning aspects have not been treated in detail here as specific NEA working groups and actions were conducted to treat them – the Committee on Radiological Protection and Public Health (CRPPH) (NEA, 2014) and Radioactive Waste Management Committee (RWMC) (NEA, 2016) (Expert Group on Fukushima Waste Management and Decommissioning R&D [EGFWMD] and Working Party on Decommissioning and Dismantling [WPDD]).

In addressing the long-term management (LTM) action, elements of interest arising from related NEA working groups such as the NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station Project and the NEA Senior Expert Group on Safety Research Opportunities Post-Fukushima (SAREF) have been considered. These are reported and discussed in the chapter describing the experience gained from Fukushima Daiichi accident for LTMA.

1.3. Structure of the report

Chapter 2 provides a description of the state of the art of long-term management and actions based on severe accidents experience (TMI-2, Chernobyl and Fukushima Daiichi), on information compiled through a questionnaire on envisaged, planned or existing regulations, guidance and practices in NEA member countries for LTMA for a severe accident in an NPP and on a review of relevant existing reports in the literature.

Chapter 3 suggests possible approaches to long-term management and actions considering diverse scenarios of SA and spent fuel pool (SFP) accidents and including methods for issues identification and risks ranking.

Chapter 4 provides a summary and a discussion of main identified challenges and open issues faced during the long-term phase of a severe accident.

Chapter 5 provides some recommendations for enhancing long-term management and actions better addressing identified challenges and open issues, including the development of provisions, of approaches and the design of future research to consolidate relevant knowledge.

Chapter 2. State of the art of long-term management and actions

2.1. Feedback from Three Mile Island unit 2, Chernobyl and Fukushima Daiichi accidents

This chapter provides a description of key technical and organisational issues identified from the long-term management of the accident at Three Mile Island unit 2 (TMI-2), Chernobyl and Fukushima Daiichi.

Synthesis of relevant Three Mile Island unit 2 experience

The TMI-2 nuclear power plant (NPP) accident that occurred on 28 March 1979, was the most serious accident in the history of US nuclear power industry (ANS, 1989). The accident resulted in partial meltdown of the reactor core, and though the reactor pressure vessel did not breach, there was a small radioactive release with no detectable consequence on plant workers' health, public health and environment. The nuclear industry had no prior experience in dealing with this kind of event; neither did the US Nuclear Regulatory Commission (NRC) have prior experience with regard to its regulatory oversight role. The guiding principle for actions to be taken by both parties in the aftermath of the accident was to ensure protection of public health and safety, and that of the environment.

The recovery and clean-up of TMI-2 (EPRI, 1992a, 1992b, 1992c, 1990a, 1990b; IAEA, 2014, 1992, 1991a. 1991b; DOE, 1993, 1990, 1988; NRC, 2016), including removal of the fuel from the accident-damaged reactor, were done for the long-term protection of public health and safety and of the environment. Implementation of recovery and clean-up activities was the responsibility of the licensee with support from US nuclear industry, US Department of Energy (DOE), and several international organisations. The NRC was responsible for the regulation and oversight of TMI-2 clean-up operations to ensure the health and safety of the public and the plant personnel, as well as the protection of the environment. NRC's involvement covered two major areas: approving the recovery methods employed by the licensee and responding to public concerns over radiation exposure resulting from the accident and clean-up activities.

The DOE initiated an extensive research programme, as directed by the US Congress, to support recovery and clean-up operations (DOE, 1993, 1990, 1988). The damaged fuel from the reactor core was removed and shipped for research to national laboratories which provided much-needed technical support for the TMI-2 recovery and clean-up programmes.

The NRC accomplished its regulatory responsibilities for all post-accident operations at TMI-2 through licensing actions; safety evaluations of recovery and clean-up activities; inspections; daily interactions with the licensee and their contractors; communications with state and local governments and the public; co-ordination with other Federal agencies involved in the clean-up; and sometimes direction from the NRC Commissioners. Formal licensing actions were required for proposed changes to NRC orders, the facility licence, technical specifications, the recovery operations plan, the organisation plan, and exemptions to regulations.

The licensee's (TMI-2 plant owner) organisational structure and functions changed as the clean-up progressed through various stages of the recovery effort, and as more information became known about the condition of the damaged reactor core, reliability of plant systems and structures, and radiological characterisation of accident-generated water, building structures, reactor components, and systems. Various organisations formed working groups to provide guidance on addressing specific issues, problems, and research activities. Independent oversight groups reviewed, monitored, and advised on the overall direction of recovery and clean-up plans and activities.

Changes in the facility's post-accident mode of operations required unique regulatory and licensing actions. In order to properly reflect evolving plant status, the NRC-issued orders, modified those orders, approved licence amendment requests, and granted relief from certain regulatory requirements. The recovery operations plan defined the surveillance requirements to be performed to ensure equipment operability as required by the plant's technical specifications. Surveillance requirements were approved by the NRC staff. The NRC granted exemptions from certain requirements of the regulations for NPPs, but only under special circumstances as permitted in the regulation. Exemptions were necessary at TMI-2 because of the plant's damaged configuration and changing status during clean-up.

The commission established a 12-member TMI-2 advisory panel in October 1980 to consult with, and provide advice to the NRC Commissioners and staff on major activities related to the decontamination and clean-up of TMI-2. The panel consisted of members from the state government (Pennsylvania), local government, and the scientific community, as well as residents in the vicinity of TMI. The NRC TMI Program Office (TMIPO) acted as a liaison between the NRC and the TMI-2 advisory panel and also provided information to the panel on the status of the clean-up. The most crucial panel influence on clean-up activities was the increased public scrutiny of both NRC and licensee decisions and activities. The panel facilitated communication with the public for both the NRC and the licensee. This communication helped sensitise the agency and the licensee to public concerns.

Advisory and working groups were formed by the General Public Utilities Corporation (GPU), the NRC, and the DOE to provide advice, and sometimes direction, on important recovery activities. Key groups that supported long-term recovery of TMI-2 included the following. The NRC "Advisory Panel for the Decontamination of TMI-2," was an influential 12-member panel that provided advice to the NRC Commissioners and staff on major activities related to the decontamination and clean-up of TMI-2. The TMI-2 "Safety Advisory Board" was established by the licensee to provide the licensee with an independent appraisal of the recovery programme that gave particular emphasis to the assurance of public and worker health and safety. This board's members were national known experts. The TMI-2 "Technical Advisory and Assistance Group" was established by the licensee, with the co-operation of the DOE and NRC, to provide independent technical assessment and advice on the decontamination and defuelling of TMI-2. The "TMI Information and Examination Program" was established by the DOE to acquire data to improve current understanding of nuclear plant accident environments and of the phenomena which contributed to those environments. Membership included the licensee, the Electric Power Research Institute (EPRI), the NRC and the DOE.

The TMI-2 clean-up effort took about 14 years, and the reactor vessel defuelling operations spanned a five-year period. A total of about 133 000 kg of fuel, cladding, structural, and control materials were removed from the reactor vessel during the five-year effort. Unique systems and equipment were designed and installed to remove damaged fuel and structural debris from the reactor vessel. The clean-up activities challenged the management of various forms and concentrations of radioactive waste. The management of highly contaminated water, fuel debris, and related solid-waste by-products included handling, processing, temporary on-site storage, transportation, and final disposal. Decontamination activities resulted in substantial quantities of contaminated water and organic resins and inorganic zeolites produced from water processing systems. Fuel debris that spread throughout the plant created unique radiological waste characteristics. Also, some waste did not fit into established regulatory waste classification categories for transportation and disposal.

The NRC's Programmatic Environmental Impact Statement (PEIS) related to the decontamination and disposal of radioactive wastes resulting from the accident was an important set of guidance documents for the NRC and licensee. The PEIS discussed the options and associated environmental impacts of four fundamental activities necessary to the clean-up: treatment of radioactive liquids; decontamination of the building and equipment; removal of fuel and decontamination of the coolant system; and packaging, handling, storing and transporting nuclear waste. These items are discussed below in some details.

In terms of radiological health and safety, there was no known technical reason for the radiological release criteria to be more restrictive than had been acceptable at normal operating facilities. However, because of the unique characteristics of the clean-up operation that were not considered and evaluated in the pre-accident safety review of the plant, there was a need to define what keeping radiation exposure as low as reasonably achievable (ALARA) meant with respect to off-site releases and occupational exposures. The PEIS provided the basis for making that determination.

The following sections briefly describe, after presenting actions taken during the "stabilisation phase", the key long-term accident management insights and issues identified from the accident at the TMI-2 plant (ANS, 1989; EPRI, 1992a, 1992b, 1992c, 1990a, 1990b; IAEA, 2014, 1992, 1991a. 1991b; DOE, 1993, 1990, 1988; NRC, 2016). These insights may serve as a basis for inputs into a long-term post-accident management plan. The items for consideration are:

- waste water management;
- decontamination of buildings and equipment;
- defuelling;
- disposal and long-term storage of fuel and other radioactive wastes;
- worker protection.

Actions taken during the stabilisation phase

During the short-term accident management phase of TMI-2, the pressing need was to stabilise the conditions throughout the plant – hence, this phase was more commonly referred to at the time as the "stabilisation phase." This phase, which lasted for just over 16 months, involved certain essential plant operational and control functions to keep the reactor cool. Specifically, these functions were: i) maintaining reactor coolant system (RCS) flow and water inventory; ii) controlling RCS pressure; iii) maintaining RCS heat removal; and iv) ensuring the core remain subcritical.

The severely damaged core and resulting large quantities of non-condensable gas generation posed a formidable challenge for the primary coolant flow from being blocked. The operations staff developed procedures to control and remove non-condensable gases, in particular, hydrogen. The degassing operation was successfully conducted by maintaining the system pressure relatively high (6.2-6.9 MPa) and by continually adding reactor coolant make-up water. The pressuriser temperature was maintained at a considerably higher level than the coolant system, which helped the gas to effervesce and to be removed by venting. Hydrogen was effectively removed in this manner in about six days.

The decay heat removal system in TMI-2 was designed for 2.4 MPa. Because of the need to control the RCS pressure in the 6.2-6.9 MPa range during the degassing operation, the decay heat removal system was not used in the first week of the stabilisation phase, which turned out to be beneficial for keeping the dose rate low. The only viable installed alternative to the decay heat removal system was to use the steam generators to achieve and maintain cold shutdown.

The RCS pressure and volume control systems were lost as a result of the accident. Operators worked around the problems temporarily while the construction of a new pressure control system (standby pressure control system [SPCS]) began. The SPCS consisted of pumps and accumulator tanks for surge suppression using bottled nitrogen. The construction was completed at the end of 1979, and the system remained operational from early 1980 to 1984 when the reactor was depressurised for vessel head removal.

The exact severity of core damage was unknown; however, the control material was essentially non-existent in the core region as a result of melting. Thus, control rods were not relied upon in any way for reactivity control and assurance of shutdown. The operators had no way of measuring the criticality margin. Thus, a high boron concentration was maintained in the coolant to ensure that the core would not become critical. Later on, it was also determined that re-criticality would be virtually impossible because of the non-optimum configuration of the agglomerated melt in relation to the surrounding water.

Besides the main task of reactor control during the short-term accident management phase (i.e. stabilisation phase), controlling radioactivity was also a high priority task as any off-site release could potentially have a large environment and health consequence. During the accident, ruptured fuel released substantial quantities of gaseous fission products into the reactor coolant. These gases were carried to various low pressure tanks in the liquid clean-up system, and were the primary source of the problem. These gaseous releases were causing serious airborne contamination inside the auxiliary building. Several actions were initiated to address the problem which include: i) releasing the contents of the waste gas decay tanks into the containment; ii) finding the source of leakage into the auxiliary building; iii) replacing the charcoal filter; iv) designing a charcoal bed adsorption system; v) designing and constructing a new (auxiliary and fuel handling building [AFHB]) ventilation system; and vi) purging the containment of krypton-85.

Many of the actions initiated during the short-term accident management at TMI-2 and equipment/procedures – existing or implemented after the accident – were continued beyond this phase. Examples are maintaining reactor control and having access into the containment and auxiliary building, waste water processing and storage, initial decontamination effort, etc.

Waste water management

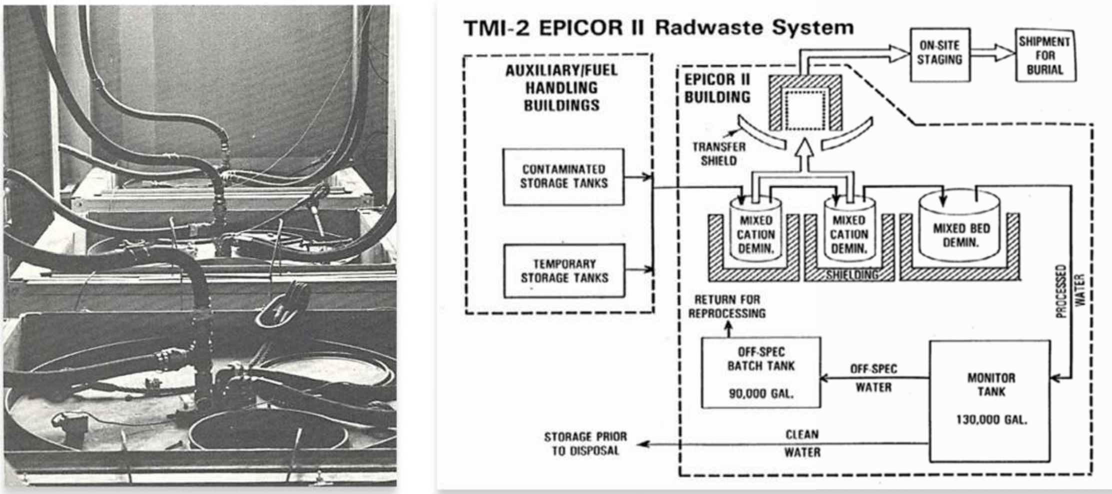
During the first hour of the TMI-2 accident, spilled reactor coolant in the reactor building sump was automatically pumped into the auxiliary building holding tanks, which then overflowed along with the sumps. Leakage from the make-up and purification system did contain fuel debris and fission products which subsequently mixed with the water in tanks, sumps, and floor drains. Over one million gallons (3.8 million litres) of contaminated water ended up in the basement (Figure 2.1) of the reactor building (creating an over two-metre deep pool) and in tanks in the auxiliary building.

Figure 2.1. **Contaminated water in the basement of the reactor building**

Source: NRC (2016).

Contaminated water prohibited access to the reactor building and large areas of the auxiliary and fuel handling buildings. Removal of the contaminated water from the buildings was the first major decontamination task (NRC, 2016). The NRC's approval was required to dispose of the decontaminated water. On October 1979, an order for modification of licence was issued to require the licensee to promptly operate the EPICOR-II filtration and ion-exchange decontamination system (Figure 2.2) to decontaminate intermediate-level radioactive waste water held in tanks in the TMI-2 auxiliary building. Processing of intermediate-level waste water inside the auxiliary building began almost immediately and was completed in little over one year time. During this period, over one million gallons of water was processed, with about half of this amount being recycled processing. After processing, the water was collected in a clean water receiving tank for sample measurements of radionuclide concentrations.

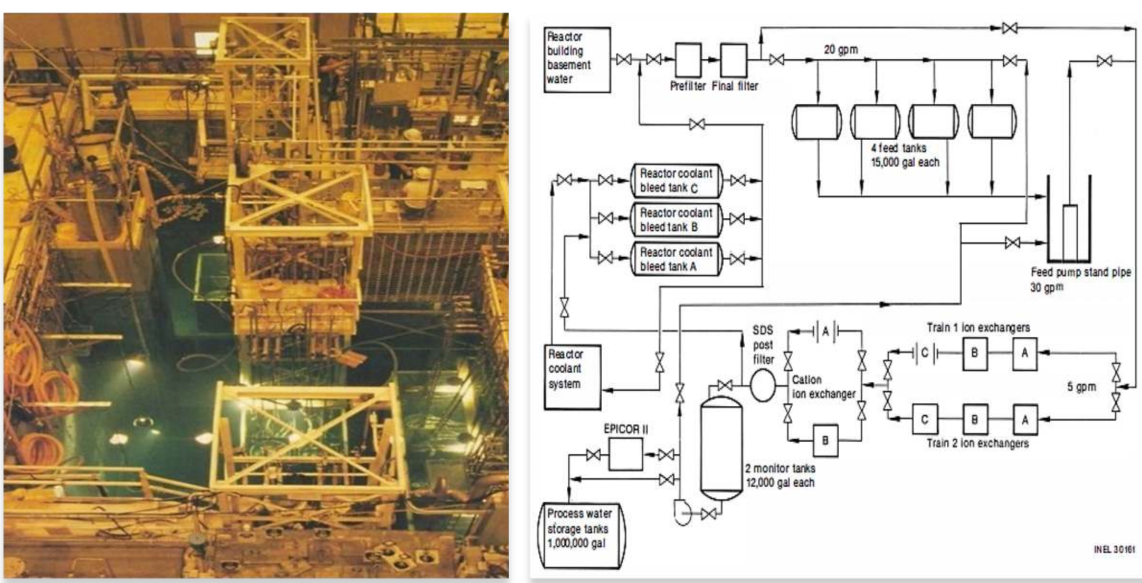
Figure 2.2. **Processing and decontamination of radioactive waste water by the EPICOR-II system**



Source: NRC (2016).

The NRC approved the use of the submerged demineraliser system (SDS) which was designed and installed to clean-up high-level radioactive accident-generated waste water from the reactor building's basement, reactor coolant system, and reactor coolant bleed tanks (Figure 2.3). The SDS consisted of a liquid waste treatment subsystem, a gaseous waste treatment subsystem, and a solid waste handling subsystem. The liquid waste treatment subsystem was designed to remove caesium and strontium from the high-activity waste water by filtration and ion exchange. The SDS started operation in June 1981.

Figure 2.3. **Processing and decontamination of high-level radioactive waste water by the SDS**

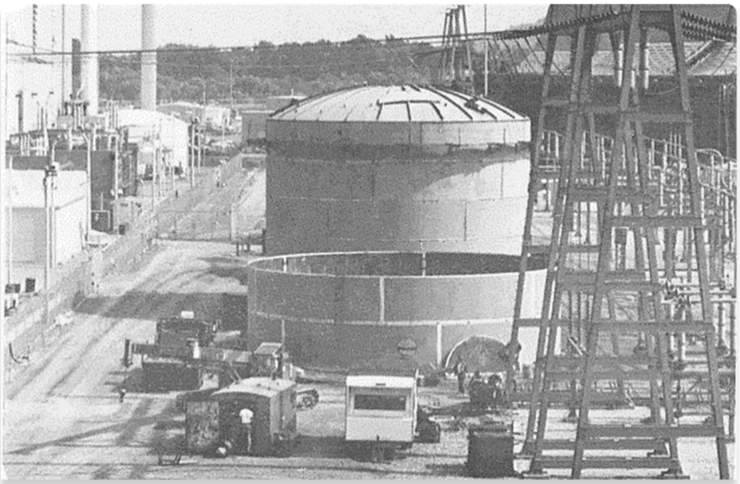


Source: NRC (2016).

From September 1981 to April 1987, EPICOR-II was used to remove residual radioactivity from SDS effluents and to process miscellaneous wastes from 11 September 1981 to April 1987. After the SDS was removed from operation in 1988, EPICOR-II was the primary system to clean-up waste water that was mainly generated from building decontamination activities. The system configuration was the same as before; however, a high integrity container loaded with zeolite resins was placed in the first position to act as a roughing filter to remove gross caesium and strontium radionuclides.

The processed water from the EPICOR-II and SDS clean-up systems was stored on-site in two 500 000 gallon (2 million litres) tanks (Figure 2.4) which became operational on July 1981. A recycling system was later installed to transfer the processed water into the reactor building for decontamination activities. This water was subsequently reprocessed by the SDS and returned back to the processed water storage tanks for further use or to await ultimate disposal.

Figure 2.4. **Processed water storage tanks to hold processed water from EPICOR-II and SDS clean-up systems**



Source: NRC (2016).

The processed water disposal system (a closed cycle evaporator) was used to dispose of 2.3 million gallons (8.7 million litres) of processed accident-generated water. The SDS removed 99% of the radioactivity and boiling the decontaminated water twice in the evaporator removed most of the remaining traces of radioactive particles. However, radioactive tritium could not be removed and was released into the environment during the evaporation process. The maximum radiation exposure from the evaporation process was less than one millirem (10 μSv), which is equivalent of about one day of background radiation.

Decontamination of the buildings and equipment

Exposed surfaces in the reactor building and the auxiliary and fuel handling building were contaminated with radionuclides. Airborne releases entered the ventilation systems and spread throughout the auxiliary and fuel handling building. Airborne releases contaminated surfaces on the upper-level floor and mid-level floor of the auxiliary building, and liquid releases to the drain system contaminated surfaces on the basement-level floor. The interior of the auxiliary building, including 26 piping systems, was contaminated by radioactive material though less severely than the interior of the reactor building. After the accident, the water in the reactor building's basement was heated by residual heat from the reactor vessel, evaporated, condensed on the cooler walls, and drained down onto the floors and back into the basement.

The overall objectives of the TMI-2 decontamination efforts were to maintain access to and operability of plant systems, to support defuelling preparations and operations, and to permit the transition of the facility to a long-term storage condition. Shorter-term decontamination objectives focused on the removal or stabilisation of contamination in order to reduce occupational exposure and to prevent release of contamination to the environment. The decontamination objective was to stabilise localised radiological conditions in the plant, regardless of whether or not access was required for clean-up activities. Longer-term decontamination objectives ensured that any remaining contamination was stable and sufficiently isolated for long-term storage.

The decontamination objectives for the reactor building working areas were to reduce radiological conditions (general area radiation, airborne gaseous and particulate activities, and surface contamination levels) to ALARA levels and to maintain those conditions in a way that would permit defuelling operations. The decontamination objectives for the auxiliary and fuel handling building were to permit access without restriction because of surface or airborne contamination, to reduce radiation exposure from gamma sources to ALARA levels, and to prevent recontamination from other clean-up activities or system leaks.

Clean-up of the auxiliary and fuel handling building started shortly after the accident. Approximately 510 000 square feet (47 300 m^2) of surface in the auxiliary and fuel handling building required decontamination when clean-up operations began. The reactor building was an effective barrier against the release of radioactivity off-site and as a result the building's interior was extremely contaminated. The decontamination work was in preparation for the removal of the damaged fuel core from the reactor vessel. Radioactive gases (Krypton 85) accumulated in the reactor building and had to be removed (vented) before the building could be safely entered. Clean-up of the reactor building did not begin immediately as the humidity in the reactor building was 100% causing precipitation in the form of rain (steadily dripping condensation) inside the reactor building.

Decontamination of building surfaces, systems, and equipment included multiple activities across the following categories:

- Loose and installed equipment: removal of miscellaneous loose equipment and debris that were in the facility at the time of the accident, such as ladders, scaffolding, tools, and portable equipment. Also, decontamination or removal of installed equipment, such as piping systems, air conditioning and exhaust equipment, cable trays, and electrical and lighting equipment.
- Interior surfaces: decontamination of interior building surfaces consisting of metal and concrete materials.
- Sludge and resins: removal of contaminated sediment (sludge) from tanks and sumps in the auxiliary and fuel handling building and from the reactor building's basement floor and sump. Also, removal of highly contaminated resins from the make-up and purification system demineralisers located in the auxiliary building.
- Recovery and clean-up of equipment: decontamination of systems and equipment used for clean-up and defuelling activities. Gross decontamination of refuelling tools and clean-up equipment for reuse or disposal was performed in two temporary equipment decontamination sites.
- Support activities: various supporting activities to ensure worker safety and to measure the effectiveness of the clean-up.

Methods used for decontamination of surfaces inside the reactor building were based on the results of the gross decontamination experiment and subsequent experience gained in decontamination of the auxiliary and fuel handling building. The following decontamination methods were reviewed by the NRC and were used:

- Abrasive blasting of steel surfaces with particulate driven at high velocity to remove contamination. This method was especially suited for small or irregular surfaces that were not compatible with other decontamination techniques. This method was also used in the on-site special decontamination facilities for tools, equipment, and other materials.

- Chemical decontamination of external surfaces of pipes, tanks, and system internals. The gross use of chemical agents was limited because of the potential for drain-off of such agents to damage water processing systems. This method was also used in the on-site special decontamination facilities for tools, equipment, and other materials.
- Dry vacuuming to remove powdered contaminants and dried residue. This method was used especially for water-sensitive components that could not be flushed with water.
- Low pressure water flush at levels between 100 to 1 000 psi (0.7 to 7.0 MPa), flow rates up to 25 g/min (95 l/min), and water temperatures up to 170°F (77°C). This method was used to clean equipment and loose surface debris. Examples included the polar crane, steam generator housing structures, missile shields, refuelling canal, and refuelling bridge.
- High pressure water flush to remove unbonded surface coatings at pressures between 2 000 to 10 000 psi (14 to 70 MPa) and flow rates of 4 to 30 g/min (15 to 114 l/min).
- Wet vacuuming to remove puddles of contaminated cleaning fluids after flushing.
- Ultra high pressure water flush to remove rust, scale, nuclear-grade coatings, and surface concrete at pressures up to 60 000 psi (410 MPa) and flow rates of 1 to 2 g/min (4 to 8 l/min).
- "Scabbling" of walls and floors to aggressively remove concrete surfaces and surface coatings. The scabblers used pneumatically operated reciprocating pistons equipped with tungsten carbide bits to pulverise the concrete surface. A vacuum system with a high-efficiency particulate air (HEPA) filter was attached.
- Steam and vacuum decontamination system to decontaminate painted and uncoated concrete, ductwork, diamond deck plates, lead bricks, penetration covers, piping, conduit cable trays, and drain covers.
- Strippable coatings that involved the application of an organic coating which contained chemicals to aid in the removal of radioactive contaminants from the surface. As the coating dried, it cracked and peeled away from the surface.
- Scrubbing to remove loosely held contamination on floors and walls using manually applied or mechanically driven rags, absorbent cloths, brushes, pads, grit, and chemical agents.

Contamination was washed from the floors, walls, pipes and other areas using high pressure water sprays (Figure 2.5) and strippable coatings (Figure 2.6). Contamination was also removed from concrete using air-operated chisels and hydraulic pounding machines to break up the top layer of concrete.

Figure 2.5. **High pressure spray decontamination**

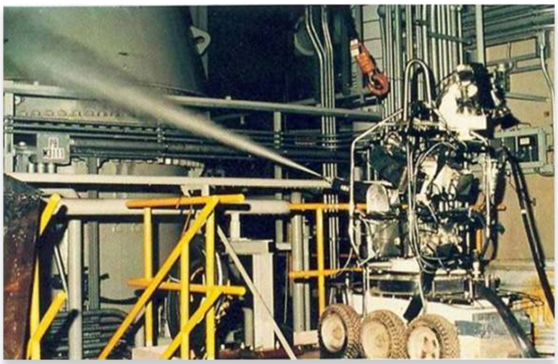
Figure 2.6. **Strippable coating for floor decontamination**

Source: NRC (2016).

Remote controlled robotic vehicles and supporting control equipment were used extensively to perform work. These vehicles were both versatile and productive, and proved useful in many different tasks, including video camera inspections, radiation monitoring, sediment sampling, acquisition of concrete core samples, high pressure water flushing (Figure 2.7), concrete scabbling, and debris pickup and removal. Specially designed robots were sent into high radiation areas to do decontamination work and obtain information on physical and radiological conditions.

In addition to the remote vehicles, fixed-position, remotely operated tools were developed for work inside the reactor vessel. The tools included a plasma arc cutting system to remove the stainless steel core support assembly, and several manipulator arms for handling damaged fuel and structural components. The use of robots at TMI-2 did not require any NRC licensing actions; however, activities in which robots were used, like most recovery and clean-up activities, required safety evaluations by the licensee and the NRC. Key robots used at TMI-2 include: remotely operated video enhanced receiver (ROVER) or remote reconnaissance vehicle to perform video and radiation surveys, collect sludge samples from the floor, collect core samples from the wall surface, and other reconnaissance activities; LOUIE I remote vehicle to measure the radiation profiles and to remove loose pre-accident debris and salt deposits on the floor; LOUIE II remote vehicle to perform remote floor scabbling and pulverise the floor surface while vacuuming loose concrete; WORKHORSE or remote work vehicle for decontamination and demolition work in the basement of the reactor building; and a Mini- ROVER which was a commercial submarine vehicle modified to remove larger fuel debris inside the pressuriser.

Figure 2.7. **Remote control robotic high pressure spray washing system**



Source: NRC (2016).

By the end of the clean-up programme, the floor contamination levels in most areas of the auxiliary and fuel handling building were reduced to those typical of pre-accident conditions. In the reactor building, radiation levels in frequently accessed areas were reduced by 85%. In November 1982, upper-level corridors of the auxiliary building became accessible to workers without the need to wear anti-contamination clothing. Respirators had not been required for entry into the upper corridors since October 1979. On 28 June 1984, workers entered the reactor building without respiratory protection for the first time since the accident, and subsequent entries were made without respirators, in accordance with ALARA principles. However, once the defuelling operations began, the hazard from discrete (hot) radioactive particles was a major concern that required respiratory protection. An elaborate system of administrative protective zones (successive rings of increasing contamination with the defuelling platform at the centre) was established to contain hot particles.

Defuelling

The TMI-2 accident resulted in severe reactor core damage and migration of molten core materials onto the reactor vessel's lower head. Core damage and relocation occurred within four hours of the accident initiation, after which long-term cooling stabilised the damaged reactor core. Various reactor core inspections and observations from early defuelling activities were documented in an Idaho National Engineering Laboratory (INEL) report which provided the technical basis for planning defuelling approaches and necessary equipment. During the period of exploration of the damaged reactor core, information from new inspections and observations altered defuelling strategies and tool designs. Camera inspections established that there was empty space where the top five feet of the reactor's core should have been. During the accident, the top section of the fuel had collapsed into a bed of rubble. Later, camera inspections inside the reactor vessel confirmed the existence of another bed of rubble at the bottom of the reactor (Figure 2.8).

Figure 2.8. **Rubble bed at the bottom of the reactor**

Source: NRC (2016).

The information gathered from these inspections was applied to the training of defuelling operators and the development of new tools and equipment. Operators trained for several months on a full-scale defuelling work platform (Figures 2.9 and 2.10) constructed outside the reactor building. The training and decontamination work reduced radiation exposure to levels comparable with levels in the refuelling of a normal operating plant.

Figure 2.9. **Full-scale defuelling work platform**

Figure 2.10. **Rotating defuelling platform operation**

Source: NRC (2016).

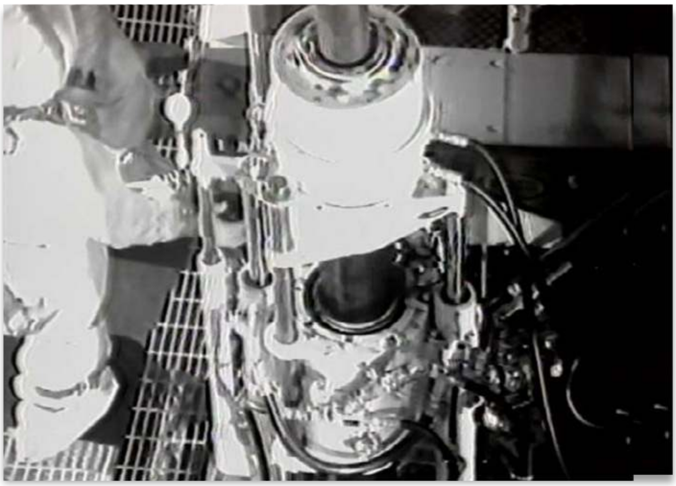
The TMI-2 clean-up effort took about 14 years with a collective manpower effort of over 3.6 million person-hours to complete. The reactor vessel defuelling operations spanned a five-year period from October 1985 through January 1990 and involved over two million person-hours. A total of about 133 000 kg of fuel, cladding, structural, and control materials were removed from the reactor vessel during the five-year effort. During July and August of 1991, the reactor vessel was drained to make final measurements of the residual fuel remaining in the vessel. An estimated residual fuel quantity that remained in the reactor vessel following defuelling was approximately 1% of the original 94 000 kg of uranium oxide fuel inventory. The distribution of core material as documented in the final defuelling report is summarised below:

- Upper core void: the upper core void or cavity consisted of 42 partially intact fuel assemblies standing at the periphery of the void. The void was about 26% of the original core volume and measured 1.5 m deep from the top of the original core to the debris bed.
- Upper debris bed: the debris bed consisted of 26 000 kg of core material, such as whole and fractured fuel pellets, control rod spiders, fuel assembly end fittings, broken fuel rods, and resolidified debris. The bed rested on top of the resolidified, hardened mass and was about 0.6 m to 1 m deep.
- Resolidified mass: the solid metallic and ceramic mass consisted of about 33 000 kg of core material. The mass rested on partially intact fuel assembly "stubs". The mass measured about 3 m in diameter, 1.5 m deep in the centre, and 0.25 m deep around the edges.
- Intact assemblies: partially intact fuel assembly stubs located under the resolidified mass and the peripheral standing assemblies comprised about 45 000 kg of core material.
- Upper core support assembly: the upper core support assembly included vertical baffle plates that formed the peripheral boundary of the core, horizontal core former plates to which the baffle plates were bolted, the core barrel to which the core formers were attached, and the thermal shield. The assembly retained about 4 000 kg of loose debris and resolidified material. A resolidified crust ranging from 0.5 to 4 cm thick was attached to the bottom of three core former plates.
- Lower core support assembly: the lower core support assembly structures retained about 6 000 kg of resolidified material in five layers around the circumference of the structures.
- Lower head region: the reactor vessel's lower head region contained about 12 000 kg of loose core debris and 7 000 kg of agglomerated core debris. The debris on the lower head was 4 m in diameter and 0.75 to 1 m deep. The surface debris had particle sizes which varied from those of large agglomerated debris (up to 0.20 m) to those of granular particles.
- Fuel debris distribution outside the reactor vessel: a total of about 228 kg of fuel debris were transported through the reactor coolant system. About 95% of the debris settled in the reactor coolant system. Approximately 10 kg total were deposited in the reactor building on the basement floor and sump (5 kg) and make-up and purification system let-down coolers (4 kg). A total of 23 kg entered the auxiliary building, mainly deposited in the three reactor coolant bleed tanks (a total of 15 kg) and in the make-up and purification system (6 kg).

Before core debris could be removed from the reactor vessel, preparations were required to allow direct access to the damaged reactor core. While many of these activities were routine during normal refuelling operations, the effects of the severe accident on reactor vessel components, the reactor building's environment, and occupational radiation exposures presented complex challenges. Preparations included consideration of numerous potential safety issues; for example, occupational exposures; decay heat removal; criticality control; boron dilution; radioactivity releases; hydrogen evolution inside the reactor coolant system; pyrophoricity (spontaneous ignition in air) of zirconium dust in the reactor vessel; heavy load drops; polar crane failure; reactor vessel draining; and fire protection. Technical considerations included potential distortion; warping or physical dislocation of the reactor vessel's head or upper plenum; reactor coolant clean-up; reactor coolant system depressurisation; and lowering of reactor coolant level. In addition, clean-up of the reactor building, which included atmospheric gases, basement water, and surface contamination, was an important prerequisite to ensure lower radiation exposures.

Unique systems and equipment were designed and installed to remove damaged fuel and structural debris from the reactor vessel. In the early defuelling phase, tools were designed for "pick-and-place" in which debris was picked up and placed into fuel containers (baskets) or specially engineered defuelling canisters. Some long-handled tools had various hydraulically actuated fittings to tackle the larger pieces and smaller bits of debris. A core bore machine previously used to obtain samples from the core was placed back into service to bore holes in the resolidified mass to help break apart the previously molten reactor core. Combinations of tools were used to assist defuelling the lower reactor vessel region, such as the core bore machine and plasma arc torch (Figure 2.11). Using long-handled tools, operators used underwater television cameras to remotely load spent fuel canisters with fuel debris.

Figure 2.11. **Core boring machine**

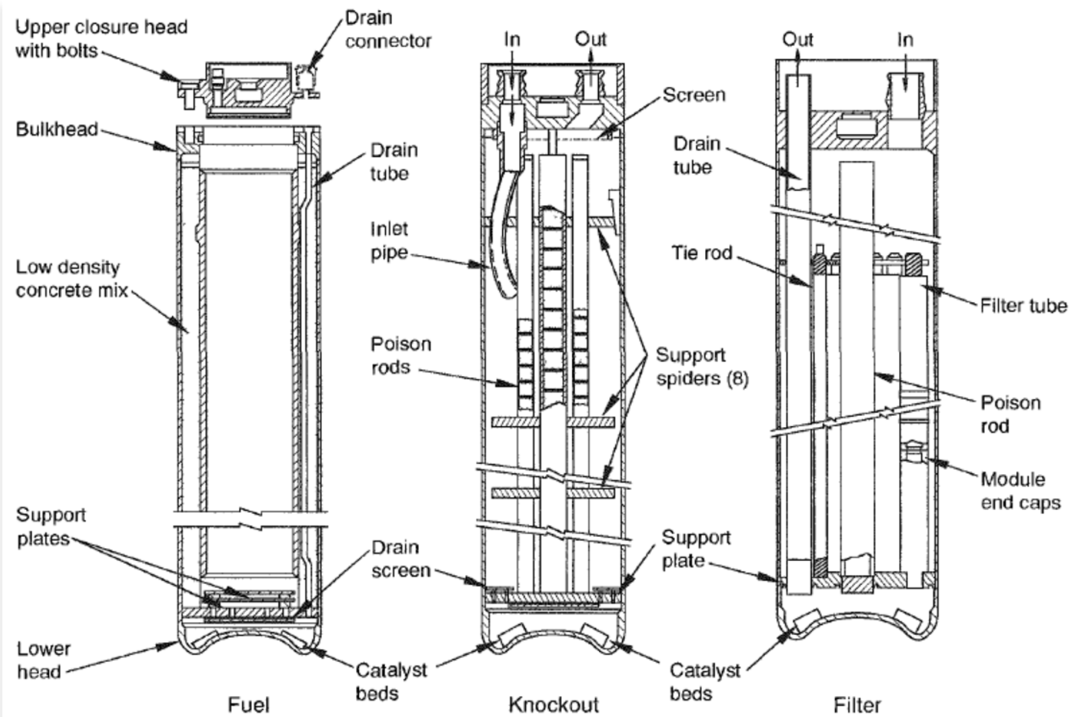


Source: NRC (2016).

The defuelling systems were designed before the extent of core damage and radiological conditions were fully understood. Three types of defuelling canisters (Figure 2.12) and associated support equipment were specially designed to remove the fuel debris from the reactor vessel and package it for transportation. Water clean-up systems were installed to ensure water clarity in the reactor vessel for removing debris and for processing defuelling canisters. The design of the canisters was decided upon early in the clean-up and dictated the design of the defuelling platform and the shipping cask. The design of the canisters was originally based on the removal of intact fuel assemblies, but later the length was reduced to fit inside the reactor vessel above the debris bed. Ultimately, very few if any intact fuel assemblies were removed. The narrow inside dimensions of the cylindrical canister design necessitated substantial cutting of distorted fuel debris in order to load through the opening of the canister.

A radiation analysis programme was undertaken to identify and quantify possible radiation sources and to design defuelling equipment to achieve dose rate goals in the defuelling area. The total occupational dose resulting from all clean-up activities was less than 62 person-Sv over the first 11 years following the accident. The cumulative occupational dose for defuelling and defuelling support activities was below 20 person-Sv. The exposure rate to defuelling workers averaged less than 0.1 mSv/h. Specific safety issues were addressed in NRC evaluations of defuelling equipment, systems, and operations.

Figure 2.12. **Canisters for fuel material, knockouts and filter tubes**



Source: NRC (2016).

The fuel canister was designed as a receptacle for large pieces of core material, which were picked up and placed either directly into the canisters, or into other containers which would then be inserted into the canister. The "knockout" canister was designed for use in the fuel debris vacuum system to separate debris particles ranging from about 140 µm up to full pellet size or larger. The filter canister was designed for use in the fuel debris vacuum system, the defuelling water clean-up system, and the canister dewatering system. The filter captured debris dust larger than 0.5 µm on sintered metal filters. Debris buckets were used to configure the debris before insertion into the fuel canister, to maximise the packing density in a canister. Two types of disposal buckets, the top loading debris bucket and side loading debris bucket, were designed to fit into the fuel canisters. An in-vessel vacuum system was designed to pick up smaller fuel debris and pass the same through knockout canisters. Any remaining debris was collected in filter canisters.

The canister positioning system, a rotating carousel installed in the reactor vessel, was used to hold up to five fuel and knockout canisters, including knockout canisters for use with the vacuum system. The height of canisters in the canister positioning system could be adjusted to three discrete elevations to allow them to be placed more closely to the debris bed as the bed got lower. The fuel handling building's canister handling bridge lifted the canister from the fuel transfer system to a submerged storage rack or to the canister dewatering station in the spent fuel pool (Figure 2.13). The defuelling canister storage racks provided storage for loaded defuelling canisters.

The existing fuel transfer system was modified to transfer defuelling canisters from inside the reactor building to the adjoining fuel handling building. Canisters were handled in a way similar to normal fuel assemblies during refuelling operations. The fuel transfer cask used to transfer a defuelling canister from the spent fuel pool to the shipping cask. The Model 125-B shipping cask was designed specifically to transport the loaded defuelling canisters.

Figure 2.13. **Canisters placed in a submerged storage rack in spent fuel pool**

Source: NRC (2016).

At the completion of defuelling, a report was developed by the licensee to document the measurements and calculations that were performed to ensure that the plant had been defueled to the extent reasonably achievable and that the potential for a nuclear criticality had been precluded during normal and accident conditions. The major objective of TMI-2 post-defuelling activities focused on preparing the plant for long-term storage. The licensee called the period preceding the ultimate disposition (either refurbishment and restart or decommissioning) of the plant "post-defuelling monitored storage" (PDMS). During PDMS, the TMI-2 facility would be in long-term monitored storage, similar to the decommissioning mode SAFSTOR (mothballing with delayed dismantling), in which the facility is secured, monitored, and maintained in a manner that ensures the protection of the public health and safety for an extended period.

Transitioning to PDMS required the following conditions: i) criticality was no longer possible; ii) potential for fission products movement was eliminated; iii) fuel was removed and shipped off-site; iv) radioactive waste was shipped or stored; v) radiation levels were reduced commensurately with the need for access to permit continued plant monitoring and to support plant-disposition decisions; vi) water was removed from plant systems and spaces, and the potential for reintroduction of water was precluded; and vii) a safe, monitored plant condition was established.

Disposal and long-term storage of fuel and other radioactive wastes

The major objective of TMI-2 post-defuelling activities focused on preparing the plant for long-term storage. The licensee called the period preceding the ultimate disposition of the plant PDMS. During PDMS, the TMI-2 facility was in long-term monitored storage, similar to the decommissioning mode SAFSTOR (mothballing with delayed dismantling), in which the facility was secured, monitored, and maintained in a manner that ensured the protection of the public health and safety for an extended period. Placing the TMI-2 facility in monitored storage was beneficial in that it eliminated any possible impact of TMI-2 decontamination and decommissioning efforts on the operating TMI-1 facility.

The TMI-2 accident and subsequent clean-up challenged the management of various forms and concentrations of radioactive waste. The management of highly contaminated water, fuel debris, and related solid waste by-products included handling, processing, temporary on-site storage, transportation, and final disposal. Decontamination activities resulted in substantial quantities of contaminated water and organic resins and inorganic zeolites produced from water-processing systems. Fuel debris that spread throughout the plant created unique radiological waste characteristics, and some had to be removed manually using chisels and other handheld tools (Figure 2.14). Also, some waste did not fit into established regulatory waste-classification categories for transportation and disposal.

Figure 2.14. **Manual debris clean-up operation**

Source: NRC (2016).

Several on-site facilities were constructed for temporary storage of solid radioactive waste products from clean-up activities that were being readied for transportation (Figure 2.15). Solid waste included spent EPICOR-I and EPICOR-II resin liners; contaminated clothing, tools, and equipment; and decontamination materials. The spent fuel canisters were stored in wet pools until technology was developed to dry the fuel; subsequent to repackaging and storing in dry storage casks.

Figure 2.15. **Temporary solid waste storage facility for radioactive wastes**

Source: NRC (2016).

The canister handling and preparation for shipment programme included all activities necessary to prepare and transfer a loaded defuelling canister from its storage rack in the spent fuel pool to the shipping cask; to insert the canister into the shipping cask; and to verify that the shipping cask was prepared for transport in accordance with its NRC-issued certificate of compliance. Canister preparations included dewatering and purging the defuelling canister with an inert cover gas, verification of final canister weights, verification that the catalytic recombiners installed inside the canister were functioning, and verification that the canister had been dewatered sufficiently to ensure that the catalytic recombiners remained operable regardless of canister orientation.

The canisters containing the fuel core debris were shipped in specially designed rail casks to a Department of Energy facility in Idaho for storage. The fuel casks (Model 125-B) consisted of five major components: the outer containment vessel, inner containment vessel, upper and lower canister impact limiters, canister shield plugs, and cask impact limiters (Figure 2.16). Each cask had its own transportation system of a skid and rail car. Gross shipping weight of the shipping cask was about 183 000 pounds (83 000 kg). Up to 21 defuelling canisters could be sent in a single rail shipment (seven canisters per cask, three casks per shipment). The fuel shipments began on July 1986, and were completed on May 1990. There were 22 rail shipments for a total of 342 canisters of core debris transported to the INEL. The final fuel shipment from Three Mile Island to the INEL started on 15 April 1990.

Figure 2.16. **Model 125-B rail shipping cask used for transporting defuelling canisters**

Source: NRC (2016).

Once a rail shipment arrived at the Central Facilities Area at INEL, both impact limiters were removed from each end of the shipping cask and the cask and skid were lifted from the rail car and placed on a tractor trailer. The cask was then transported to wet storage at the INEL Test Area North. Here each canister was removed from the shipping cask, filled with water, placed into a storage module, and transferred to the storage pool in a designated location. A vent tube was installed on each canister for continuous venting. The empty shipping cask was surveyed for contamination and prepared for rail shipment as regular freight back to TMI-2. The 342 stainless steel fuel canisters of core debris were stored in underwater storage from 1986 to 2001 at the INEL. During the 2000 to 2001 period, these canisters were transferred to the TMI-2 independent spent fuel storage installation (Figure 2.17), also located at INEL, for interim storage of the TMI-2 core debris.

Figure 2.17. **Independent spent fuel storage installation at INEL**

Source: NRC (2016).

Worker protection

Post-accident radiological conditions at TMI-2 were substantially different from those normally encountered at commercial operating nuclear plants because of the magnitude and specific mix of the radionuclide contamination. Radiation surveys made shortly after the accident showed that general area radiation readings ranged from 1.5 to 5 mSv/h in the fuel handling building, and 0.5 to 50 mSv/h in the auxiliary building. Hot spots were measured in the auxiliary building reaching up to 1.25 Sv/h, and exceeding 10 Sv/h in some cubicles. During the first entry into the reactor building in July 1980, dose rates at the 305-foot (93 m) entry-level elevation ranged from 4 to 6 mSv/h. Localised areas of high radiation were measured at 180 mSv/h over the open stairwell and 20 to 50 mSv/h at floor drains. Surveys performed during the second entry at the next-higher level, the 347-foot (106 m) operating-floor elevation, showed general radiation readings of 1 to 4 mSv/h. Below the 305-foot (93 m) entry-level elevation was the 282-foot (86 m) basement-level elevation, which was flooded with highly contaminated water and sludge. A telescoping radiation detector was inserted down through one of the reactor building stairwells and measured 400 to 450 mSv/h at 5 to 7 feet (1.5-2.1 m) from the surface of the basement water. Once the water was drained and processed through the submerged demineraliser system, dose rates from the remaining sludge ranged from 10 mSv/h to 10 Sv/h per hour, depending on location and distance from the floor.

A concern at TMI-2 was high-energy beta contamination from fission products in the reactor coolant. Areas in the auxiliary building that had experienced coolant leakage were measured in the 0.10 to 1 Sv/h gamma range with associated beta dose rates in the 1.0 to 100 Gy/h range. Similar gamma-to-beta ratios were measured on surfaces in the reactor building. The high-energy beta emitters present at TMI-2 required special radiological protection practices, such as monitoring equipment, personnel dosimetry, heavy protective clothing, procedures, and training.

Many programmes and activities helped to improve radiation protection practices at TMI-2. New approaches were needed in a number of basic worker protection and dose reduction areas, including protective clothing, respiratory protection, dosimetry, radiation field and contamination characterisation, exposure-tracking systems, dose reduction planning, procedures, training, and robotics. In the summer, when the temperature in the reactor building approached 33°C (90°F), an ice vest was initially used by workers to control heat stress and extended work period. An extensive heat stress programme and protocol was established that pioneered the use of cool suits and similar technologies. Also, an air conditioning system was installed inside the containment. Several respiratory protection breathing apparatuses were developed or adapted to extend stay-times in the reactor building, including a power air-purifying respirator and a power air-purifying hood. Some of the radiological protection programmes and activities are summarised below:

Dose reduction techniques included i) shortening the transit time of workers in the reactor building by opening both personnel airlocks and modifying the ingress/egress paths; ii) decontaminating by water flushing discrete radiation sources, such as the air coolers, elevator shaft, and enclosed stairwell; iii) eliminating other discrete radiation sources by removal of trash and contaminated equipment; and iv) placing shielding at the 305-foot elevation (93 m), such as lead curtains around the core flood tank, lead sheets on the covered floor hatch, and water columns and bladder shields around the open stairwell, elevator, and enclosed stairwell. Some of the more complex dose reduction activities included extensive pre-task planning and mock-up training for each task, decontamination of selected surfaces with chemicals, removal of paint, and scabbling (the mechanical removal of a thin layer) of concrete floors. These efforts resulted in significant reductions in the dose rate in the reactor building. In July 1984, workers entered the reactor building without respiratory protection for the first time since the accident, and, in accordance with ALARA principles, subsequent entries were made without respirators.

Eventually, the most effective reduction in the dose in the reactor building was the efforts that reduced dose to the defuelling crew on the defuelling platform. This included establishing a path to the work platform by shortening, cleaning, and shielding the path; and thoroughly reducing the dose in the area or the work platform where the defuelling operators worked. The dose reduction effort did not try to decontaminate the entire reactor building, but focused on areas where people worked. As the result, cumulative exposure of the defuelling campaign was roughly only one-third of the total exposure during the first 11 years following the accident.

Special instrumentation, systems, and techniques were developed or modified to measure and characterise the unique radiation situation at TMI-2 for ensuring worker safety and determining the effectiveness of decontamination processes. Key instruments included: i) thermo-luminescent dosimeter pseudo cores to take beta radiation measurements of the building floors; ii) wall and floor sampler to mill the concrete surface and collect the sample in a filter for off-site analyses; iii) modified handheld ion chamber detector to provide omnidirectional detection for gamma measurements; iv) modified handheld tungsten-shielded, Geiger-Mueller detector with a conical lead collimator on the face of the probe for rapid and accurate directional exposure measurements; and v) mobile radiochemistry laboratory to perform transuranic and radionuclide analyses of high activity liquid and solid samples.

In November 1982, upper-level corridors of the auxiliary building became accessible to workers without the need to wear anti-contamination clothing (respirators had not been required for entry into the upper corridors since October 1979). The following month, the auxiliary building's corridors in the basement (at the 281-foot elevation, 86 m) became accessible without the need to wear respiratory protection masks. Overhead areas, such as ceiling and cable trays, were decontaminated only to the extent that they would not re-contaminate the floor below. In such cases, a radiation work permit was required to access ceiling areas.

Although worker activities at TMI-2 have been quite different than those at operating power plants, the cumulative doses at TMI-2 since the accident had been lower than the average doses experienced at operating reactors. By the end of 1989, with the clean-up about 99% completed, the collective dose to all workers was 62 person-Sv. This was comparable to the collective occupational radiation exposure that was estimated in the original PEIS.

Insights and lessons learnt

Because the Three Mile Island accident was the first time that a large power plant experienced severe core damage much of the necessary equipment, procedures, and regulatory mechanisms to handle both the short-term and long-term aspects of the accident simply did not exist. Most of the equipment and procedures for long-term management activities had to be specifically designed and developed following the accident.

The TMI-2 clean-up effort took about 14 years, and the reactor vessel defuelling operations spanned a 5-year period. A total of about 133 000 kg of fuel, cladding, structural, and control materials were removed from the reactor vessel during the five-year effort. Unique systems and equipment were designed and installed to remove damaged fuel and structural debris from the reactor vessel. The clean-up activities challenged the management of various forms and concentrations of radioactive waste. The management of highly contaminated water, fuel debris, and related solid waste by-products included handling, processing, temporary on-site storage, transportation, and final disposal. Decontamination activities resulted in substantial quantities of contaminated water and organic resins and inorganic zeolites produced from water processing systems. Fuel debris that spread throughout the plant created unique radiological waste characteristics. Also, some waste did not fit into established regulatory waste classification categories for transportation and disposal.

At the completion of defuelling, it was shown that the plant had been defueled to the extent reasonably achievable and that the potential for a nuclear criticality had been precluded during normal and accident conditions. The major objective of TMI-2 post-defuelling activities focused on preparing the plant for long-term storage. The period preceding the ultimate disposition of the plant was called "post-defuelling monitored storage" (PDMS). During PDMS, the TMI-2 facility would be in long-term monitored storage.

Due to the unique situation and conditions, many activities related to the long-term management of TMI-2 accident were new, and new methods and processes had to be developed. The following insights can be mentioned.

- New processes and communication with the public
  - The DOE, in conjunction with other Federal agencies, implemented national policy affirmed by two US presidents, in support of activities related to the clean-up of the TMI-2 accident. The DOE activities, including acceptance and transport of the TMI-2 core debris, were frequently reviewed by Congressional committees through the process of testimony on technical progress and DOE budget authorisations. Accordingly, there was approval at the highest levels of government for the clean-up activity.
  - New regulatory practices were applied as changes in the facility's post-accident mode of operations required unique regulatory and licensing actions. In order to properly reflect evolving plant status, the NRC-issued orders, modified those orders, approved licence amendment requests, and granted relief from certain regulatory requirements. The NRC granted exemptions from certain requirements of the regulations for nuclear power plants, but only under special circumstances as permitted in the regulation. Exemptions were necessary at TMI-2 because of the plant's damaged configuration and changing status during clean-up.
  - The TMI-2 programme did a credible job in preparing a public relations plan before the campaign started. The programme accepted and enhanced all established public relations procedures, prepared and distributed programme briefs and videos, co-hosted a media day, made public announcements, performed pre-notification activities, and met with some state and public officials.
  - Gaining public trust was a significant challenge. In many cases it was more difficult to obtain public trust and acceptance than to find a technical solution. One case was the disposition of tritium-contaminated water. Although a technical solution could be found for the safe disposition of tritium-containing waste water it was more difficult to obtain public trust and acceptance for its disposal.

- Decontamination and management of radioactive waste and effluents
  - Management of large amounts of contaminated water was one of the major challenges. A large volume of accident-generated contaminated water accumulated in-plant tanks and containment basement floor and sump. Although most of the radioactive contaminants could be removed from water, tritium could not. This water was stored and used for plant decontamination purposes and shielding of equipment located in the spent fuel pool (no spent fuel at TMI-2). Eventually, this water was evaporated over a 2.5-year period. The evaporation was completed 13 years after the accident.

- Shortly after the core degradation, gaseous fission products collected in the reactor coolant in the liquid clean-up system caused serious airborne contamination inside the auxiliary building and required mitigative actions including diverting effluents into containment, designing new filtration and ventilation systems, and venting containment gases.
- Decontamination posed challenges as radioactive contamination penetrated many concrete and other porous surfaces requiring surface removal for decontamination. Eventually, with the various decontamination methods the contamination levels in the auxiliary building were reduced to those typical of pre-accident conditions.
- Recontamination limited decontamination efforts. Sources of recontamination included wastewater from flushing, migration of radioactive materials through epoxy coatings, and redistribution of radioactive particulates by the air handling system.
- All of the fuel and debris removed from accident was shipped off-site for federal sponsored research and temporary storage until a permanent waste disposal site is established. The TMI-2 site was decontaminated to the extent that any potential for a significant release of radioactive material to the environment was eliminated. Radioactive material was removed and other sources of radioactivity were isolated so that any potential radioactive release was to be within 10 CFR 50 Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criteria 'As Low as Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents", guidelines for off-site dose consequences. Radiation levels inside most areas of the plant were reduced to levels through decontamination to allow for safe entry for routine monitoring by survey teams. The water processing programmes removed all water containing forms of radiation to the extent practical. Aregular monitoring programme was established to ensure the continued safety of the plant in the post-defuelling monitored storage state.
- The shipping campaign of the debris from Three Mile Island to Idaho was a very successful project. It has been used as a model for transporting spent fuel since then. There were a number of things which helped the success of the shipping campaign. Keys to success were teamwork, attention to detail, emphasis on prevention of and/or early detection of problems, and a stringent quality assurance aimed at preventing problems.

- Risk management and occupational safety
  - Defuelling safety concerns included criticality, pyrophoricity, radioactive gas and dust production, radiolytically generated hydrogen and oxygen, and steam generation in-core debris canisters in the event of a fire. Defuelling operations took precedence over data collection requiring prioritisation in data to be collected.
  - Because the exact severity of core damage was unknown at the time and the operators had no way of measuring the criticality margin the control rods could not be relied upon for reactivity control and assurance of shutdown. Therefore, a high boron concentration had to be maintained in the coolant to ensure that the core would not become critical given the knowledge that was available. Boric acid was chosen from different potential poisons as part of criticality control but it also raised concerns of boron dilution events.
  - Potential pyrophoric of metallic zirconium (spontaneous combustion) was an early concern during core cutting operations. Evaluations concluded that the formation of zirconium powder during the accident was extremely unlikely due to the dynamics of the accident. In addition, testing of residue material from the reactor coolant system, control rod mechanism leadscrew, and reactor vessel plenum cover resulted in no pyrophoric reaction.
  - In-containment dose reduction was emphasised over decontamination for defuelling operations since early decontamination efforts proved to not be effective. Dose reduction techniques involved installation of shielding, choosing entry and transit routes based on dose, and use of decontamination techniques where necessary.

- The use of robotics to perform monitoring and decontamination reduced radiation exposure to workers. The use of robotics could also enable work to be performed in radiation fields that were too high for humans to practically work in.
- For defuelling, a radiation analysis programme was undertaken to identify and quantify possible radiation sources and to design defuelling equipment to achieve dose rate goals in the defuelling area. Training and decontamination work reduced radiation exposure to levels comparable with levels in the refuelling of a normal operating plant. This involved operators training to defuel using a full-scale mock-up to work out potential issues and to minimise time involved in actual defuelling. The mock-up was also used to test defuelling tools before being used in the containment. The mock-up included a working defuelling work platform, large water tank and a quarter section of the massive lower core support structure.
- Gaseous fission products which could lead to increase personal exposure was a concern during defuelling. An off-gas system created airflow that prevented radioactive gases from collecting under the work platform from reaching personnel working on the platform.
- More than 99% of the fuel and debris in the reactor vessel was removed to eliminate the potential for re-criticality.
- Heat stress was an issue for workers in protective gear especially in summer months.

Method and equipment development

- Special instrumentation, systems, and techniques were developed or modified to measure and characterise the unique radiation situation at TMI-2 for ensuring worker safety and determining the effectiveness of decontamination processes.
- As-built dimensions of the TMI-2 reactor vessel internals did not always match designed dimensions. This required inspections of the as-built dimensions to ensure that recovery equipment would work.
- To ensure that spatial material information about different core materials and fission products was obtained, DOE initiated a sampling programme using a core boring machine since this information would have been lost by the proposed defuelling processes.
- The core bore samples provided insights into fission product release from the fuel, fission product retention in the core, maximum temperature during the accident, and reactor core material interactions. The core bore machine later proved useful to break up the solidified monolith in the core region and drill through and cut portions of the lower core support assembly.
- It was difficult to obtain data about core conditions sufficient to define debris removal tooling requirements, Camera surveys were insufficient for this purpose. Grab samples were required. Additional tools were needed (handheld and pneumatic tools, core bore machine, plasma arc cutting torch) when unexpected conditions were encountered.
- The melt solidified into a heterogeneous mass of substantially-oxidised core constituents (fuel [uranium, zirconium], structural materials [Fe, Ni, Cr], and control rod materials [Ag, In, Cd]) that resisted efforts to break it apart. Having been unsuccessful with breaking the frozen mass apart with impact chisels and wedges, the core bore drilling machine that was originally used to retrieve bore samples from the core region was refitted to do so.
- The plasma arc cutting tool, which chosen over several other methods for cutting up components and material in-vessel, experienced some difficulties including frequent torch burnouts, difficult starting, loss of torch functionality due to borated water ingress requiring removal and reassembly, torch gases lifting debris into cutting tool positioning machinery requiring disassembly and cleaning of the machinery.

- The extent of degradation of the core was not known prior to defuelling. As such defuelling canisters were prepared for both intact fuel bundles and debris. It was difficult to estimate the number of defuelling canisters required because less mass was loaded into each canister than expected. Around 340 canisters were used. The initial estimate was for 240 canisters.
- Although using robotics for defuelling was considered, a manual “pick-and-place” method was chosen since it was more flexible and years were expected for the development of the proposed robotic system. An existing robotic arm was brought on-site but not used due to complexity in use, the potential for damage to controlling wires and hoses, and expected difficulties in decontamination.
- The hydraulic fluid in the defuelling hydraulic system was changed from one containing boric acid to one containing a borate ester mixture because boric acid precipitation was damaging hydraulic tools.
- The transport equipment interfaces, both at the Idaho National Laboratories and TMI-2, took a lot of planning, teamwork, and honest and open communication. The integrated test of all cask handling and cask loading equipment was very valuable for confirming cask-to-handling equipment fit up, training of clean-up personnel, development of procedures, and generally proving system performance. The test resulted in a much smoother installation and start-up of equipment at the TMI-2 facility.
- In order to keep the fuel rubble in a safe physical dimension (for criticality safety), a filler material needed to be used to ensure this physical configuration. It was decided to use a low density concrete for this filler material. The use of a low density concrete in the canister as filler material was not the best choice. The presence of water in the concrete mixture resulted in a surplus production of hydrogen that had to be vented. As such, the canisters could not be permanently sealed during subsequent storage.

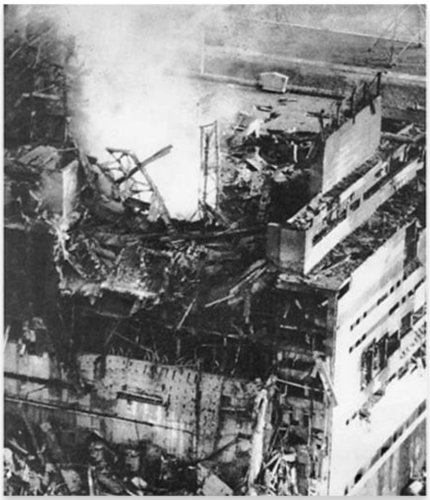
Synthesis of relevant Chernobyl experience

The accident at unit 4 of the Chernobyl NPP (Figure 2.18) at 01:23:40 of 26 April 1986 was one of the most serious accidents in the history of nuclear power. This reactivity initiated accident (NEA, 2010; De Geer et al., 2017) resulted in huge and rapid increase of neutron power that disrupted reactor and destroyed significant part of the reactor building (Figure 2.18a). Many other structures, safety barriers and safety systems were also affected.

Figure 2.18a. **Ruined Unit 4 after graphite fire ended**

Source: Krasnov et al. (2016).

Figure 2.18b. **Photo of the destroyed reactor in the first hours after the accident: April 26, 1986**

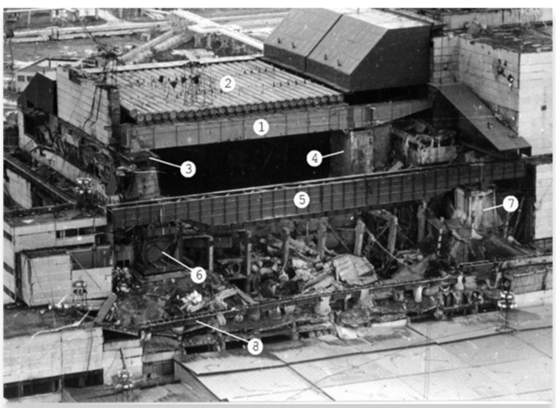


Source: Krasnov et al. (2016).

Figure 2.18c. **Photo made from helicopter on May 3, 1986**

Source: Krasnov et al. (2016).

Figure 2.18d. **Installation of supporting structures of the original confinement (shelter), view during shelter construction**



1) Girder B1; 2) Roll-up of pipes; 3) Reinforced upper part of the wall along the axis #50; 4) Ventilation shaft; 5) "Mammoth" girder; 6) West pillar of the "Mammoth" girder; 7) East pillar of the "Mammoth" girder; 8) "Octopus" girder.

Source: Krasnov et al. (2016).

Figure 2.18e. **External protective structures of shelter before new confinement installation**

1) Roof of reactor hall; 2) Roof of turbine hall; 3) Cascade wall; 4) West (large) counterfort wall; 5) South shields; 6) South "brassy" shields; 7) North "brassy" shields; 8) North (small) counterfort wall.

Source: Krasnov et al. (2016).

Figure 2.18f. **View of the new confinement that covered the old one on 29 November 2016**

Source: Krasnov, ISPNPP, http://dazv.gov.ua/images/pdf/ukraino-japonskyj-komitet/fifth/Viktor%20KRASNOV.pdf.

Extremely high releases of fission products, long-distance transport of aerosols and hot particles, and global contaminations of land were stimulated by reactor explosion and fine fuel fragmentation on 26 April 1986 as well as by the strong graphite fire which resulted in very high plume visible from 27 April to 9 May (Figure 2.18b and 2.18c).

A significant fraction of the nuclear fuel (3.5%) was released, most of which was deposited in the vicinity of the damaged plant. There is some discrepancy in element specific release fractions published in the literature (Abagyan et al., 1986; IAEA, 1986; Legasov et al., 1988; Lewis, 1986; llyin and Pavlovskij, 1988a, 1988b). Recent published data (Borovoy and Velikhov, 2012) indicate the following releases:

- noble gases: 100%;
- iodine radionuclides, including ^{131}I: (50-70)%;
- ^{137}Cs and ^{134}Cs: (33 ± 10)%;
- low volatile nuclides, such as ^{90}Sr and ^{144}Ce, and non-volatile nuclides, such as ^{239}Pu and ^{241}Am, in the dispersed fuel – up to 5%.

In the long term, ^{137}Cs and ^{90}Sr are the most hazardous nuclides contributing to the contamination of land. Thousands of square kilometres of land were affected with the highest contamination of Cs deposited by rainfalls.

Contamination by dispersed fuel worsened the radiological situation on the site during the early phase. Besides ^{137}Cs (γ-emitter) and ^{90}Sr (β-emitter), the most hazardous isotopes over the first 10 to 20 years after the accident are Pu isotopes (α-emitters) and then ^{241}Am (^{241}Pu transforms into ^{241}Am by β-decay).

About 30 years have passed since the accident and work on the mitigation of its consequences has been performed continuously over the years. Many aspects rendered the long-term mitigation of Chernobyl accident highly challenging, specifically during the emergency phase right after the accident. They were:

- High on-site contamination levels due to fuel fragments and dust dispersal. Even if the fuel fraction released from the damaged unit was limited, the high specific activities of fragments and dust resulted in very high dose rates.

- Highly damaged structures and systems due to explosions and fires, complete loss of the means of surveillance.
- Facing a completely new situation, absence of adequate protection devices, remote devices for cleaning the site, systems for removing rubble and finally adequate vehicles to transport waste materials.
- Difficulties to assess existing risks (radiological, criticality, thermal [i.e. corium-concrete interaction]) with the knowledge existing at that time.

Characterisation of dose rates, contaminations and releases

According to Borovoy and Velikhov (2012), there was a lack of reliable information about dose rate at the site during the first hours after the accident because:

- The operating staff had no appropriate dosimeters. They had dosimeters with an upper measurement limit of 3.6 R/hr (0.01 mSv/s). Dosimeters with wider measurement ranges were locked in a room that operators were not able to open.
- The NPP top management was reluctant to accept and broadcast that a severe accident had occurred in unit 4. During the night of 26 April, the plant director reported to authorities: "Reactor survived; we continue water cooling; the radiation level is at the normal level". In fact, the results of the radiation measurements carried out by the head of rescue team in the vicinity of unit 4 indicated the upper limit of 200 R/hr (0.556 mSv/s) of the DP-5 dosimeter he was using. Early morning of the next day, the director reported to the authorities that radiation was at the level of 3-6 R/hr (0.0083-0.0167 mSv/s) and that the reactor survived.

This miss-information resulted initially not only in a wrong understanding of the danger posed by the accident to the public but also in deaths of several people who tried to inspect the unit either on their own initiatives or at the request of the management of the plant.

The accurate in-site radiation monitoring started at 14h00 on 27 April, i.e. about 13 hours after the beginning of the Chernobyl accident. A team of external experts came to Pripyat, the town where the plant personnel lived. The first dose maps prepared with participation of military staff showed dose rates between 100 and 1 000 R/hr (0.278 and 2.778 mSv/s) at different locations close to unit 4. Twenty-nine dosimeter units were established on 29 April in order to improve radiation monitoring. Later 36 dosimeter stations were created under the supervision of the Soviet Army.

Extensive efforts were made during the active phase of the accident to sample the air above the reactor and in the vicinity of the site. For many reasons (unstable nature of the releases, changes in weather, high radiation fields, work undergoing around the damaged unit), the accuracy of the measurements was not high. However, these samplings provided a significant result: besides the volatile fission products releases (noble gases, iodine, caesium and tellurium isotopes), most other nuclides (niobium, zirconium, ruthenium, cerium) were also released as small solid particles.

Efforts were also made during the active phase of the accident to measure depositions of fission products around the site. Although the dose rates could be easily measured, the contribution of γ, β and α emitters could not be easily determined as these required complex radio-chemical analyses. However, it was established readily that the main part of the fuel particle fallout occurred in the vicinity of the damaged plant and that the on-site contamination was essentially due to those fuel particles.

Studies were conducted to determine the amount of fuel remaining in the damaged unit. Based on many measurements and samplings, it was concluded that approximately 3.5% of the fuel was released and deposited around the damaged plant with 0.3% on the site and 1.5% within the exclusion zone. Even though it was a small fraction of the fuel inventory, fuel particles, pieces of assemblies and fuel pins on the site were the main contributors to the measured dose.

Extensive off-site measurements were organised by the government with participation of 1 130 institutions of Ukraine. Fifteen thousand dosimeter stations and 94 mobile laboratories were created (Kachalovsky, 1989). The measurements obtained from this extensive network enabled:

- Decision making for evacuation of people from the city of Pripyat, from a 10 km zone and finally from a 30 km zone, where land surface contamination by Cs, Sr and Pu exceeded 15, 3 and 0.1 Ci/km^2 (555 000, 111 000 and 3 700 Bq/m^2) respectively. With these criteria for evacuation, 49 360 people were evacuated from Pripyat on 27 April 1986, 10 799 people from the 10 km zone on 2 May and 3 and 30 136 people from the 30 km zone between 4 and 7 May. Altogether, almost 100 000 people from 49 settlements were evacuated in 1986. Between 1989 and 1991 additional evacuation of people from 11 settlements was performed: the criterion used was that of a dose limit of 350 mSv for the whole life of a person.
- Planning and realisation of the radiation decontamination work.
- Determination of the protection measures in the so-called "controlled zones" which would be much less contaminated. Such controlled zones were created in 77 settlements were people continue to live.

More details about the on-site and off-site radiation measurement results and data assessments may be found in IAEA, 2006; Izrael et al., 1990; Rimsky-Korsakov et al., 2009; Ivanov et al., 1994; Kononovich et al., 1994; Borovoi, 1989; Devell et al., 1986; Kashparov, 1994; Borovoy, 1996b; Izrael, 1998.

Countermeasures undertaken in the short term and their effects

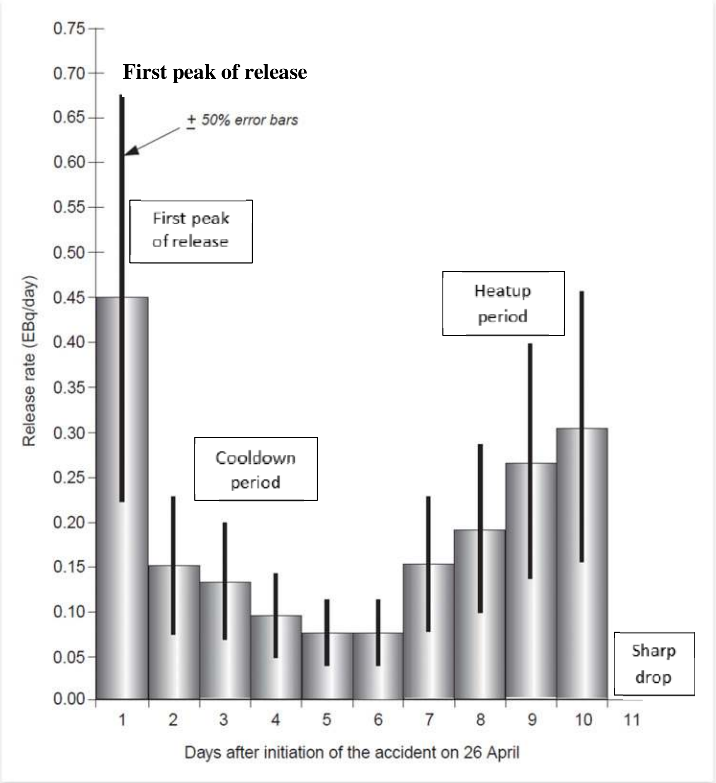
Implementing adequate countermeasures during the emergency phase was found to be particularly difficult since it was not possible to understand the situation and to assess the efficacy of countermeasures.

In the immediate days following the accident, authorities largely feared further uncontrolled escalation of the accident. They therefore focused on firefighting, the prevention of re-criticality and of the penetration of molten corium into the pressure suppression pool, located below the reactor cavity in this particulate design. It was feared that melt could penetrate further through the basement into the soil, to pose the risk of groundwater contamination. Many countermeasures were implemented to reduce these risks. They resulted in high exposure of radiation to personal involved in the countermeasure operations.

The large radioactive releases occurred from 27 April to 10 May 1986 (Figure 2.19). During that period, a large amount of different materials was dumped from helicopters on the top of the destroyed reactor. A total of 5 000 tonnes of materials were dumped until 10 May (Krupny, 1996). These included 40 tonnes of B_4C to reduce re-criticality risk, 800 tonnes of dolomite to stop the graphite fire, 1 800 tonnes of clays and sand on top of the debris to reduce radioactive aerosols emission and 2 400 tonnes of lead to cool down the molten fuel. Approximately 14 000 tonnes of solid materials were dumped in April-May 1986. Besides these, 140 tonnes of liquid polymers and 2 500 tonnes of tri-sodium phosphate were dumped in the open top of the damaged reactor to reduce formation of radioactive dusts.

As it is shown in Pasukhin (1997), and as it is common in almost all helicopter drops, only a small fraction of these materials reached the target. However, it helped to stop the graphite fire, which was maintaining high temperatures for the release of the volatile fission products from damaged core. The environmental releases and dose rates above the debris and in the vicinity of reactor building were dramatically reduced. Perhaps natural phenomena, such as reduction in decay heat with time, natural cooling of debris and the reduction in the amount of graphite available for burning in the hot zone, also helped to decrease the radioactivity releases (Figure 2.19).

Figure 2.19. **Kinetics of integrated daily radioactive release during the initial phase of the accident**



Notes: Estimated uncertainty is ± 50%. Activity is normalised for the radioactive decay at 6 May. 1Ebq (Exa-Becquerel) = 1 0^{18} Bq.

Source: Adapted from OECD (2002).

Dumping of large masses of solid material from high altitude had also several negative effects (Kluchnikov et al., 2011). It damaged roof plates of turbine hall and supporting structures of de-aerator vessel. It also created additional radioactive dust, which deposited in the north of the destroyed reactor building (Petelin et al., 2003) and formed a new area for decontamination.

The risk of steam explosion after melt penetration into the compartments, which were originally filled with water, was found to be low, as it was determined that these compartments were almost dry by the time of corium melt arrival.

It was decided to construct a protective concrete slab below the reactor building basement to reduce the risk of groundwater contamination. Its construction started on 3 May 1986. A total of 388 mine-workers were involved in the construction of a tunnel of 138 m length and 1.8 m diameter which was used for radiation protection of workers and concrete slab building. Extraction of soil and pouring of ~2.5 m thick concrete slab having lateral dimensions of ~30 × 30 m was accomplished. A water cooling system made of a layer of 100 mm diameter tubes and a temperature control system were installed inside the concrete slab to prevent structure ablation by the core melt. This work was completed by the end of June 1986. Later on, it was seen through boreholes that only one concrete horizontal wall, out of the four in the reactor building on the path of the corium melt/debris had been eroded by the melt.

Construction work was performed to provide passages and the site was cleaned before the mitigation measures were started. This phase was very tough since no adequate protection for workers and equipment was available to reduce radiation exposure to acceptable levels. The situation was continually improved by installing protection and remote devices. The site clean-up consisted of: i) removing rubble and damaged equipment, ii) decontamination of external surfaces of buildings, iii) removal of the upper layer of soil (5 to 10 cm), iv) laying concrete slabs and clean sand and gravel on the ground and v) covering the surfaces by films. Thanks to these measures, the site dose rate was reduced to 5-10% of the pre-existing dose by 10 May.

At the end of May 1986 additional on- and off-site measures were undertaken for further site decontamination and preparation of the restart of Chernobyl units 1 and 2. These included construction of decontamination facilities, preparation of temporary storage places for radioactive wastes, protection of rivers and lakes, construction of new roads and supply systems, and construction of the isolation walls between unit 4 and unit 3 (Kochetkov et al., 1989).

Later on, extensive construction work was performed to prevent the contamination of the Pripyat and Dniepr rivers, on whose banks about 32 million people lived. The most critical times for possible contamination of these rivers were the autumn of 1986 and the spring of 1987. The Soviet Government constructed a total of 131 filtering and isolating dams (a) in the lower part of Pripyat River (b) along the Dniepr River and (c) along the Kiev artificial lake. A total of 5 million tonnes of soil were employed in these construction projects. In addition, four bottom traps and five underwater dams were also built. These measures reduced contamination of hydrosphere (Kachalovsky, 1989) to about 15% of the pre-existing level.

An additional measure (Borovoy and Velikhov, 2013) for protection of rivers and lakes was building of the ground wall around the NPP site, i.e. leak-tight wall in the soil having design thickness of ~1 m, depth of 30 m in some places and length of 8.5 km, to isolate the hydrosphere from the contaminated groundwater. However, the works have been stopped after confirmation of no melting through of the fundament plate of unit 4 and new evidences of insignificant contamination of Pripyat River with the groundwater. The main contamination source was proven to be surface washing and surface contaminated waters. Only 2.8 km of wall were built along the river. Later, it was found that this wall noticeably increased the natural level of groundwater at the site.

The above-listed projects resulted in quite high exposures for the civil and military personnel engaged in these projects. Altogether 6 000 persons participated in this urgent (day and night) decontamination and emergency work. Two thousand of the six thousand persons reached their radiation dose limit of 25 rem (250 mSv) (Nosovsky et al., 2006) before the beginning of September and they were replaced. Extremely high collective dose of 22 877 person-rem (228.77 person-Sv) is reported by Kochetkov et al. (1989) during the 8 months of 1986, but in 1987 and 1988 it was reduced, respectively, to 7 242 and 4 953 person-rem (72.42 and 49.53 person-Sv). The number of persons receiving the 5 rem (50 mSv) dose limit was reduced by a factor of 12 between 1986 and 1987. Almost a factor of 47 reduction is reported for the exposures above 25 rem (250 mSv) during the same time period.

Fuel debris bed characterisation and isolation

One of the major concerns during the first days after the accident was the possible re-criticality of the debris bed. Measurements of the ratios of $^{134}I/^{131}I$ activity were carried out in parallel to dumping boron carbide into the remainder of the core material contained in the core cavity in the reactor. These continuously repeated measurements indicated no formation of short living iodine isotopes, which confirmed sub-criticality of the debris.

The second urgent task was to identify the mass of the fuel and activities of fission products, which remained in the debris bed, as well as the properties of the fuel (corium) debris, in order to predict further accident progression and to provide necessary mitigation measures.

Up to 1988, the dose measurements were taken mainly at the periphery of the reactor. It was imperative to find methods and means of approaching the places where fuel had accumulated, in order to examine potentially dangerous locations and to collect fuel and aerosol samples.

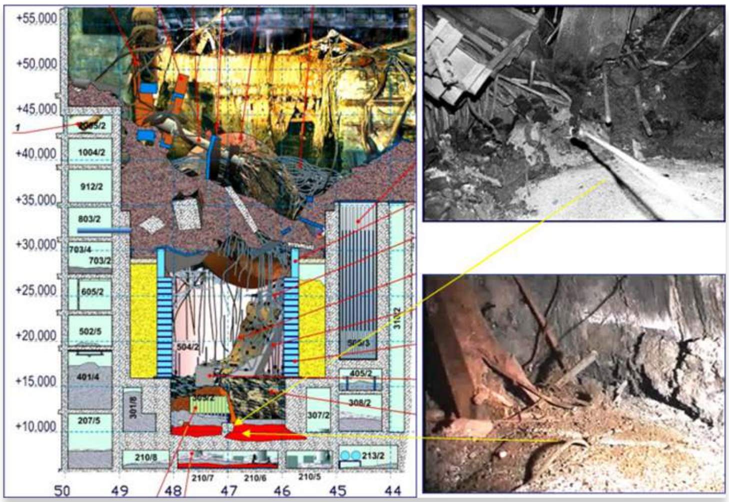
During the summer of 1986, preliminary results on fuel distribution were obtained based on the external sample studies showing that more than 96% of reactor fuel remained in the debris bed, i.e. less than 4% of the fuel released during the accident. Fuel release of 3.5% was reported to the IAEA (Lewis, 1986). Later, after more complex studies it was concluded that more than 95%, i.e. more than 180 t of the fuel inventory, was in the debris bed (Borovoy and Velikhov, 2012).

Studies of highly contaminated compartments and sampling of surface contamination, aerosols and fuel debris were started. These remote measurements and samplings were performed through the existing pipes, e.g. inlet and outlet collectors of control rod cooling system, and through boreholes drilled from the accessible and decontaminated rooms. Visual, photographic and video-observations were collected. Radiation and thermal measurements performed to detect the location and the radioactivity of fuel and to obtain more accurate information on the condition of the engineering constructions. More than 40 holes were bored and a unique procedure for extraction of highly active core samples was developed. Many boreholes were drilled and later used for installation of gamma and neutron detectors to obtain information on corium re-criticality control, as well as measurements obtained from TCs and other sensors (Belyaev, Borovoj and Gagarinskij, 1989). Temperature and radiation measurements complemented by the detailed sample studies enabled primary detection of the locations (Figure 2.20) and masses/activities of fuel debris and lavas formed during molten corium-concrete interaction (MCCI).

Immediately after the accident, the decision was made to construct a confinement building around the damaged unit to prevent further releases and to provide some shielding against gamma-radiation. This construction, called the sarcophagus or shelter (Figures 2.18e, 2.18f and 2.18g), used to the maximum extent the remaining structures of the damaged unit. Most part of the fuel is contained in the sarcophagus in the form of active fragments scattered and deposited during the accident, as remnants of fuel in-core channels or fuel assemblies in southern spent fuel pool (SFP), as of finely dispersed fuel particles deposited on surfaces in the sarcophagus and as solidified lava. Some amount of U and Pu is also found in water accumulated in the sarcophagus.

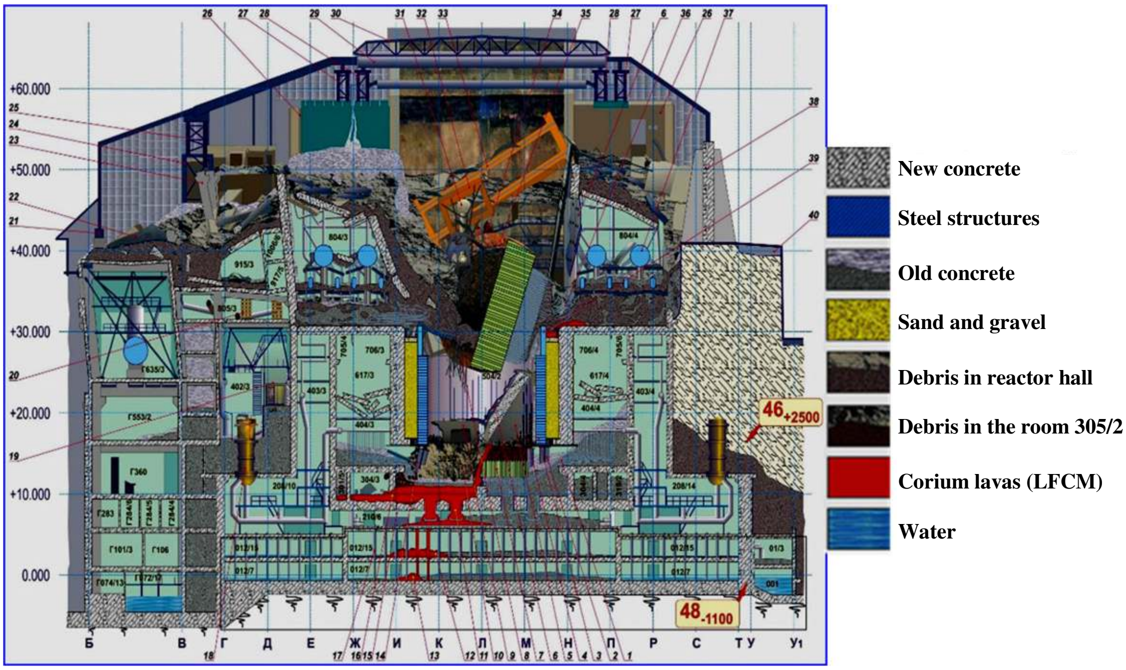
Design and construction of the sarcophagus (Figures 2.18d, 2.18e and 2.21) were completed extremely fast – in three and in six months respectively. However, the strength of the structure supporting the sarcophagus was not known at that time and the built confinement was not leak tight. Uncontrolled water penetration/leakages and aerosol releases took place and affected the near-environment. The safety of the sarcophagus was of prime concern for the long-term management of the highly radioactive materials on the site of Chernobyl plant. Sarcophagus failure hazards were increasing with time (Figure 2.18e) and after 30 years, i.e. in November 2016, the original sarcophagus building was completely covered by a more reliable and leak-tight confinement, designed for 100 years. This new confinement was designed by an international team of Architects and Nuclear Engineers. With the new confinement (Figure 2.18h), provisions were set to start the dismantling of the sarcophagus and the recovery of the damaged fuel and different wastes. The safety of these actions should now become a prime objective of the long-term management of the Chernobyl site.

Figure 2.20. **Left: location of Chernobyl lavas according to first measurements (in red), top right: photo of LFCM cluster in room 305/2, bottom right: photo of LFCM covered by concrete**



Source: Krasnov et al. (2016).

Figure 2.21. **Section of shelter (sarcophagus) isolating the destroyed unit 4**



1 - pipes of bottom water mains (BWM); 2 - northern additional support; 3 – water tank of containment - scheme «L»; 4 – peripheral range of cooling channels; 5 - inclined ferroconcrete slab (a fragment of separator box wall); 6 - metal cladding of heat protection of separator box; 7 - scheme «OR»; 8 – fuel channels in reactor vault; 9 - ferroconcrete structure; 10 - wall of «loose» FCM; 11 - steam-discharging valve; 12 - LFCM cluster in pressure suppression pool No 2 (PSP-2); 13 - LFCM cluster in pressure suppression pool No 1 (PSP-1); 14 - southern additional support; 15 – through fusion penetration in sub-reactor slab; 16 – fusion penetration in room 304/3; 17 - steam-discharging pipe; 18 – main circulating pump; 19 - inclined gallery («Dosifeyev stair»); 20 – tie cladding; 21 - «Octopus» beam; 22 – lead sheets; 23 - inclined columns; 24 - western support of «Mammoth» beam; 25 - «Mammoth» beam; 26 - wall on axis 50; 27 – block of B1 beams; 28 – block of B2 beams; 29 - pipe sheathing; 30 – light roofing; 31 – reloading machine (RM); 32 – RM carriage; 33 – RM bridge; 34 – diagnostic buoy; 35 - upper metal structure of reactor - scheme «E»; 36 - pipes of upper steam-water mains (SWM); 37 - steam-separator; 38 - northern buttress wall; 39 – water tank of containment - scheme «D»; 40 - cascade wall.

Source: Krasnov et al. (2016).

Activities associated with the safety of the sarcophagus

Characterisation of fuel-containing materials

Most part of the reactor fuel and fission products is contained in the sarcophagus in the following forms:

- fragments of fuel, fuel rods, assemblies and core structures;
- aerosols, hot particles and dusts;
- corium (lavas or lava-type fuel-containing materials [LFCM]);
- water solutions and suspensions.

Chernobyl lavas were formed during interactions of molten fuel with reactor structural materials and concrete (Table 2.1), but also with unknown fractions of materials dumped from helicopters on top of the debris during first days of the accident. Their formation and properties were influenced by the decay heat and the heat of Zr and graphite oxidation reactions. Different kinds of lavas (see Tables 2.2, 2.3 and Figure 2.22) formed at different temperatures/locations, were sampled and studied in Kluchnikov et al. (2011), Arutyunyan et al. (2010), Borovoy and Velikhov (2012), and Krasnov and Khan (2015).

Note that sampling of lavas was done several times manually (!) during the first days and later using robots, but majority of samples were obtained through boreholes drilled from decontaminated rooms between 1988 and 1991 (Figure 2.23). Diameter of boreholes was in the range of 60 to 150 mm and the lengths were up to 26 m. Main fraction of the drillings was done from the compartments located in the west part and a smaller fraction from the south to the north. Some of the boreholes were drilled with an inclination. The boreholes were used for:

- estimation of internal damage of reactor and equipment;
- determination of the main locations of lavas and material sampling (Figure 2.24);
- installation of different instrumentations, including cameras, temperature sensors, γ- and neutron detectors.

Later, installed instruments were used to develop a complex radiation and safety monitoring systems, such as "finish" (Borovoy and Velikhov, 2012) and others.

Table 2.1. **Estimated masses of materials within the reactor location and masses involved in Chernobyl lavas**

Material	Masses within the reactor location* after the accident, tonnes	Masses of the materials in lavas, tonnes
Fuel (U)	120	90
Steel	1 300**	< 20***
Serpentinite	580	160
Concrete of under-reactor plate	Minor	130
Concrete of reactor building	950	480
Sand of lateral reactor shield	300	280
Zr	No data	45
Graphite	750	Minor

Note: * In the compartments #504/2 (reactor pit) and #305/2 (under reactor pit); ** Non-melted materials are excluded; *** 330 tonnes of steel melted and spread separately of lavas.

Source: Adapted from Borovoy and Velikhov (2012).

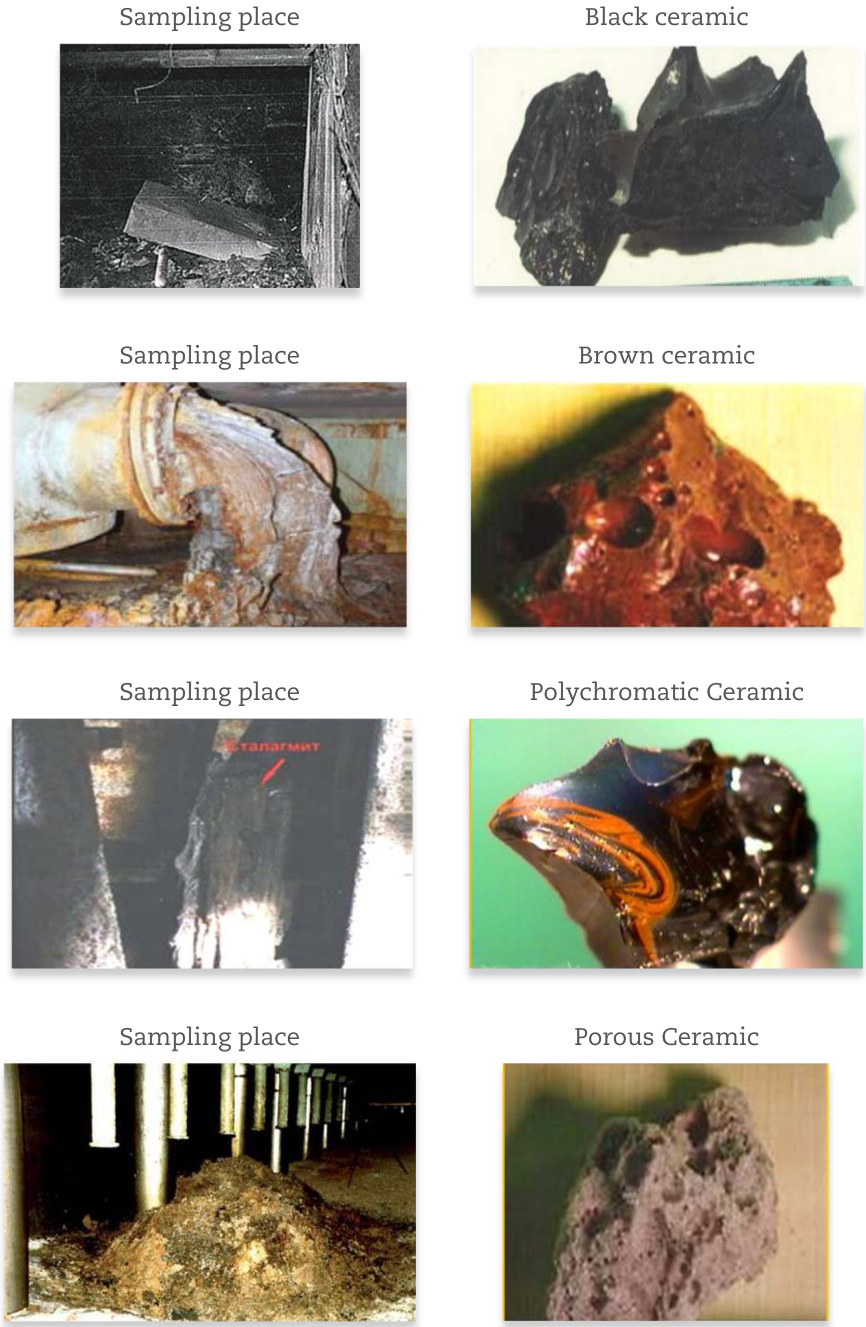
Table 2.2. **Masses of fuel and corium (LFCM) in different shelter compartments**

Compartment (number)	Fuel-containing materials type	Mass of fuel estimated for 06.09.2010, t (U)	Comments
Central hall (914/2)	Fuel, LFCM	More than 22	With consideration of 48 fresh fuel assemblies (5,5 t)
Southern spent fuel pool (505/3)	129 spent fuel assemblies	14.8	LFCM may be present
All top rooms, including central hall (CH) (mark +24.000 and above)	Fuel and dust	~5 on blockage surface in CH, ~totaly 30	Estimated 30 t include surface contamination inside blockage in CH and all other rooms
304/3	LFCM	6 ± 2	«Horizontal lava flow»
301/5+301/6+303/3	LFCM	5 ± 2.5	«Horizontal lava flow»
217/2	LFCM	0.4 ± 0.2	«Elephant foot», «stalactites»
Sub-reactor 305/2 and 504/2 (to +24.0)	Fuel, LFCM, dust	80 ± 30	Estimated for 7 FCM clusters. Start of all LFCM flows. FCM in a break through between rooms 304/3 and 305/2
SDC (210/5+210/6+210/7)	LFCM	12 ± 6	«Large vertical flow» and «small vertical flow»
PSP2 – (012/14+012/15+012/16)	LFCM	3-12	
PSP1 – (012/5+012/6+012/7)	LFCM	1.0 ± 0.5	
Fuel under cascade wall	Fuel, dust	(0.9 ± 0.3)	
Water in all rooms	Soluble uranium salts, slurry	0.004	
Fuel on "shelter" object site	Fuel, dust, fragments	0.75 ± 0.25	

Note: See compartment and room locations in Figures 2.23 and 2.24.

Source: Klyuchnikov et al. (2011).

Figure 2.22. **Different types of corium (fuel-containing materials) detected in the Chernobyl shelter**



Source: Krasnov et al, (2016).

Chemical (Table 2.3) and phase compositions of lavas are very complex but majority of the lavas are alumino-silicate glass-ceramics with abundant inclusions and pores. Differences between black and brown ceramics in uranium inventory are visible in Table 2.3. Macro- and microstructures of lava are not uniform, see examples in Figure 2.26, and typically represent large inventory of phases. Main part of the reactor fuel formed $(U,Zr)O_{2-x}$ or $(Zr,U)O_{2-x}$ type solid solutions with tetragonal and monoclinic structures, UO_2, U-rich zircon and a smaller part of U is dissolved in the other phases.

Table 2.3. **Averaged oxide content of Chernobyl lavas**

LFCM type	Main oxides, mass. %													
	SiO_2	Al_2O_3	Fe_2O_3	FeO	MgO	CaO	Na_2O	TiO_2	ZrO_2	BaO	UO_2	MnO	Cr_2O_3	NiO
Black LFCM 304/3	70.6	7.4	0.25	0.23	3.9	6.7	6.2	0.21	5.8	0.13	4.3	1.9	0.30	$1.2 \cdot 10^{-3}$
Black LFCM 217/2	66.6	8.7	0.40	0.36	3.8	8.5	5.6	0.27	5.8	0.15	5.0	3.8	0.33	0.19
Black LFCM 210/6	62.1	7.2	2.91	2.63	5.1	6.0	5.2	0.19	5.5	0.18	5.8	0.40	0.40	0.39
Brown LFCM 210/7	64.0	6.8	0.64	0.57	7.0	6.7	5.4	0.24	6.6	0.19	9.4	0.53	0.39	0.36

Source: Klyuchnikov (2015).

Table 2.4. **Masses of dusts in different compartments of shelter**

Place (room)	Square, m^2	Volume,m^3	Dust amount, t of U	Uncertainty, t
2005/2	100	250	0.080	0,010
2014/3	27	140	0.022	0,003
2014/4	27	140	0.022	0,003
1001/2	54	150	0.043	0,005
1001/4	54	145	0.043	0,005
1002/2	35	95	0.028	0,004
1003/3	105	290	0.084	0,011
1003/4	105	290	0.084	0,011
1004/2	330	900	0.260	0,033
1005/2	35	300	0.028	0,004
1011/1	21	70	0.017	0,002
1011/2	21	70	0.017	0,002
902/2	17	15	0.014	0,002
903/2	39	100	0.031	0,004
904/3	23	70	0.018	0,002
2005/2	100	250	0.080	0,010
2014/3	27	140	0.022	0,003
904/4	23	30	0.018	0,002
905/3	30	110	0.024	0,003
905/4	30	110	0.024	0,003
906/2	28	90	0.022	0,003
907/2	32	95	0.026	0,003
909/9-10	26	70	0.021	0,003
909/11-12	26	80	0.021	0,003
909/13-16	72	150	0.058	0,007
910/2	205	400	0.160	0,021
911/2	12	20	0.010	0,001
912/2	81	230	0.065	0,008
915/3	210	700	0.170	0,021
916/3	14	35	0.011	0,001
916/4	14	35	0.011	0,001
918/2	17	45	0.014	0,002
919/5-6	50	140	0.040	0,005
919/5-8	50	140	0.040	0,005
926/3	31	130	0.025	0,003
926/4	31	130	0.025	0,003
935/1	4.5	20	0.004	0,000
935/2	4.5	20	0.004	0,000
804/3	470	7 000	0.380	0,047
804/4	470	6 000	0.380	0,047
612/2	80	900	0.064	0,008
505/4	40	500	0.032	0,004
219/2	63	1 850	0.050	0,006
061/2	45	1 000	0.036	0,005
305/2,504/2,404/3, 404/4	740	5 300	0.590	0,074
Total			3.115	0.391

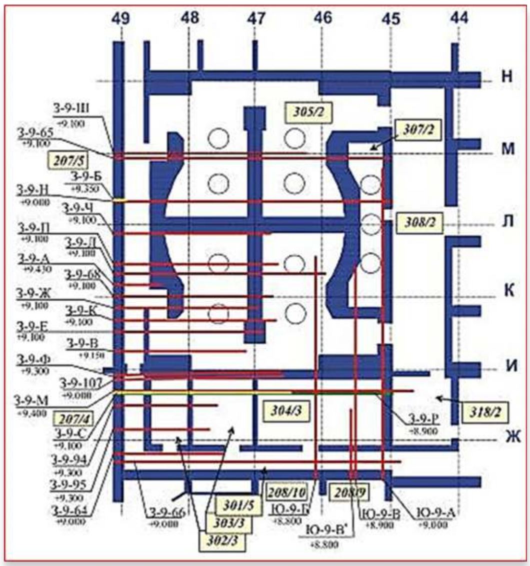
Note: see locations of some compartments and rooms in Figures 2.23 and 2.24; example of isotope specific activities of dust given in Table 2.5.
Source: Krasnov et al. (2016).

Table 2.5. **Average radionuclides contents in fuel particles**

Nuclide	^{90}Sr	^{90}Y	^{137}Cs	^{238}Pu
Activity, Bq/g (U) (01.01.2015)	$5.97 \cdot 10^8$	$5.97 \cdot 10^8$	$7.23 \cdot 10^9$	$6.19 \cdot 10^6$
Nuclide	^{239}Pu	^{240}Pu	^{241}Am	^{241}Pu
Activity, Bq/g (U) (01.01.2015)	$5.00 \cdot 10^6$	$8.18 \cdot 10^6$	$2.41 \cdot 10^7$	$2.41 \cdot 10^8$

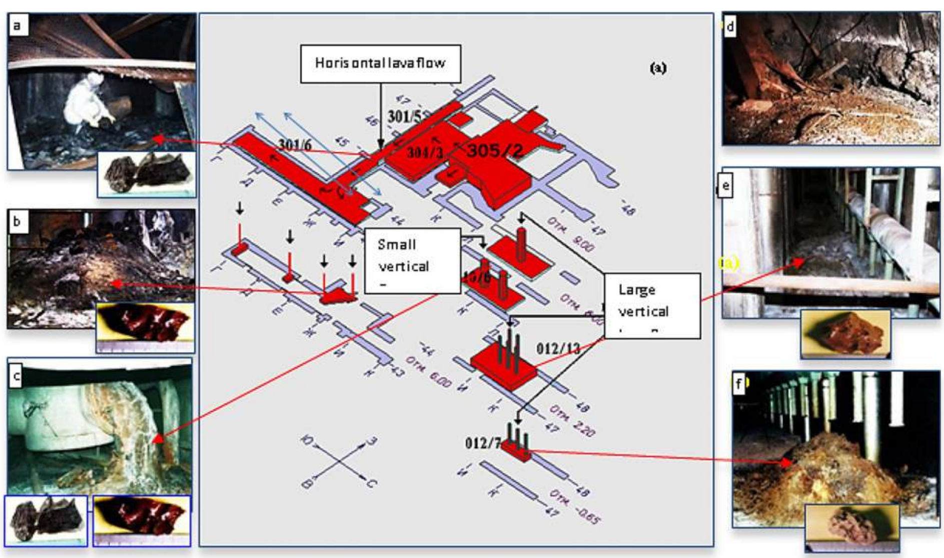
Source: Krasnov et al. (2016).

Figure 2.23. **Scheme of horizontal and inclined boreholes drilled for sampling and installation of instrumentations**



Source: Krasnov et al. (2016).

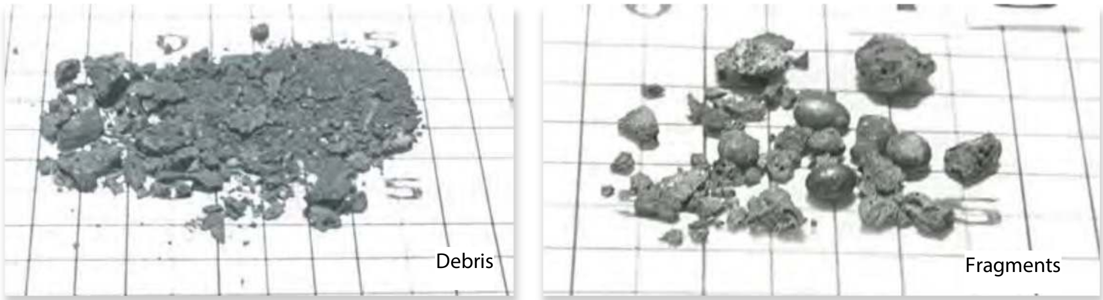
Figure 2.24. **Scheme of melt flows and locations of lavas in different compartments**



(a) Black ceramic in the room #304; (b) "Elephant foot" in the room 217/2; (c) Brown and black ceramics in steam distribution corridor 210/6; (d) Lavas with high U content in the room 305/2; (e) Brown ceramics in the pressure suppression pool #2; (f) Brown ceramics in the pressure suppression pool #2.

Source: Krasnov et al. (2016).

Figure 2.25. **Typical views of samples retrieved through the boreholes**

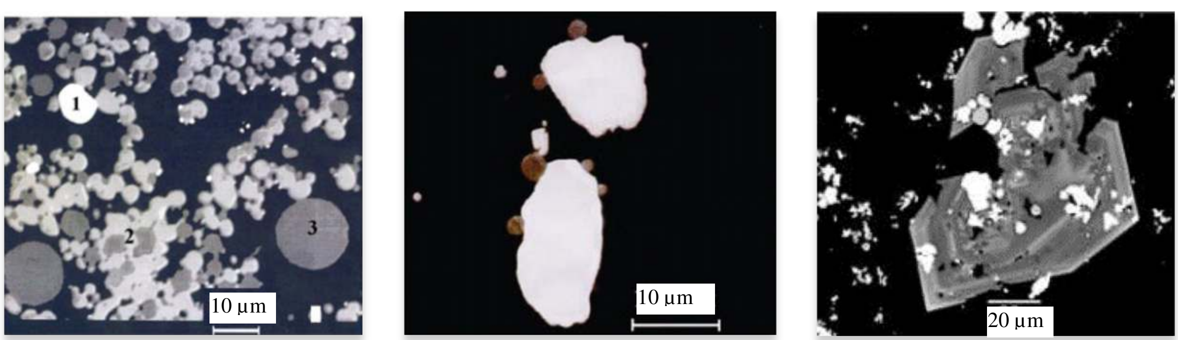


Source: Borovoy et al. (2012).

U-rich zircon was attributed as a new technogenic mineral: Chernobylite (Figure 2.26 right), i.e. a crystalline zirconium silicate (Anderson et al., 1993a) $(Zr_{1-x}U_x)SiO_4$, which contains 6-12 mass % of U. Natural crystalline zircon with such a high U content is unknown. Possible formation mechanism is suggested as isomorphous inclusion of U into the mineral structure after replacing Zr. A study (Shiryaev et al., 2016) concluded that at least some zircon crystals rotated during last stages of the lava solidification, indicating that they were formed at temperatures significantly higher than solidus temperatures of lavas.

Microstructure and other properties of the Chernobyl lavas were extensively studied, in particular, by Anderson et al. (1992, 1993b), Borovoy et al. (1990), Pazukhin (1994, 2002, 2006, 2008), and Pazukhin et al. (2002), but some important features related to the prediction of their long-term behaviour remain poorly studied.

Figure 2.26. **Examples of (Left) Brown Ceramic, (Centre) Black Ceramic and (Right) Chernobylite microstructures (scale given in μm)**



1. Uranium oxide (white); 2. Inclusions of (Zr, U)Ox (grey and light brown); 3. Globules of steel.

Large uranium oxide inclusions (white) and small globules of steel (grey)..

BSE image of several inter-grown skeletal zircon crystals with abundant urania inclusions. Note that some UO_2 dendrites cut zircons growth zones.

Source: Arutyunyan (2010).

- Criticality control of fuel and corium in the shelter (Sarcophagus)

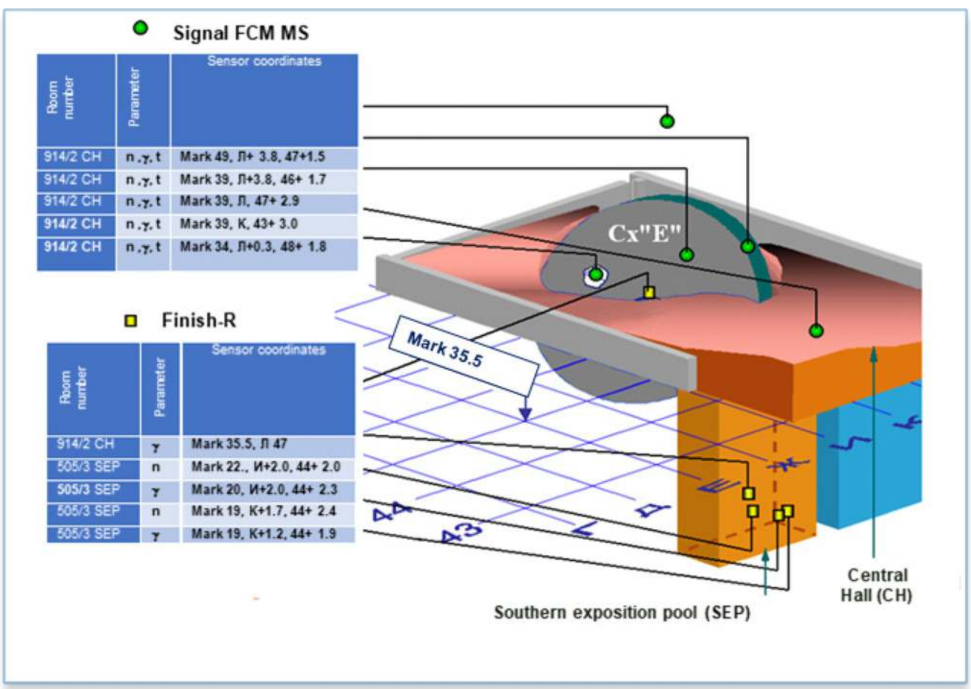
The first attempts of measurements of the multiplication factor in the shelter were made in 1991 (Lebedev and Shikalov, 1995). The measurements were performed using neutron pulse method, which is one of the active methods, along with the standard geophysical equipment of neutron logging of the bores. Unfortunately, it was not possible to vary the frequency of the neutron pulse emission and the time of registration of the system response was limited to 2 ms. Consequently, the delayed neutrons were not measured and the precision of the multiplication factor was not sufficient.

New monitoring systems, such as “finish”, “pilot” and “KSFCM” were constructed in the shelter using neutron, γ and temperature detectors which helped to obtain the measurements of neutron flux density in many locations. Data accumulated before 1990 confirmed conclusions, reached earlier, on the technical justification of the criticality safety of shelter (Belyaev et al., 1990). The K_{eff} measured by active methods was below 0.7 and the values measured by passive methods were below 0.4, which confirmed that at that time the risk of re-criticality was insignificant. However, it was indicated in this and in the following study (Borovoy et al., 1996a) that the barriers which prevented water penetration into lava after the accident (elevated temperatures, glassy-like lava surface, boron and gadolinium added solutions) were becoming less efficient with time: lava temperatures had decreased, glass-like structures were slowly disintegrating, making them water-permeable, boron and gadolinium concentrations in water were decreasing in time.

Several incidents in 1990 and 1996, which may indicate re-criticality in the shelter, were reported (Borovoy and Velikhov, 2012). The first one happened after the rainy summer of 1990. The readings of neutron detector installed on the surface of lava in the room 304/3 (Figure 2.23) through the borehole increased approx. 60 times (Figure 2.27). In the context of detector shift from its original position as one of the possible reasons of the counting rate increase, the detector position was visually inspected with a periscope at the beginning of the count rate increase: on 27.09.90 (before the time window shown in Figure 2.27). No changes were detected in comparison to the previous view. On the same day the detector and the system channel were checked with ^{252}Cf neutron source (~$1 \cdot 10^7$ neutron/s) temporarily installed in the room. Response of the measurement system was according to the expectations and there appeared to be no problems with the measurements.

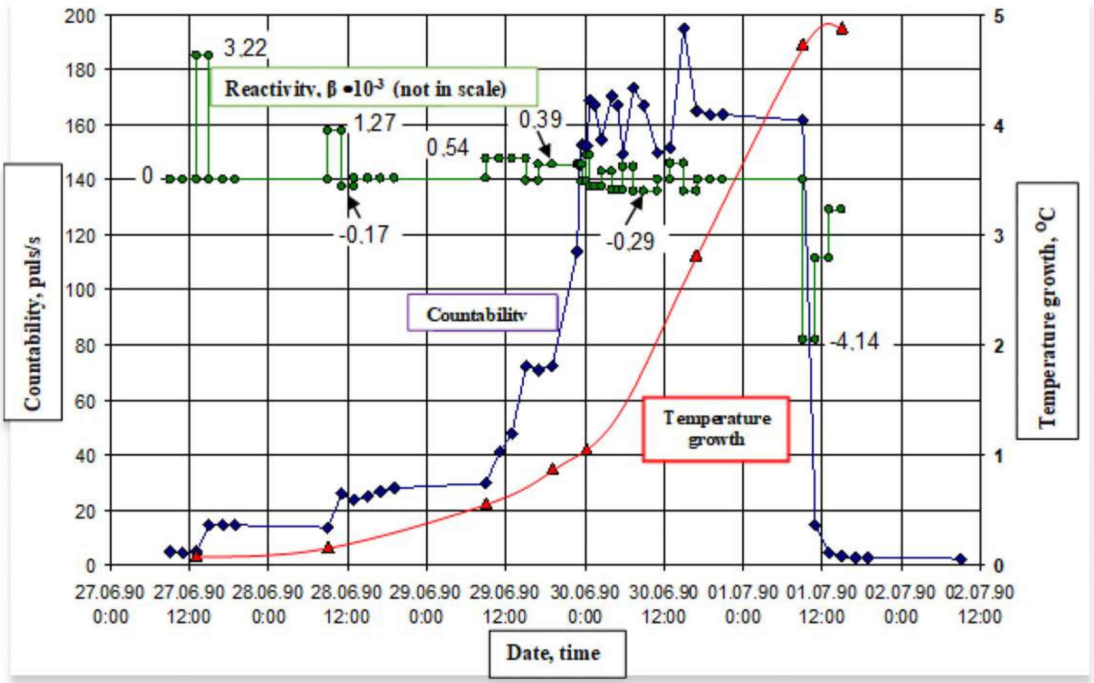
After the readings reached 160 count/s, gadolinium nitrate solution was injected in two portions of 80 l each into the west and east parts of the room. First portion reduced the readings rapidly to 30 count/s. Slow further reduction to 2.4 count/s was observed after the 2nd portion was injected during the day.

Figure 2.27a. **Locations of sensors of “Finish-R” and “Signal” systems for monitoring nuclear safety of fuel-containing materials**



Source: Krasnov et al. (2016).

Figure 2.27b. **Neutron detector readings (count/s) in the room # 304/3 from the morning of 29.06.90 to the evening of 30.06.90**



Source: Krasnov et al. (2016).

Three times smaller increases of counting rate occurred on 12 and 16.09.1996. They happened after the rainy weather also. Studies performed after these data were gathered and showed that the neutron count rate correlates with the water ingress into the shelter.

Many calculations and analyses of the incidents were performed with reactor and Monte Carlo codes, but there are still different opinions about the nature of the incidents. Authors (Frolov, 1996) concluded that water penetration into corium debris zone enriched with fuel resulted in reactivity increase with a rate of 10^{-4} β/centigrade and that re-criticality of this zone was due to the delayed neutrons at $K_{eff} \approx 1.000009$. Alternative reason of reported neutron anomaly can be connected with possible increase of neutron detector efficiency/sensitivity caused by softening of neutron spectrum due to water moderation, i.e. with the inaccuracies of the measurements.

- Studies of debris and lava ageing

Lava ageing can potentially increase the risk of fission product release due to leaching and lava dispersion. Several phenomena can be responsible for Chernobyl lava ageing, such as matrix leaching by water and material interactions with atmospheric gases/vapours, self-irradiation, thermomechanical cracking and fragmentation.

Intensive studies of long-term ageing of natural lavas having chemical and phase composition similar to Chernobyl lavas, which can be classified as alumino-silicate glass-ceramics with abundant inclusions and pores, of borosilicate glasses and 30-year-old lava samples and aerosols formed in different compartments provided new information reviewed by Shiryaev et al. (2016), which is useful for assessment of long-term behaviour of the lavas.

The closest to Chernobyl lavas in silica inventory are natural lavas from Mt. Cameroon volcano in Africa, which is very high and has a wide range of variation of atmospheric conditions, such as humidity and temperature. According to the long-term observations, the characteristic thickness of the alteration crust at the lower part of the volcanic mountain does not exceed 10 cm in 100 years, but tephra and pumice can be altered at much higher rates, presumably due to much higher surface area. These observations confirm importance of leaching and material interactions with atmospheric gases/vapours.

Investigations of various types of nuclear glasses have shown significant effects on glass performance at accumulated doses exceeding ~10^{18} α-decays/g and/or ~10^{9} Gy. In the case of Chernobyl lava, assuming a homogeneous distribution of α-emitters, such doses can be accumulated in ~10^{4} years. If lava structures are analogous to nuclear glasses self-irradiation would not be an extremely severe problem for LFCMs (Pazukhin et al., 2002). However, macro- and micro-structural inhomogeneities, e.g. concentration of α-emitters in aggregates and inclusions, and variety of properties of lavas in different locations are detected. Also, LFCM structure is different and can be less stable in comparison with the accurately synthesised amorphous borosilicate glasses. Thus, sensitivity of lavas to the internal radiation can be much higher. Nevertheless, the direct SEM/EDX studies of several samples collected long ago, stored in dry conditions and recently examined by Pazukhin et al. (2002), did not reveal catastrophic cracking, fragmentation and powdering. Only some cracking is reported around the large inclusions of UO_2 and Zr-U-O phase.

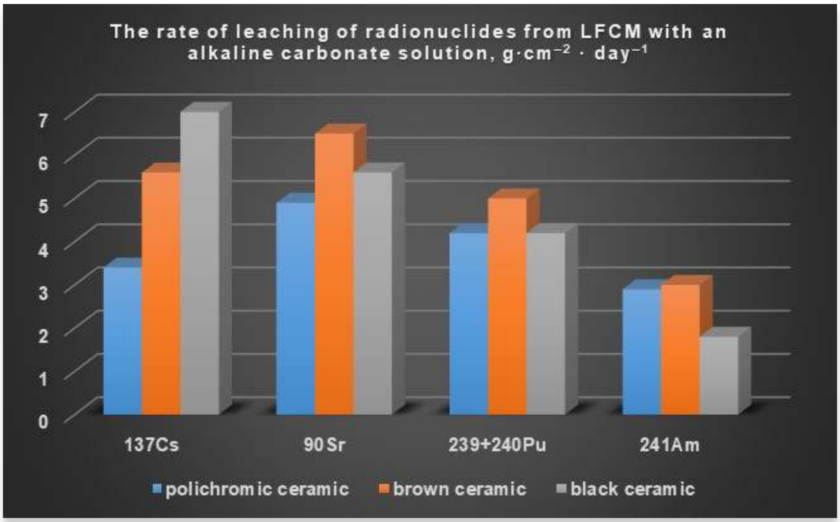
Further, temperature gradients in the lavas can reach several tens of K and can be different at different locations. Such thermal gradients can result in limited thermal stresses. Thermal cycling of the lava samples in laboratory fatigue tests demonstrated remarkable stability of the bulk material even in the presence of solid inclusions. However, high thermal stresses may have occurred during solidification of lavas. These thermal stresses can increase the risk of mechanical failure and fragmentation.

Another possible mechanism of ceramic fragmentation, which is very typical for porous natural rocks of Nordic countries, can be associated with water penetration into open micro and macroporosity and subsequent freezing of water at low temperatures. Such phenomena are limited by the decay heat of the lavas having noticeable content of fuel and fission products. It was found in the experiments (ISTC, 2000) with LFCM thermal cycling in the range of (-20)-(+50)°C and parallel microscopy of micro-cracks that such extreme conditions do not cause significant propagation of the micro-cracks. It was explained by the water molecular interaction with silicate clusters at the micro-level with formation of bounded water which cannot form ice. However, as it was confirmed by visual monitoring of lavas in the shelter, water freezing can cause significant damage of macrostructure of the lava when water penetrates into the large pores.

According to Borovoy and Velikhov (2013), the main phenomena responsible for the lava destruction and fragmentation/powdering are leaching and material interactions with atmospheric and technogenic gases/vapours. The most affected and fragmented currently are black and porous ceramics (Figure 2.22). Note that the black ceramic has the highest contents of fuel and fission products. The most resistant are brown and polychromatic ceramics (Figure 2.22). The damage of different ceramics is controlled not only by the ceramic properties, but specific location and configuration of lavas allowing or preventing continuous interaction of the ceramic with water. Nevertheless, spontaneous fragmentation of several specimens collected in the shelter and kept in storage without access of water was observed after 20 years of storage in Kurchatov and Radium Institutes.

Leaching rates of different nuclides from Chernobyl lavas by a water solution having compositions typical for the shelter differ for various lavas as shown in Figure 2.28 (ISP NPP, 2007).

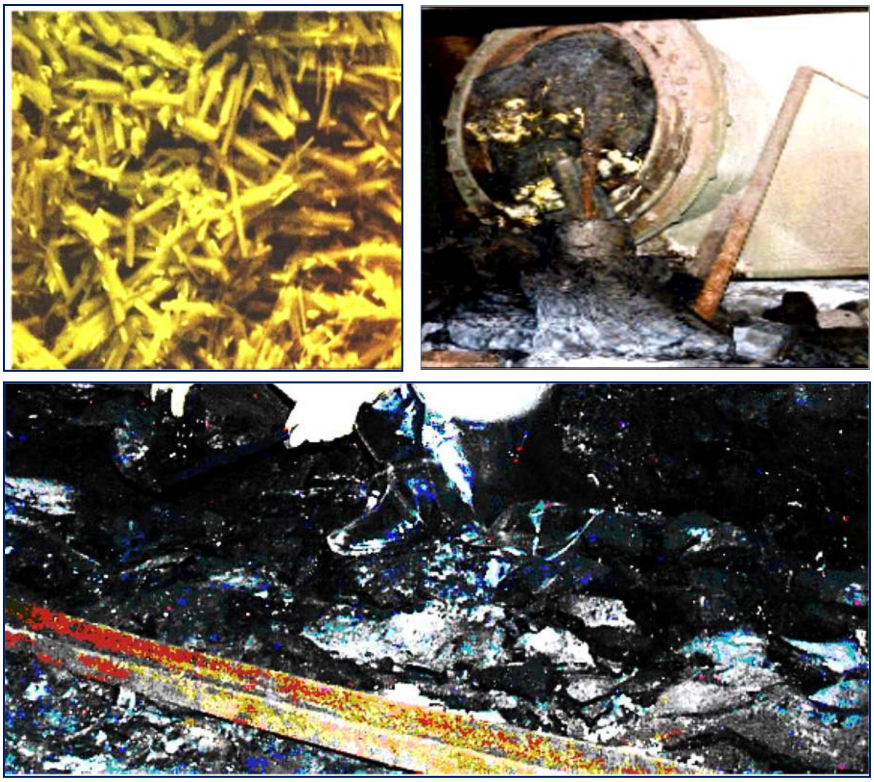
Figure 2.28. **Leaching rates of different radionuclides from polychromic ceramic, brown ceramic and black ceramic**



Note: The damages to the surface layer of lavas in the shelter can be seen visually in some cases, e.g. presenting dark green or yellow colour surface structures. An example is shown in Figure 2.29. It contains the following phases: (UO2) CO – Rutherfordine; UO3 • 4H2O – Epijanthinite; UO3 • 16CO2 • 1.91H2O; UO4 • 4H2O; Na4 (UO2) (CO3)3; Na4UO2 (CO3)3 • 3Na2CO3.

Source: Krasnov et al. (2016).

Figure 2.29. **New formations on LFCM surfaces: (top left) in steam release corridor (top right) in steam discharging valve (bottom)black ceramics**



Source: Krasnov et al (2016).

It may be concluded that there are no doubts about ageing, degradation of properties and spontaneous fragmentation of lavas but the degradation rate and related radiological hazards remain uncertain. Consequently, at the current level of knowledge these phenomena cannot be quantified or even well predicted. As it is shown by Shiryaev et al. (2016), spontaneous fragmentation and powdering of lavas create secondary releases of fission products, both with aerosol particles and water solutions/suspensions. Ageing issues should be addressed very seriously in the safety analyses and management of the unit 4 of Chernobyl NPP.

Spent fuel pools

There are two spent fuel pools in the 4th unit of Chernobyl: the North and the south pools. The North SFP was empty by the time of the accident; the south pool contained 129 spent fuel assemblies corresponding to 14.8 t of uranium. The south spent fuel pool has a size of 10.6 (length) x 4.2 (width) x 18 m (height). All fuel assemblies were contained in cylindrical casks.

Five boreholes were drilled in the SFP walls to check in 1990 the status of fuel assemblies located in the pool. The periscope observations showed that the pool was dry and that its lid was destroyed, but materials from the reactor hall did not relocate into the pool. The casks were at their original positions but debris was visible at the bottom of the pool through the gaps between the casks. Dose rate close to this debris was at the level of 5 000 R/hr (13.89 mSv/s). The absorbers installed in the pool for criticality control during normal operation were also visible.

Dose measurements through the boreholes indicated that the dose rate in the upper part of the SFP decreased from 3 500 R/h (9.72 mSv/s) in 1990 to 370 R/h (1.03 mSv/s) in 2013 but the neutron flux in the lower part increased from 140 to 400 n/(sm^{2}s) during the same time. This suggests possible fuel relocation to the pool bottom, e.g. due to fuel overheat and degradation in dry conditions.

Water management in the sarcophagus

Water management in the sarcophagus is considered critical since water can:

- fragment fuel-containing material increasing the amount of mobile radioactive particles inside the sarcophagus;
- cause an increase of the criticality of fuel-containing material in the course of time, as their cooling and fragmentation proceeds (with generation of nuclear hazardous compositions);
- contribute to the weakening of building structures;
- lead to groundwater contamination with radionuclides.

The water balance has not been understood completely but its most important contributions have been identified.

Before implementation of the new confinement in 2016 (Figure 2.18h), the following sources of water in the shelter were found (with a total flow rate of approximately 1 600-2 000 t/year before 2004):

- Leakages of atmospheric depositions (rain and snow) through the roof and walls. Between the years 2004 and 2008 extensive work on shelter reconstruction was completed, including hydro isolation of the building and renovation of its roof. More than 600 m^2 of roof were replaced. These measures practically eliminated rain and snow penetration into the shelter.
- Condensation of humidity of air inside the shelter, which contributes with approximately 600 t/year but with visible variations between the years because of the weather differences.
- Operation of the system for dust suppression employing spray nozzles under the roof, which supply 300-400 t/year before 2004 and approximately 100 t/year after it.

The contaminated water releases from the shelter are due to:

- Leakages into the sump water system of the unit 3, as detected by the trace method with sodium bromide. This water, which is classified as liquid wastes, is accumulated in the sump tank and periodically pumped for reprocessing. The flow rates are between 580 and 1 090 t/year (Krasnov and Khan, 2015).
- Leakages of water into the soil and mixing with groundwater. The flow rate is unknown. Systematic monitoring of groundwater contamination at the site has been conducted since the year 1992. Currently, approximately 50 boreholes are integrated in the survey grid. In the boreholes, the groundwater levels are measured, gamma-spectrometry logging is conducted in monitoring mode, and groundwater sampling is made.
- Evaporation and release into atmosphere with air (~380 м^3/year). The source term associated with the water evaporation is unknown but is estimated to be quite low.

The water volumes accumulated in the shelter also differ from year to year. In 1996, 3 000 m^3 of water was reported in the shelter, in 2007 – 331 m^3, in 2008 – 322 m^3, in 2009 – 318 m^3 and in 2010 – 337 m^3. The water volume has remained nearly constant in the last few years.

The water is mainly contaminated with Cs isotopes, ^{90}Sr, U and Pu (Table 2.6). Significant increase of Pu concentration in water, but also U concentration in water sampled from several locations, is visible from the comparison of data for the years 2000 and 2010. It confirms disintegration and fragmentation of lavas in the shelter. Data reflecting increasing trend of Pu and U leaching are marked in the Table 2.6 with the grey background.

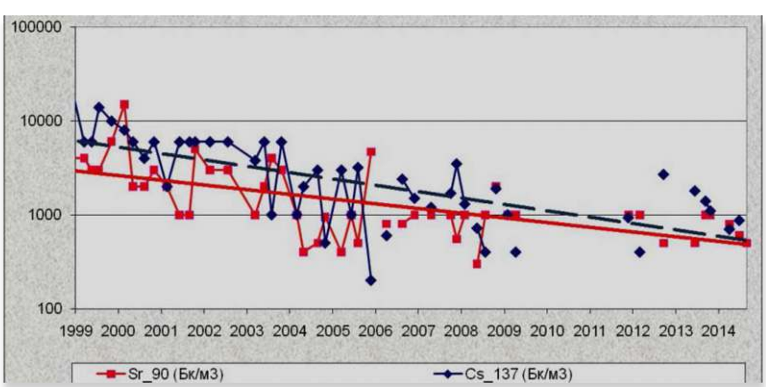
Table 2.6. **Average yearly data on contamination of water in the shelter**

Sample place #	Isotope specific activity, Bq/m^3						Concentration of U, mg/l	
	^{137}Cs		^{90}Sr		Integral Pu			
	2000	2010	2000	2010	2000	2010	2000	2010
6	6.2 ·10^{10}	3.6· 10^{10}	1.0 10^{10}	5.4 ·10^{9}	4.0 10^{6}	6.8 10^{6}	48	31
20	1.2 ·10^{10}	3.9 10^{10}	4.2· 10^{8}	1.1 10^{10}	2.7 10^{5}	1.3 10^{7}	1.7	42
30	5.2 ·10^{9}	8.1 ·10^{9}		1.7 10^{9}	3.6· 10^{5}	2.7 10^{6}	2.7	8.1
32	1.3 ·10^{11}	3.9 10^{10}	2.2 10^{10}	5.5 10^{10}	4.2 10^{6}	1.6 10^{7}	109	34

Source: Krasnov et al. (2016).

Radionuclide releases with water can be associated with dissolved compounds and suspensions of solid particles. Content of solid particles is at the level of (0.02 ± 0.6) g/l. The dynamics of groundwater contamination by Cs and Sr measured in a borehole close to the shelter is shown in Figure 2.30. There is a clear tendency towards the reduction of Cs and Sr concentrations in groundwater.

Figure 2.30. **Cs and Sr specific activities in groundwater (Bq/m^3) near the shelter**



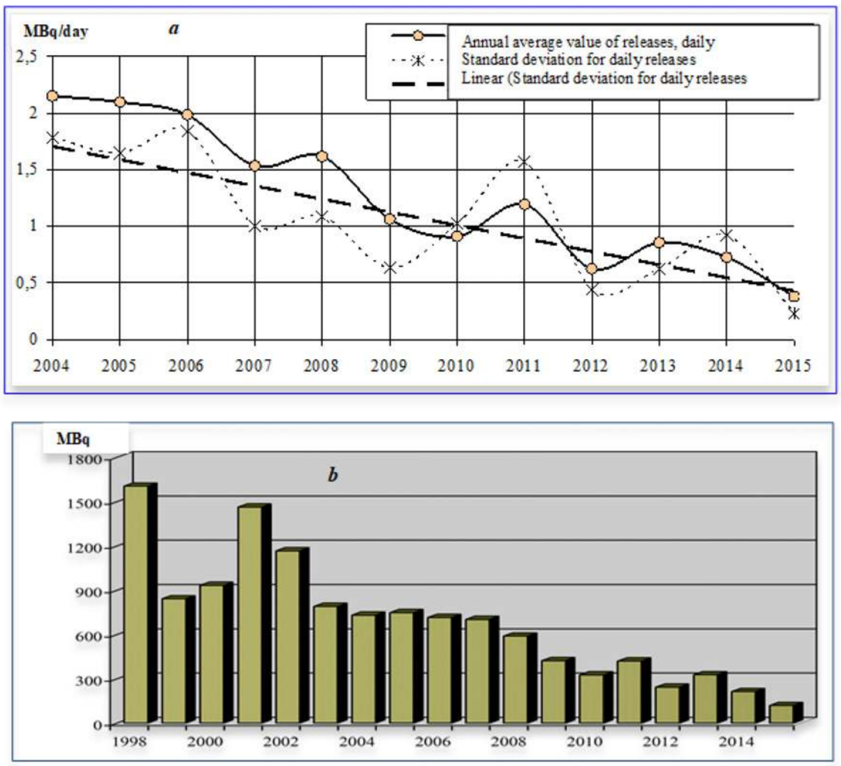
Source: Krasnov et al. (2016)

Countermeasures related to the sarcophagus radiation safety

Surveillance means were implemented for radioactive aerosol releases out of the sarcophagus and for water inside the sarcophagus before implementation of the new confinement.

A dust-suppressing installation using water solutions sprays was implemented in the central hall to reduce radioactive aerosols concentrations. This implementation was very effective in reducing airborne aerosols concentrations but water was sprayed and added inside the sarcophagus. Many other technical measures reported earlier enabled significant suppression of the aerosol source term (Figure 2.31) for the time before the new confinement installation.

Figure 2.31. **Dynamics of daily (a) and annual (b) releases of β-active aerosols with air leakages from 1998 to 2015**



Source: Krasnov et al. (2016).

Tritium specific activity in the groundwater at the site during 1997-2000 increased from 2 to 5 000 Bq/l. Spatial distribution of tritium in the groundwater and other features, such as similar distributions of pH index, concentrations of sulphates, chlorides, phosphates and nitrates indicate leakage of contaminated water from the unit 4 into the groundwater (Shestopalov et al., 2000) and soluble radionuclide releases with groundwater into the geological environment. Very limited quantitative data about tritium source term can be found in literature.

The new confinement moved on place in November 2016 has the objective to control radioactive releases of aerosols and contaminated water, even if the original sarcophagus collapses and re-criticality events occur.

- ### Long-term management of damaged fuel and wastes

Plans for fuel and lavas recovery and removal from the shelter have not been finalised yet. According to the most optimistic prognosis these activities will start after 2030. It is very important to remove debris and waste during the lifetime of the new confinement.

The amounts of fuel and lavas to be removed are shown in Table 2.2 but the masses of highly contaminated solid radioactive wastes, including the wastes of units 1-3 are much higher.

It is not clear, at the moment, if the removed corium and fuel will be reprocessed and how the final disposal will be realised.

Long-term water and solid waste management

Effective waste management in Ukraine would need the development of a national programme for radioactive waste disposal, covering: (a) facilities for reprocessing and temporary storage, and (b) building a final repository for the high-level wastes. Chernobyl units 1 to 4 are not the only sources of the wastes. Presently in Ukraine there are:

- 2 917 960 m^3 of wastes accumulated at the temporary storages;
- 3 720 m^3 of solid wastes and approximately 800 m^3 of liquid wastes at 800 places inside the 30 km zone around the plant.

The presently available technology of liquid waste reprocessing is based on water evaporation. This technology cannot be used for contaminated water from the shelter and the 3rd unit because of very high concentrations of Pu and other transuranium elements but also organic substances, such as oil and its products, surfactants and film forming compounds used for suppression of dust, aerosols and fission product release from corium and fuel debris. There is clearly a large space for development of new decontamination and reprocessing technologies and new practices for the cleaning of the NPP and the 30 km zone and finally its rehabilitation.

Comprehensive recommendations on management of radioactive wastes produced by the Chernobyl accident but also state-of-the-art review of dedicated techniques and experiences are formulated by NEA Radioactive Waste Management Committee (RWMC) and its Expert Group on Fukushima Waste Management and Decommissioning (EGFWMD) established in 2014. The work covers physical and chemical nature of the wastes, radiological characterisation, waste classification/categorisation, conditioning, reprocessing and disposal.

According to the recently published NEA report (NEA, 2016):

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

The report concludes on waste storage/disposal that: Waste generated on the site of Chernobyl NPP resulted from the operational activity of the ChNPP units and waste resulting from clean-up after the accident, and mainly from decontamination activity. These wastes are stored in existing (“old”) temporary storage facilities for solid and liquid waste. A set of new facilities were constructed on the ChNPP site to start retrieval, characterisation, sorting, treatment and conditioning of stored waste. As a final product, it is expected to produce drums and containers with solidified (cemented) waste to be acceptable for disposal in the near-surface disposal facility.

The first experience of licensing such a facility – the Engineered Near-surface Disposal Facility (ENSDF), – has highlighted the lack of data on radiological characteristics of conditioned waste, because of a lack of information about waste to be retrieved from the "old" storage facilities. This has meant that cautious assumptions are made about the waste to be disposed of at the facility, which in turn means that the facility may not be used to its full environmental capacity. It is recognised that operators of treatment plants should now provide better radiological characterisation of the waste forms being produced.

Moreover, the requirements of the Ukrainian legislation for radioactive waste disposal have been found to be very conservative for some parameters compared with IAEA Safety Standards. One lesson learnt is that Ukrainian legislation should be updated to take account of IAEA Safety Standards, at the same time as radioactive waste classification is updated and general acceptance criteria for future disposal facilities are developed.

Finally, the benefits from international co-operation for different decommissioning activities are discussed in the report.

Summary of long-term management of the Chernobyl accident

Leaving aside the causes that lead to the Chernobyl-4 accident and the possible lessons learnt from the early phase of the accident for reactor safety and Bolchoi Mochnosti Kanalnyi (high power channel-type reactor) reactor design improvements, the following may be discussed in the context of the long-term management of this accident.

It should be noted initially that the time to reach the stabilised controlled state of the unit 4, as defined in Section 3.1, was very long. The reactor was in an "extreme-damage" state, since the reactor core and building were almost completely destroyed. Although the source term and the radiation levels were reduced significantly after about ten days when the core graphite fire was extinguished, there was still considerable release of radioactive aerosols and very high dose rate from local contaminations at the site. It took six months to construct the sarcophagus (or shelter) which isolated radioactive materials and equipment from the environment. This duration of six months for the shelter construction project is in fact extremely short, considering the radiation levels, destroyed systems and buildings, the size of the shelter and the functional requirements for the shelter. Shelter construction was extremely costly in terms of doses for "liquidators" – the staff that was sent to do the accident clean-up.

Several complicated, expensive and urgent technical measures, such as dose rate mapping, primary decontamination of turbine hall roof and of the plant site in the vicinity of the other units, isolation of destroyed unit 4 from unit 3, allowed restart of the first two units and restart of unit 3 later – in November of 1987. Unit 2 was shut down in 1991, unit 1 in 1996 and unit 3 in 2000. The decommissioning of units 1-3 is being carried out separately from that of unit 4.

Much more time was necessary to diagnose locations of lavas containing the melted fuel and establish control of temperatures and criticality of the debris as well as to suppress formation of dusts at its surfaces. Status of spent fuel was checked only in 1990, i.e. almost four years after the accident. However, the possible damage to the spent fuel and prognosis of its behaviour in dry conditions are still not evident.

The on-site decontamination and protection of rivers and lakes also took several years. However, some uncontrollable leakages of contaminated water from the shelter and uncontrollable releases into the hydrosphere can have happened. Further, solid wastes produced during decontamination and equipment contaminated during these operations were placed in temporary surface storages of trench type.

The liquidators worked in extreme conditions and many decisions with respect to the necessary prompt measures to stabilise the damaged reactor unit were made with no chance of having the necessary measurements, diagnostics and modelling support in advance. Even now, after 30 years, there are continuous discussions about the effectiveness of different measures undertaken directly after the accident, such us building a concrete slab below the reactor, dropping different materials from helicopters to stop the fire, to form a filtration pad on the top of the destroyed reactor, pouring concrete from the top, building protective wall along the river and several other measures, which were taken at that time.

A relatively stable state of the debris in the shelter was reached in the early 1990s but with the very high price of the high doses to the liquidators. Following that, various long-term management actions contributed to maintain this stable state for approximately 20 years in parallel to the operation of other units on the site and following their shutdown. The main efforts undertaken for the long-term management at the site were as follows:

- Monitoring of radiation inside the shelter and at the site including measurements of radioactive releases from the shelter in the form of radioactive aerosols and contaminated water.
- Detection of uncontrolled air and water leakages and sealing of the shelter building.
- Reduction of radioactive dust formation by sprays of solutions forming a protective polymer film on the surfaces of lavas and internal shelter structures/equipment.
- Keeping optimal humidity inside the shelter to minimise both dust formation and radioactive water accumulation.
- Diagnostics and monitoring of corium containing lavas at their locations. Control of debris temperature and reduction of criticality risk using spray system operated with water but also with boron and gadolinium solutions.
- Further on-site decontamination and building of radiation protective structures for operating team as well as radioactive waste storage facilities.
- Extensive research programmes necessary to understand accident progression, in particular temperatures reached in the course of the accident, ex-vessel corium behaviour and MCCI, to complete database representing corium locations, compositions and properties in order to support long-term management and engineering at the site but also to develop prognosis about long-term evolution of Chernobyl lavas, environmental releases, accumulation of radioactive wastes and corresponding risks.
- Monitoring of mechanical and structural stability of various structures and components. Three main parts were detected in 1989 as having critical deformations and mechanical ruptures: i) in the upper zone of deaerator support stack, ii) Ceiling of south room of main recirculation pumps (#402/3) and iii) Ceiling of south room of ventilation pipes (#805/3). These structures were renovated and strengthened in priority. Many other structures were serviced, renovated with new reinforcement or replacement later on, e.g. main ventilation pipe (1990), south and west walls of shelter, its roof (2000-2008), and other structures later on. In spite of many completed renovation and reinforcement measures, it has been concluded that the sarcophagus is mechanically unstable and not leak tight. Decision was made to build a new confinement around the old one.
- Design and construction of the new shelter with a lifetime of 100 years. It has been completed and installed in late 2016.

The listed efforts are not sufficient for completion of long-term management of Chernobyl accident but they provide the starting conditions for the future phase of spent fuel and corium debris removal and reprocessing. The detailed plan, which is expected to be completed within the life time of the new confinement, and technical measures for debris removal are currently in preparation.

The following studies of the Institute for Safety Problems of NPP (ISPNPP, Ukraine) can be listed among the currently ongoing research which can further contribute to the long-term management of Chernobyl NPP:

- complex assessment of cumulative environmental impacts of radiation-hazardous objects of Chernobyl evacuation zone:
  - development of measures to minimise the risks of re-criticality incidents;
  - study of radioactive aerosol behaviour during commissioning and operation of the new confinement;

  - research and development to minimise the risks associated with contaminated waters in the new confinement;
  - statistical analysis of radiation monitoring results and ecological studies of the new confinement and the plant site;
  - development of system design for monitoring of new confinement internal space and data acquisition system.
- study of Chernobyl lava in the conditions of new confinement and development of methods and technologies of lava reprocessing;
- uncontrolled releases from shelter and aerosol behaviour during new confinement construction;
- assessment of evolution in the evaporation-condensation dynamics of moisture in the shelter covered by the new confinement, behaviour of Chernobyl lavas and dust formation;
- radio-hydrogeological monitoring in the site;
- development of characterisation methods for solid radioactive wastes and packages of waste reprocessing plant;
- assessment of protective polymer coating behaviour in the under-roof space of the shelter.

Experimental study of fission product leaching from Chernobyl lava samples by water will be carried out at Khlopin Radium Institute (KRI), Russia in 2018-2019. This study is supported in the international call of the NEA TCOFF project on Thermodynamic Characterisation of Fuel Debris and Fission Products based on Scenario Analysis of Severe Accident Progression at Fukushima Daiichi.

Synthesis of relevant Fukushima Daiichi experience

Generalities on long-term management and recovery actions at Fukushima Daiichi

The safe long-term management (LTM) at the Fukushima Daiichi NPP is a highly challenging task that requires the allocation of enormous resources as well as the development and use of innovative technologies to deal with very difficult activities.

After the accident, reaching a stabilised controlled state (sub-criticality and cooling of the damaged fuel and limitation of radioactive releases) for the damaged 1, 2 and 3 units and unit 4 SFP was the priority. This was particularly challenging at Fukushima Daiichi because of the damages to the infrastructures due to the seisms, the tsunami, the hydrogen explosions and the impact of fires and because of the large on-site contamination that resulted from the loss of confinement in units 1 through 3. Contamination spread was also aggravated by hydrogen explosions and fires. First actions after the accident concerned the re-establishment of ground transports, clean-up of the site to enable access and implementation of systems, equipment and instrumentation for securing and monitoring a stabilised controlled state and to reduce radiological dose rate.

On 17 April 2011, Tokyo Electric Power Company (TEPCO) established the “Roadmap towards Settlement of the Accident at Fukushima Daiichi Nuclear Power Station, TEPCO”. This Roadmap set two steps for stabilisation of the plants as targets: “Radiation dose is in steady decline” as “Step 1” and “Release of radioactive materials is under control and radiation dose is being significantly held down” as “Step 2”.

Step 1 was completed on 15 July 2011, and step 2 was completed on 16 December 2011. By completion of step 2, the following were achieved and the plants were declared as in a "cold shutdown state" by the government (NDF, 2016):

- the reactor pressure vessel (RPV) bottom temperatures and temperatures inside the primary containment vessel (PCV) are kept below approximately 100 degrees centigrade;
- the steam generation from the PCVs is suppressed via controlling water injection which also suppresses the release of radioactive materials with steam from the PCVs;
- the mid-term safety of the circulating water cooling system is secured.[1]

What also made and is still making the situation challenging at Fukushima Daiichi is that three units have been damaged with different accident short-term evolutions and accident management actions resulting in different unit damaged states.

It should be emphasised that LTM management actions, while aiming at maintaining and monitoring a stabilised controlled state of the degraded fuel in the damaged units, are now largely addressing radioactive waste management and the preparation of damaged fuel retrieval issues. The radioactive liquid wastes management is a critical issue at Fukushima Daiichi notably due to the groundwater intake in the damaged facilities and the large volumes of contaminated water produced. Details of radioactive waste are summarised in *Management of Radioactive Waste after a Nuclear Power Plant Accident* (NEA, 2016). Damaged fuel retrieval is a critical step towards safe decommissioning.

The situation on-site has been largely improved since the accident. Indeed, several important tasks have been accomplished:

- re-establishment of ground transport which was a priority issue;
- clean-up of the site resulting in a reduction in radiological dose rate, eased access to critical parts of the plant for safe LTM and for the preparation of further recovery actions;
- implementation of closed-loop damaged core cooling in units 1 through 3;
- completion of the removal of the fuel from unit-4 (done late 2014);
- expanded and improved contaminated water treatment system to reduce chlorine (added with seawater injection during emergency phases) and radioactivity contents (reduction in main contributors' contents, i.e. ^{137}Cs and ^{90}Sr) in water;
- implementation of measures to reduce groundwater intake and contaminated water wastes;
- installation of new tanks and associated systems for contaminated water storage.

The detailed timeline of these operations is shown in Table 2.7.

It is to be expected that the situation at the site will remain very complex. A range of challenging issues remain, such as the persistent underground water ingress to main buildings and the accumulation of contaminated water on-site, the long-term management of radioactive waste as well as those related to the removal of nuclear fuel, damaged fuel and fuel debris.

1. "Mid-term" refers to the period of time between completion of step 2 and the decommissioning of the reactor. Mid-term safety is ensured by provisions that were taken by TEPCO and reviewed by the Nuclear and Industrial Safety Agency (NISA). Reliability of the components of the circulating water cooling system is ensured by diversification and other measures. Means to detect anomalies and alternative measures in the case of inoperability of the system have been provisioned. Also, provisions were set to mitigate any significant radiological risk.

Table 2.7. **Timeline of events and operations at the Fukushima Daiichi NPP following the 11 March accident up to July 2017**

2011 All reactors in cold shutdown	Mar 11	The 2011 Great East Japan Earthquake struck
	Mar 11	Second wave of Tsunami arrived, main buildings of units 1 through 6 flooded
	Mar 11	Station blackouts at units 1 through 5
	Mar 12	A hydrogen explosion took place in unit 1 reactor building (R/B)
	Mar 14	A hydrogen explosion took place in unit 3 R/B
	Mar 14	A blowout panel was forced open to release pressure inside unit 2 R/B
	Mar 15	A hydrogen explosion took place in unit 4 R/B
	Mar 24	Three workers exposed to high radiation in unit 3 turbine building (T/B) basement
	Apr 2	Highly contaminated water leaked into the sea near unit 2 water intake
	Apr 4	About 10 000 m^3 of low-level contaminated water discharged into the sea
	Apr 11-14	Completion of silt fence installation near units 1-4 water intakes
	May 11	Highly contaminated water near unit 3 water intake leaked into the sea
	May 31	Full-scale operations of alternative cooling equipment in unit 2 SFP started
	Jun 17	Operations of water treatment facilities to mainly remove caesium started
	Jun 27	Full-scale operations of circulating cooling water in units 1 through 3 started
	Jun 30	Full-scale operations of alternative cooling equipment in unit 3 SFP started
	Jul 31	Full-scale operations of alternative cooling equipment in unit 4 SFP started
	Aug 10	Full-scale operations of alternative cooling equipment in unit 1 SFP started
	Aug 18	Operations of water treatment facility Simplified Active Water Retrieval and Recovery System (SARRY) to mainly remove caesium started
	Oct 28	Completion of building cover installation for unit 1 R/B
	Dec 16	All of the reactors in cold shutdown (below 100°C) declared by the government
2012 Start making full-scale measures against contaminated water	Feb 3	About 10 L of highly contaminated water leaked from a storage tank
	Apr 5	About 12 m^3 of contaminated water leaked from water treatment facility
	Apr 19	Permanent shutdown of units 1 through 4 declared
	Apr 25	Installation work started for seaside impermeable wall
	May 11	Completion of seabed paving work at units 1 through 4 water intakes
	Sep 22	A steel frame fell into unit 3 SFP during rubble removal work
	Sep 23	Completion of Multi-Nuclide Removal Facility (ALPS) construction
	Oct 2	Start construction of groundwater bypass pumping system
2013 Struggled in efforts to deal with issues	Mar 11	Unit 2 blowout panel closed to prevent further spread of radioactive materials
	Mar 18	Power outage occurred and caused temporary suspension of SFP cooling system
	Mar 30	Started test operations of Multi-Nuclide Removal Facility (ALPS)
	Apr 5	Confirmed contaminated water leakage from underground reservoir
	Apr 12	Start of dry cask operations to store spent fuel in the shared pool
	Jun 19	A high level of radioactivity detected in monitoring wells at bank protection
	Jun 30	Start of Entrance Control Building operations
	Jul 20	Completion of building cover construction at unit 4 for spent fuel removal
	Aug 12	Ten workers exposed to radiation from dust scattered during unit 3 rubble removal
	Aug 19	About 300 tonnes of contaminated water leaked from bolted-type storage tanks
2014 Completion of fuel removal from Unit 4 spent fuel pool	Jan 31	Permanent shutdown of units 5 through 6 declared
	Mar 8	Completion of rubble removal inside the unit 4 SFP
	Mar 28	A fatal accident at solid waste storage facility: a worker died in a mudslide
	Apr 1	Fukushima Daiichi Decontamination and Decommissioning Engineering Company set up
	Apr 9	Start of groundwater bypass pumping operations
	May 15	Accumulated water leaks found around expansion joints of unit 3 main steam pipes
	May 21	Started discharging groundwater pumped up by Groundwater Bypass system
	May 27	Accumulated water leakage found in unit 1 Pressure Suppression Chamber
	Jul 29	Completion of drainage B and C channels reconfiguration to discharge to the port
	Sep 17	Test operations of Advanced Multi-Nuclide Removal Facility (ALPS) started
	Oct 18	Test operations of High-Performance Multi-Nuclide Removal Facility (ALPS) started
	Dec 22	Completion of fuel removal from unit 4 SFP
	Dec 26	Water treatment facilities SARRY and KURION (named after the US waste treatment company) improved to remove Strontium

Table 2.7. **Timeline of events and operations at the Fukushima Daiichi NPP following the 11 March accident up to July 2017** (cont'd)

2015 Completion of contaminated water treatment	Jan 10	Reverse-osmosis concentrated water treatment operations to mainly remove Strontium started
	Jan 19	A fatal accident: a worker fell from the top plate of a water storage tank
	Feb 12	Investigation inside unit 1 PCV carried out using muon technology
	Feb 24	Tainted rainwater leakage found from drainage into the sea
	Apr 10-20	Investigation inside unit 1 PCV carried out using a robot
	Apr 17	Start transferring drainage K water into the port with temporary pumps
	Apr 23	Completion of seabed paving work for the entire port to seal in floating mud
	May 27	Highly contaminated water treatment complete, stored in tanks after the accident
	May 31	Large Rest House operations started
	Jun 30	Highly contaminated water removal inside unit 2 seawater piping trench complete
	Jul 30	Highly contaminated water removal inside unit 3 seawater piping trench completed
	Aug 2	Completion of fuel handling machine removal in unit 3 SFP
	Aug 8	A fatal accident: a worker died in the middle of cleaning a construction vehicle
	Sep 3	Start of groundwater pumping by Sub-drain System installed around buildings
	Sep 14	Start of treated groundwater discharge to the port, pumped up by Sub-drain System
	Oct 5	Completion of roof panel removal of unit 1 building cover
	Oct 20, 22	Investigation inside unit 3 primary containment vessel (PCV) carried out using cameras
	Oct 26	Completion of Seaside Impermeable Wall closure
	Nov 26	Investigation around equipment hatch of unit 3 PCV carried out using small robot
	Dec 8	Expansion of the areas where workers can move around in normal work uniforms
	Dec 11	Highly contaminated water removal inside unit 4 seawater piping trench completed
2016 Start of full-scale investigation for fuel debris removal	Feb 9	Completion of landside impermeable wall installation
	Mar 8	Adjustment of radiation protection areas and appropriate worker's outfits
	Mar18	Operation of miscellaneous solid waste incineration facility started
	Mar 22	Investigation inside unit 2 reactor carried out using muon technology
	Mar 28	Reconfiguration of drainage K outlet to the port completed
	Mar 31	Freezing of landside impermeable wall started
	April 12	Shielding materials started to be installed on the top floor of unit 3 R/B
	Jun 20	New drainage channels constructed on the premises
	Sep 21	Groundwater levels increased in the area 4 m above sea level due to extended rain
	Sep 30	Construction of new main administration building completed
	Nov 10	Completion of water panel removal (18 panels) from unit 1 R/B cover
	Dec 5	Unit 3 core injection stopped temporarily but resumed with an alternative pump
2017 Fuel debris removal methods for each units to be determined	Jan 17	New roof installation began on unit 3 R/B for spent fuel removal
	Feb 16	Unit 2 PCV investigation conducted with a robot
	Mar 18	Unit 1 PCV investigation conducted with a robot
	Jul 19	Unit 3 PCV investigation conducted with a robot
	Jul 31	Installation of unit 3 spent fuel removal cover dome started
	Aug 22	Closure of last part of landside impermeable wall started

Present knowledge of PCV and RPV status

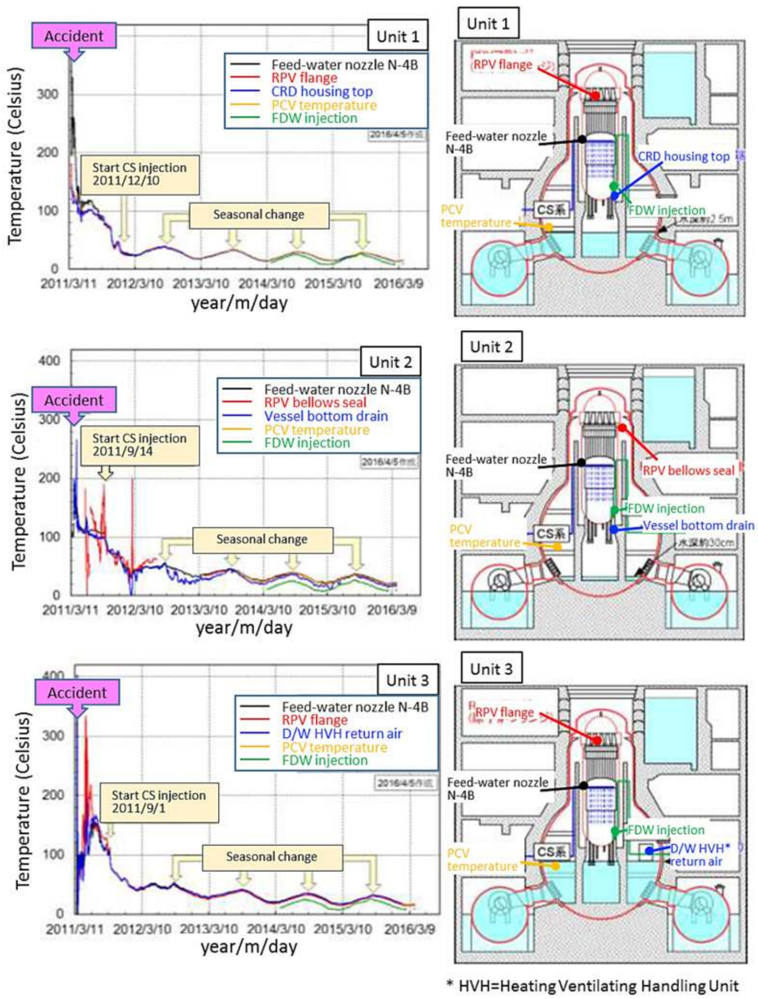
The knowledge of PCV leak paths and of the location and status of the fuel debris is of prime importance to progressing safely in the fuel recovery preparation and plan the corresponding actions. The investigations on the conditions of the PCVs and RPVs are progressing based on the observations reported by TEPCO (from visual inspections, dose rate measurements, identification of leakage paths etc. using robotics means) and with the support of accident analyses (NDA, 2011). However, to this date, the access to both PCV and RPV is still challenging due to high dose rates and the existing rubble. Further examinations and recovery actions are engaged and planned to gain further access and insights to PCV and RPV status. Of particular interest for recovery actions are the PCV leak paths and the location and status of the damaged fuel (usually called "debris" in discussed fuel retrieval plans).

Temperature and dose rate in PCV have shown stable decrease and settling during the past five years since the accidents (Figures 2.32 and 2.33). Based on inspections conducted so far, possible water leak paths of PCV have been located for each unit (Figure 2.34). For the suppression chamber in unit 2, further efforts for identifying leak paths under the water surface level have been made as described later.

Progressions towards decontaminating reactor buildings and removal of debris are briefly described as follows:

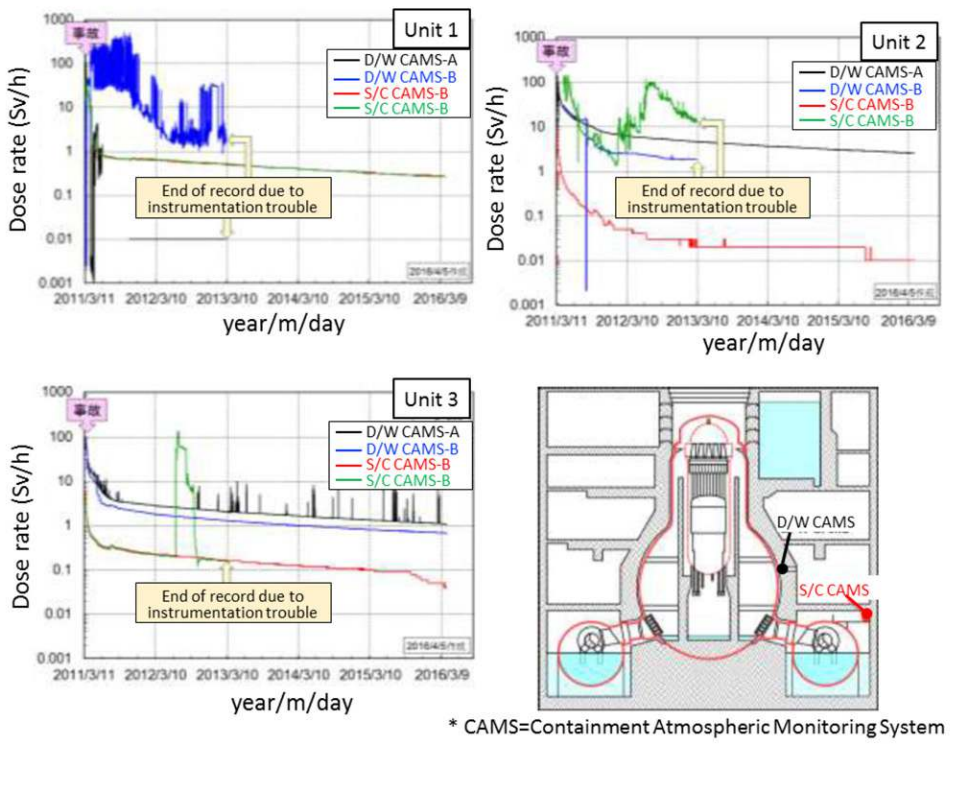
- For unit 1, structures covering the damaged reactor building were removed and after the removal of the cover, removal of rubble on the refuelling floor will be started. Robots have been used to identify leakage paths from the PCV.
- For unit 2, the reactor building was not damaged and robots were used for examinations in the refuelling floor zone (retrieving concrete samples) and in the PCV.
- For unit 3, rubble removal above the refuelling floor from the damaged reactor building using remote controlled heavy machinery was completed. Rubble was also removed from the SFP to prepare for fuel removal. Spent fuel removal equipment is being installed after shielding installation.

Figure 2.32. **Temperature trends during five years after accidents in Fukushima Daiichi units 1, 2 and 3**



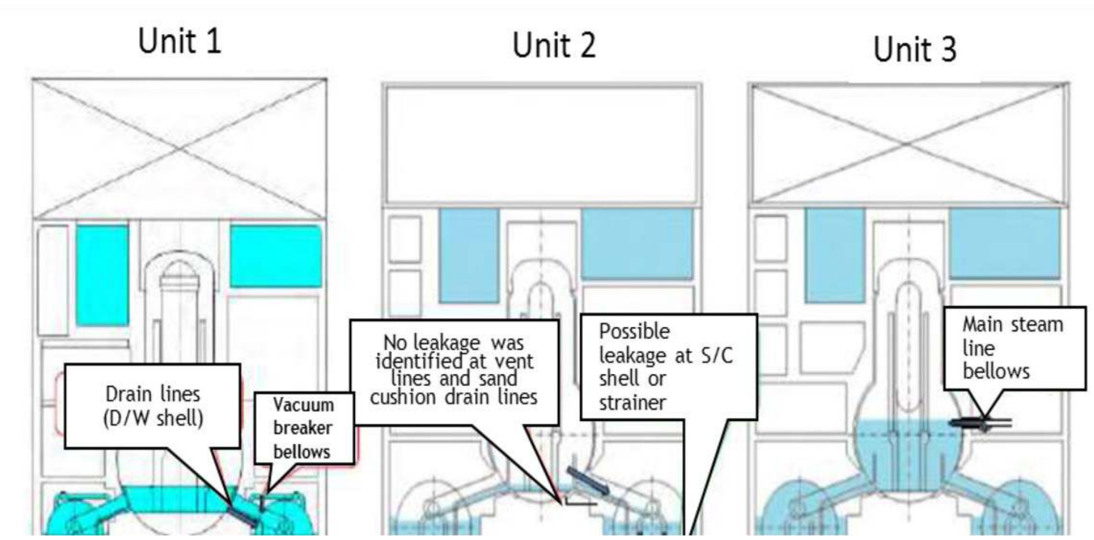
Source: TEPCO (2018).

Figure 2.33. **Dose rate trends during five years after accidents in Fukushima Daiichi units 1, 2 and 3**



Source: TEPCO (2018).

Figure 2.34. **Possible leak paths in Fukushima Daiichi units 1, 2 and 3**



Source: TEPCO (2018).

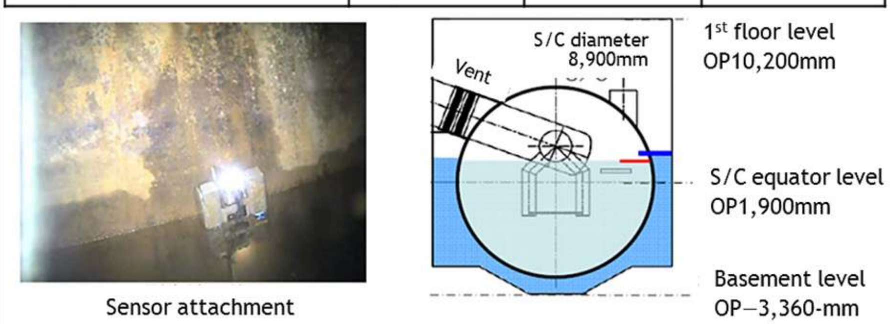
Containment overview and identified major leakage paths

Unit 1: A large amount of water and steam during the accident has likely been discharged into the torus room through possible failure portions of the containment and suppression chamber (S/C), where leaking water flow indications were found by visual inspections on the PCV (such as vacuum breaker lines on the S/C, and sand cushion drain pipes between the dry well and the torus room). There seems to be no leakage from S/C lower parts below water level in the torus room, because the water level inside S/C is maintained high. S/C itself is assumed leak tight, except the leaking portion at the expansion joint cover of the pipes on the upper part of the S/C. On the other hand, leakage holes in the PCV dry well are considered small, as water level inside of the PCV can be maintained at 2.5 m above the dry well bottom floor.

Unit 2: Water is leaking from the lower part of the S/C. The PCV dry well water level stays relatively low since injected water flows into the S/C and is assuredly leaking from the S/C. A remote controlled device with an ultrasonic water level sensor was sent from the torus room to the S/C and was moved along its outer peripheral surface. It was identified that there is a difference in water level between inside and outside of S/C. The water level of S/C depends on PCV pressure increase caused by nitrogen injection. It is interpreted that a moderate change in this water level (about 10 mm to 30 mm) is caused by submerged leak path(s) in S/C (Figure 2.35).

Figure 2.35. **S/C water level measurement in Fukushima Daiichi unit 2**

Measurement date	2014/1/14	2014/1/15	2014/1/16
S/C water level (mm)	~OP 3,120	~OP 3,160	~OP 3,150
Torus room water level (mm)	~OP 3,230	OP 3,190	~OP 3,160
Difference (mm)	~20	~30	~10



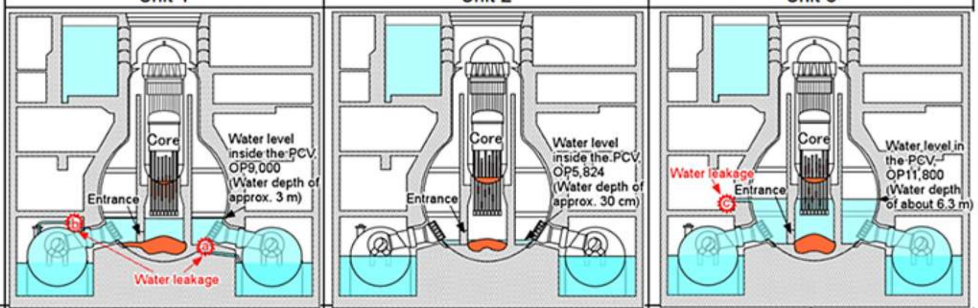
Source: TEPCO (2018).

Unit 3: Water is leaking through the PCV dry well from the expansion joint of the main steam line (MSL). No leakage is occurring from the PCV dry well below the MSL and no abnormality is reported about S/C. The water level can be maintained at 6.3 m above the dry well floor (up to the level of the MSL penetration).

RPV status and estimated degraded fuel location

Analyses of cooling water temperature evolution at different RPV locations (heat balance method) provides indications that a large part of fuel debris was released from unit 1 RPV whereas some degraded fuel is still present in the unit 2 and 3 RPVs. Taking into account these observation results and Modular Accident Analysis Programme (MAAP) calculation results reported in the NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (NDA, 2011), the present status of debris in RPV and PCV are estimated for units 1, 2 and 3 as depicted in Figure 2.36.

Figure 2.36. **Current status in the three damaged units of Fukushima Daiichi units 1, 2 and 3**



[Unit: ton]		Unit 1		Unit 2		Unit 3	
	Location	Range of Estimation*1	Typical Value*2	Range of Estimation*1	Typical Value*2	Range of Estimation*1	Typical Value*2
Estimated Distribution	Core	0-3	0	0-51	0	0-31	0
	RPV Lower Head	7-20	15	25-85	42	21-79	21
	Inside RPV Pedestal	120-209	157	102-223	145	92-227	213
	Outside RPV Pedestal	70-153	107	3-142	49	0-146	130
	Total	232-357	279	189-390	237	188-394	364
Current Status	Dose Rate measured in PCV	Approx. 5-10 Sv/h (measured on 10-16, April, 2015, in gas phase of 0.7 m from the water, on the grating)		Approx. 31-73 Sv/h (measured on 27, March, 2012, in gas phase of 3.7 to 6.7 m from the water, around X-53 penetration)		Approx. 0.75-1 Sv/h (measured on 20, Oct., 2015, in gas phase of 0.55 m from the water, around X-53 penetration)	
	Observed water leakage points	Water flows were detected from one sand cushion drain pipe (ⓐ) and one expansion joint cover (ⓑ) of the S/C.		It is expected water flow from somewhere below water level inside the torus room, since there is no trace of water leakage in gas phase.		Water flow was detected from the expansion joint (ⓒ) of the MS Line D in the MSIV room.	
	PCV internal investigation etc.	• No large scale damage on the existing equipment (PLR pump, wall inside the PCV, HVH etc.) • Deposits are located in a wide range of the bottom of the D/W • PLR pipe shielding unit were dropped.		• Structures on the lower part of the RPV were confirmed by the photos taken inside. Damage on the RPV may not be a large-scale		• No damage was confirmed on the PCV internal structures and walls through the PCV internal investigation using the investigation device inserted from the penetration of PCV.	

Source: TEPCO (2018).

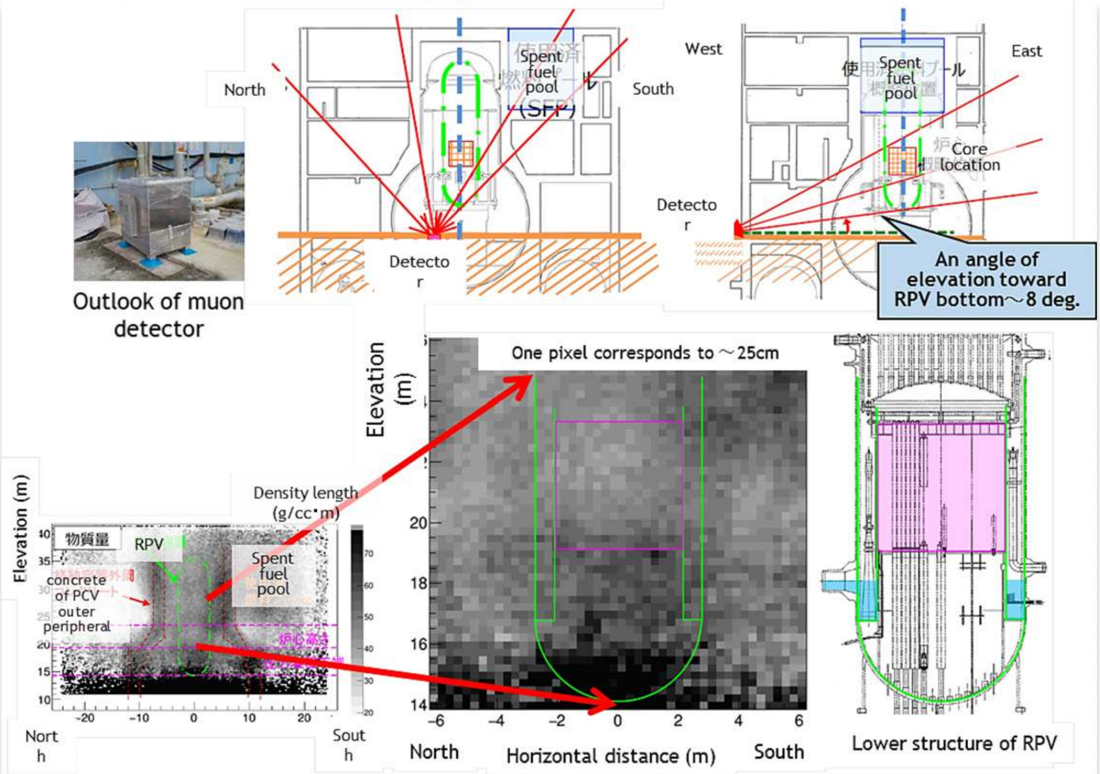
A summary of the investigations performed in damaged units 1, 2 and 3 is provided in Table 2.8.

An advanced radiographic measurement based on the muon tomography technology has been implemented in units 1 and 2. It was observed that major structural parts can be visualised as light and shade pixel maps. In unit 2, it was found that high-density materials reside in the lower head of RPV (Figure 2.37). In comparison with the calculation results shown in Figure 2.36, this observation indicates that a significant part of debris is located in the lower head. It was also suggested that a substantial amount of fuel debris may remain in the lower and peripheral parts of the core although images of these parts present uncertainties caused by internal structures.

Investigations using self-propelling robots were conducted for units 1 and 2 in January to March 2017. Results are summarised hereafter.

Unit 1: The Hitachi-developed Primary Containment Vessel Internal Survey Equipment ("PMORPH") robot made several investigations into the PCV in five days and completed its mission on 22 March. Equipped with a dosimeter and waterproof camera, it took dose measurements (Figure 2.38a) and digital images (Figure 38b) at ten different points. Deposits were found at the bottom of the PCV and on piping. The deposits will be analysed after taking samples. Radiation dose levels decrease upon submerging into the water and rise again when nearing the floor (Figure 2.38c). There has been little change in dose levels above the grating compared to the previous survey in April 2015 and no significant damage was found to the existing structures during this period. It will take more time to conclude whether fuel debris exists at the measured points or how far it spread over the PCV basement.

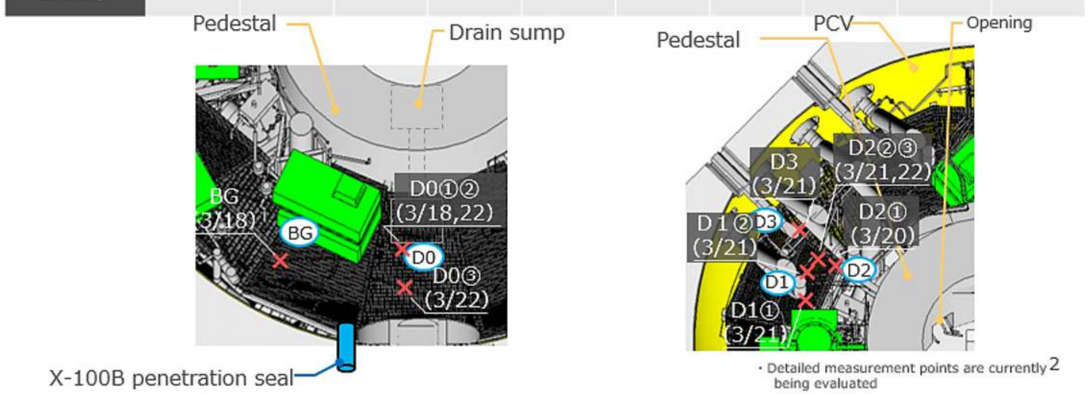
Figure 2.37. **Muon tomography visualisation in Fukushima Daiichi unit 2**



Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2016/images/handouts_160728_01-e.pdf, see also www.irid.or.jp/en.

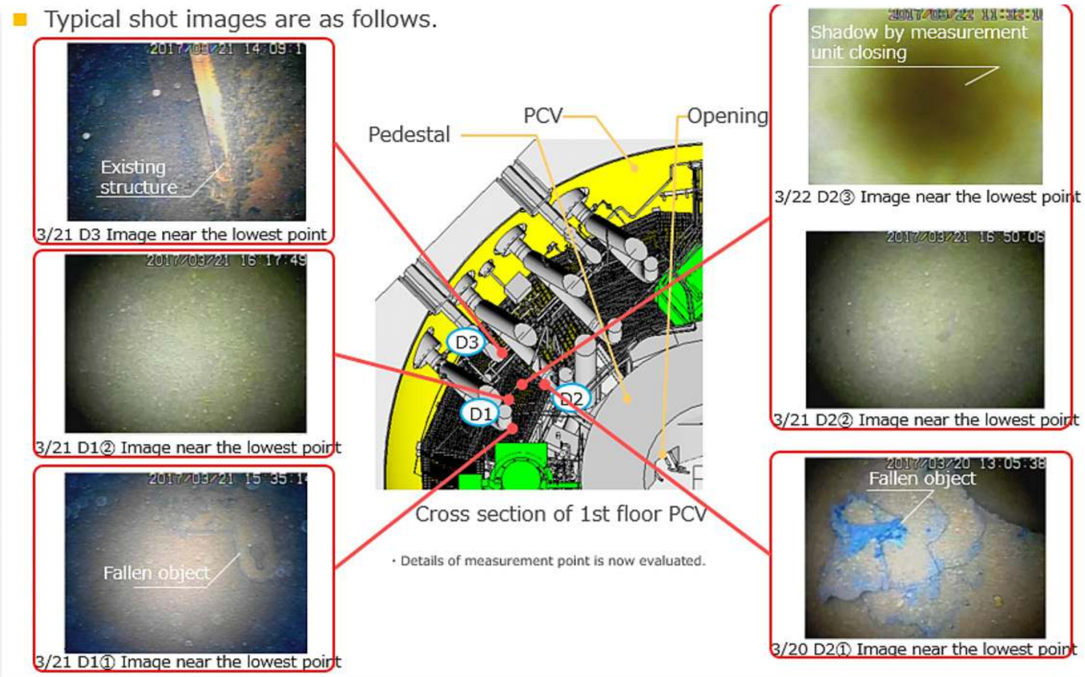
Figure 2.38a. **"PMORPH" robot dose rate measurements in unit 1 PCV**

Measurement Point (Measurement date)	BG (3/19)	D0 ① (3/18)	D0 ② (3/22)	D0 ③ (3/22)	D1 ① (3/21)	D1 ② (3/21)	D2 ① (3/20)	D2 ② (3/21)	D2 ③ (3/22)	D3 (3/21)
Dose levels above the grating [Sv/h]	3.8	7.8	6.7	3.6	8.4	8.2	12	9.2	9.3	10
Dose levels at lowermost point [Sv/h] (Height from the floor)	11 (Approx. 0.3m)	1.5 (Approx. 1m)	1.6 (Approx. 0.6m)	5.4 (Approx. 0.3m)	6.3 (Approx. 0.9m)	5.9 (Approx. 0.9m)	6.3 (Approx. 1m)	7.4 (Approx. 0.9m)	9.4 (Approx. 0.9m)	3.0 (Approx. 1.6m)



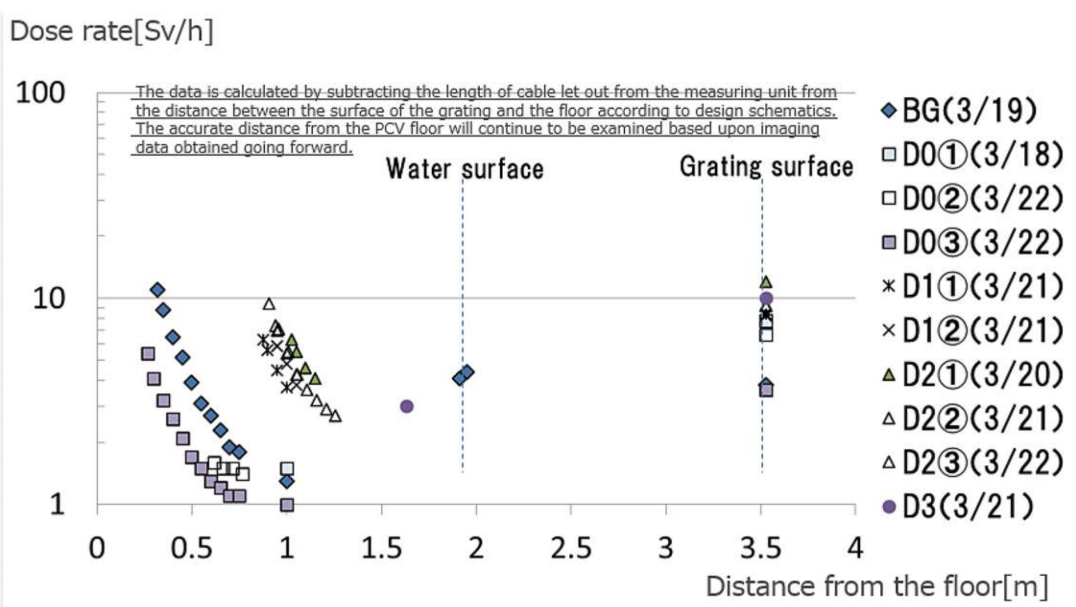
Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170327_01-e.pdf, see also www.irid.or.jp/en.

Figure 2.38b. **"PMORPH" digital images in unit 1 PCV**



Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170327_01-e.pdf, see also www.irid.or.jp/en.

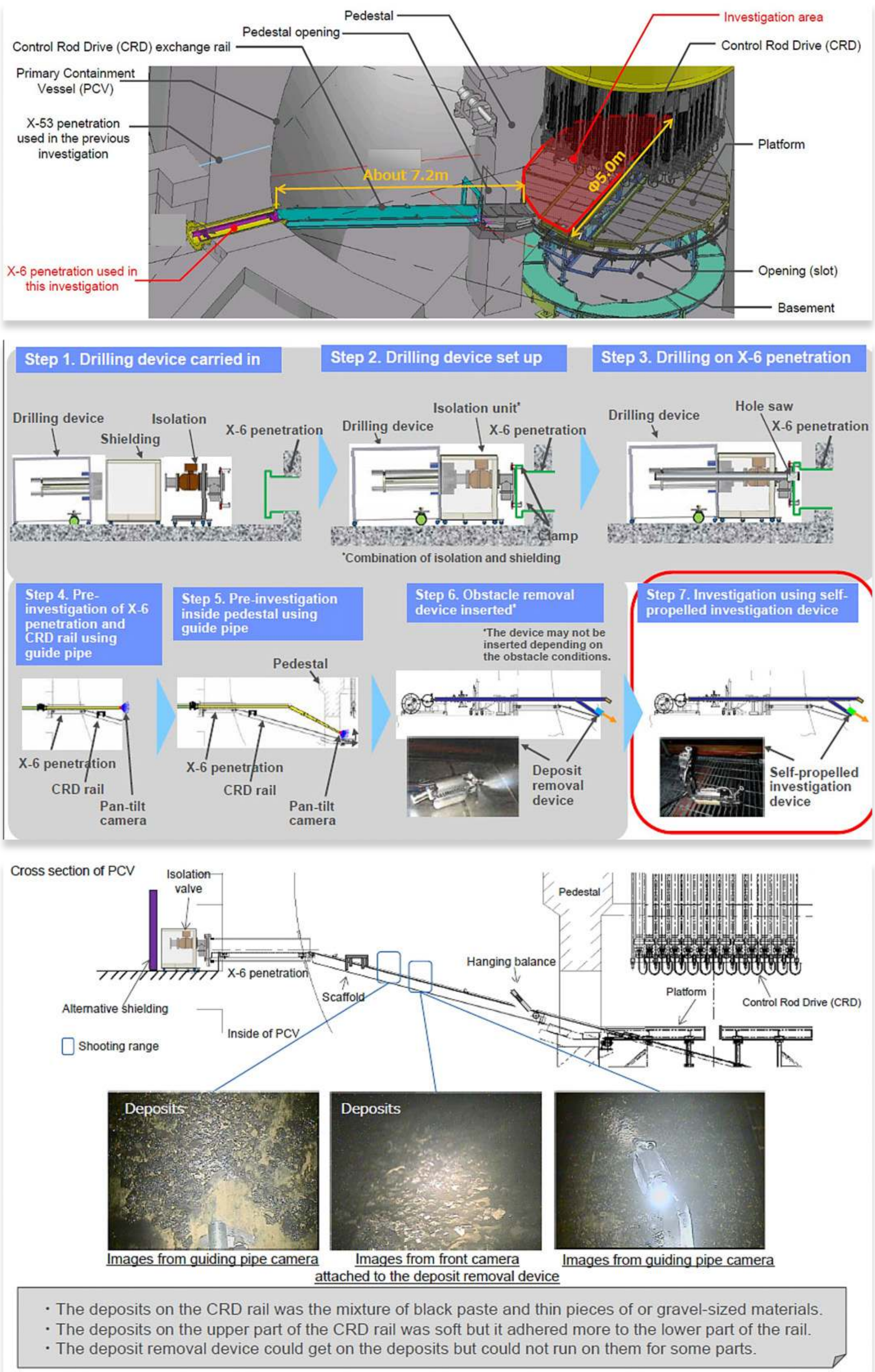
Figure 2.38c. **"PMORPH" radiation dose levels in unit 1 PCV**



Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170327_01-e.pdf, see also www.irid.or.jp/en.

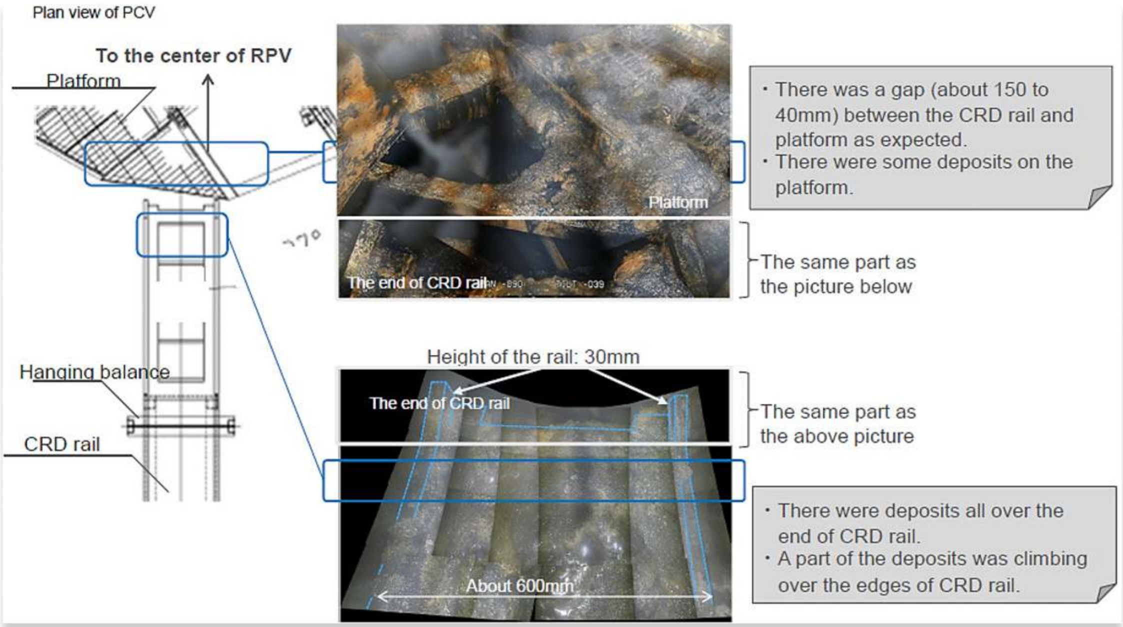
Unit 2: In order to investigate the PCV and clarify the conditions of debris and surrounding structures, Toshiba's self-propelled robot was deployed from January to February 2017 and the results displayed in Figures 2.39a-c were obtained.

Figure 2.39a. **Robot inspection of the unit 2 control rod drive rail area**



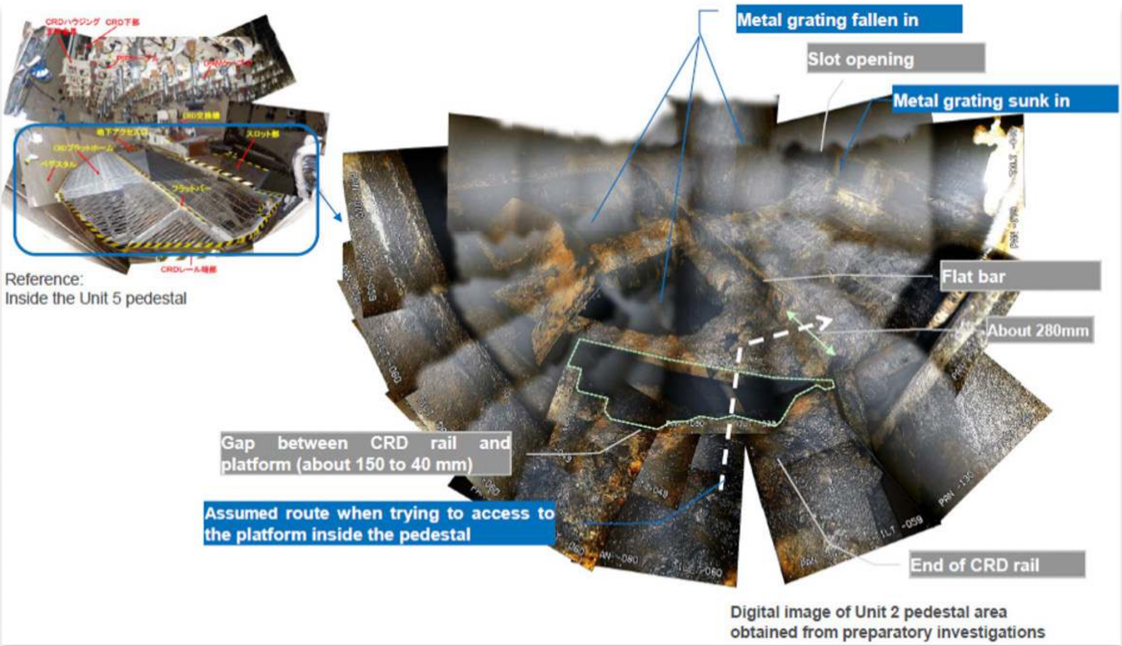
Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170215_01-e.pdf, see also www.irid.or.jp/en.

Figure 2.39b. **Robot inspection of the unit 2 entrance of pedestal area**



Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170215_01-e.pdf, see also www.irid.or.jp/en.

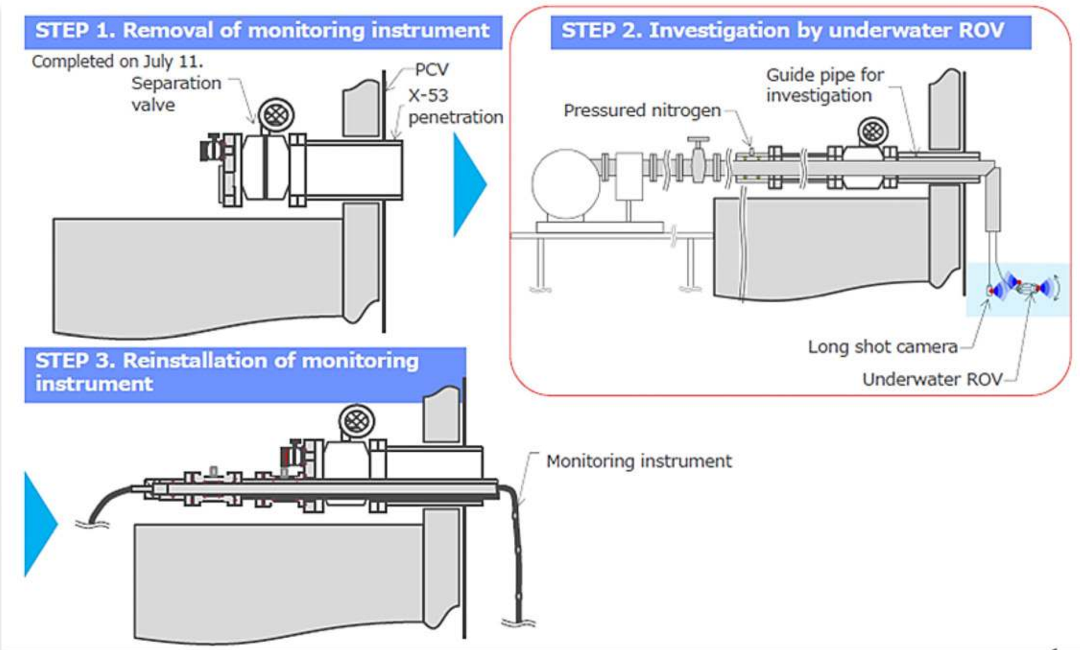
Figure 2.39c. **Robot inspection of the unit 2 pedestal area**



Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170215_01-e.pdf, see also www.irid.or.jp/en.

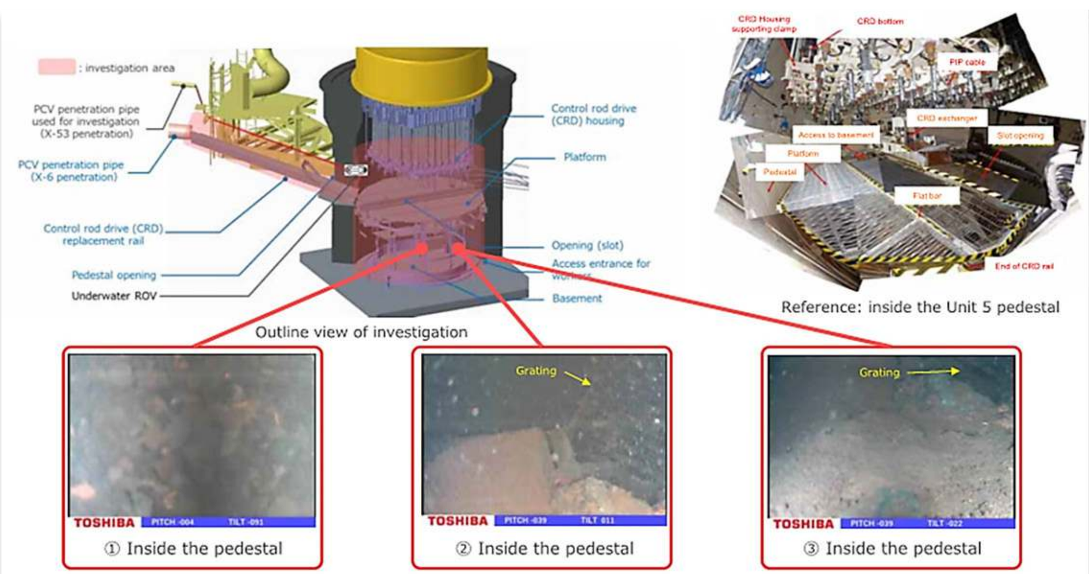
Unit 3: TOSHIBA's submersible remote operated vehicle was launched for an exploration of the inside of the unit 3 PCV in July 2017 and the results displayed in Figures 2.40a-c were obtained. The exploration revealed damage to multiple structures inside the pedestal. Also, likely melted materials that are consolidated and some fallen substances such as grating and sediments were confirmed. The conditions inside the pedestal will be examined precisely based on the image data obtained through the consecutive explorations.

Figure 2.40a. **Unit 3 internal investigation using underwater remote operated vehicle**



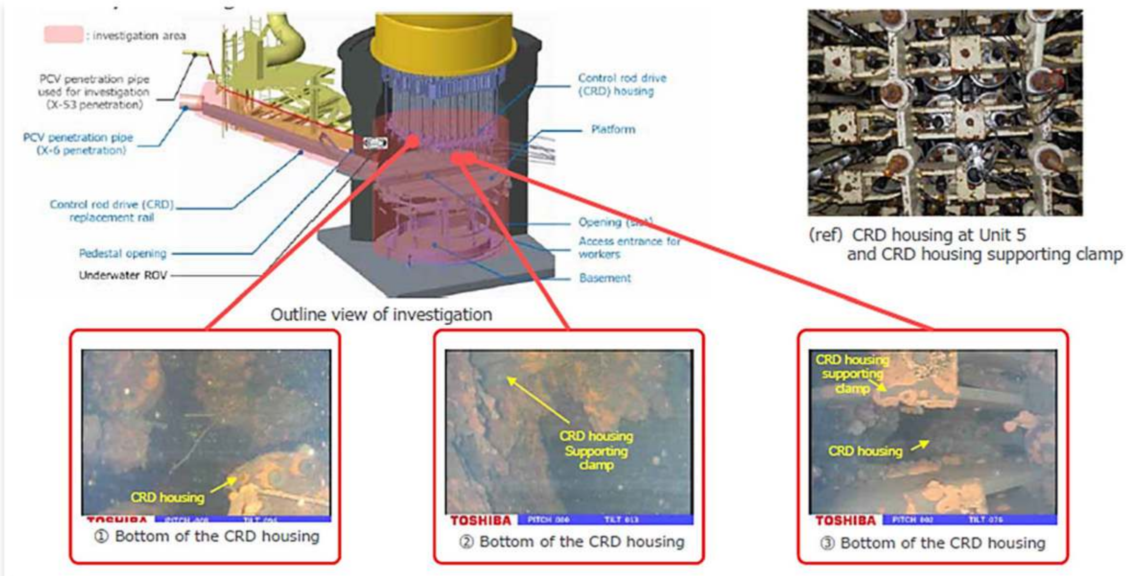
Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170713_02-e.pdf, see also www.irid.or.jp/en.

Figure 2.40b. **Unit 3 PCV internal investigation results**



Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170722_01-e.pdf, see also www.irid.or.jp/en.

Figure 2.40c. **Unit 3 PCV internal investigation results**



Source: TEPCO, www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170721_01-e.pdf, see also www.irid.or.jp/en.

Table 2.8. **Summary of investigations performed in units 1, 2 and 3**

<table>
<tr><th>Unit</th><th colspan="2">Place of investigation</th><th>Investigation results</th></tr>
<tr><td rowspan="8">Unit 1</td><td rowspan="3">PCV internal investigation</td><td>1st internal investigation (October 2012)</td><td rowspan="2">• Confirmation of no major damage to various facilities inside PCV (air conditioning equipment, recirculation pump and piping, pedestal wall, etc.).
• Confirmation of temperature, radiation levels, and water level in PCV: atmospheric temperature: about 18~21°C, radiation levels: about 5~10 Sv/h, water level: approximately 2.8 m from the D/W basement floor.
• Collection of accumulated water.
• Installation of permanent monitoring instrument.</td></tr>
<tr><td>2nd internal investigation: Investigation outside the pedestal (April 2015)</td></tr>
<tr><td>3rd internal investigation: Investigation on the lower outer part of the pedestal (scheduled in March 2017)</td><td>• Measurement of atmospheric temperature, radiation levels, underwater radiation levels, sediment collection, replacement of permanent monitoring instruments.</td></tr>
<tr><td rowspan="4">Investigation inside the building</td><td>Investigation around the lower vent pipe (November 2013)</td><td>• Confirmation of leakage from sand cushion drain line as a leakage point from PCV.</td></tr>
<tr><td>Investigation of the pressure suppression chamber upper part (May 2014)</td><td>• Confirmation of leakage from the expansion joint of the PCV vent pipe vacuum break line as a leakage point from PCV.</td></tr>
<tr><td>TIP room investigation (September to October 2015)</td><td>• Instrumentation penetration had high radiation levels of 100 mSv/h or more and others had low radiation levels.</td></tr>
<tr><td>Investigation of the main steam isolation valve room and the air lock chamber (November to December 2015)</td><td>• In the main steam line isolation valves room, the status of the bellows section could not be confirmed due to the inference of the equipment.
• The air lock chamber has high radiation levels of 7 Sv/h or more at the lower part of the PCV penetration section, and the inside of the bellows cover is estimated to be a source of contamination.</td></tr>
<tr><td>Others</td><td>Muon measurement (February to May 2015)</td><td>• Confirmation of not much fuel in the core area.</td></tr>
</table>

Table 2.8. **Summary of investigations performed in units 1, 2 and 3** (cont'd)

Unit	Place of investigation		Investigation results
Unit 2	PCV internal investigation	1st internal investigation (January 2012) 2nd internal investigation (March 2012) 3rd internal investigation: Investigation inside the pedestal (August 2013) 4th internal investigation: Investigation inside the pedestal (January to February, 2017)	• Confirmation of sediments and fall-off and deformation of gratings inside the pedestal. • Confirmation of no serious damage to the control rod drive housing support near the pedestal entrance. • Confirmation of temperature, radiation levels, water level inside PCV. 1st and 2nd time: Atmospheric temperature: about 43°C~46°C Radiation levels: about 31~73 Sv/h Water level measurement: about 0.3 m 3rd time: Atmospheric radiation levels: about 24 Sv/h~36 Sv/h Collection of accumulated water, Installation of permanent monitoring instruments 4th time: Atmospheric temperature: 16.5°C Radiation levels: about 210 Sv/h* on the control rod drive rail *As a result of the validation implemented after the 4th investigation, this radiation levels are amended; 210 Sv/h to 70 Gy/h. • Sediment collection.
	Investigation inside the building	Investigation of the torus room wall (July 2014) Investigation of the pressure suppression chamber lower part (September 2014)	• No leak point from the PCV was found.
	Others	Muon measurement (March to July 2016)	• Confirmation on the presence of high-density substances considered to be fuel debris at the bottom of the pressure vessel, in the lower part of the core and in the outer periphery of the core.
Unit 3	PCV internal investigation	1st internal investigation (October to December, 2015)	• Damage to the structures or wall inside the PCV was not found, and internal radiation levels were confirmed to be lower than those of units 1 and 2. • Confirmation of temperature, radiation levels, water level inside the PCV. Atmospheric temperature: about 26~27°C Radiation levels: about 1 Sv/h Water level: about 6.3 m Collection of accumulated water Installation of permanent monitoring instruments
	Investigation inside the building	Investigation of the PCV equipment hatch (September and November, 2015)	• Confirmation of contamination and rust around the equipment hatch in lower level areas due to the water level in the PCV.
		Investigation of the main steam isolation valve room (May, 2014)	• Confirmation of leakage from the main steam pipe expansion joint as a leakage point from the PCV.
		Investigation of the inside PCV by submersible remote operated vehicle (July,2017)	• Confirmation of multiple damaged substances and some part of the control rod drive housing supporting clamp inside of the pedestal. • Confirmation of likely melted materials that are solidified, some fallen substances such as grating and sediments.

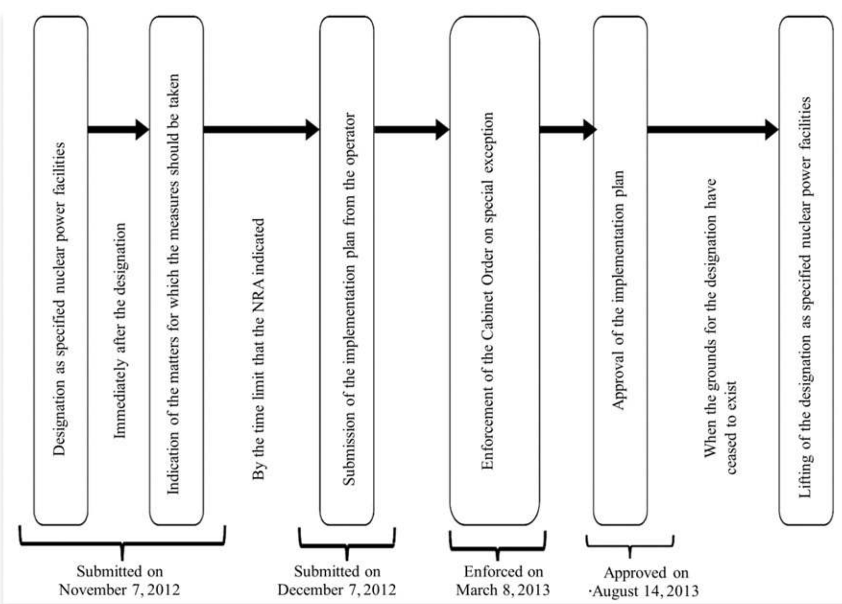
Regulatory framework and implementing long-term management plans

After the tsunami had induced the accident at the Fukushima Daiichi NPP, emergency measures were taken to cope with the resultant hazardous situation based on Article 64, paragraph (1) of the Reactor Regulation Act. The reason why it was necessary to revise the said Act adding Article 64-2, 64-3, and 64-4 was that the accident was beyond the framework of pre-existing Article 64 and that it should take quite long time for recovery or decommissioning.

Based on Article 64-2, paragraph (1), the Nuclear Regulation Authority (NRA) of Japan designated the Fukushima Daiichi NPP as specified nuclear power facilities, which are defined by this provision as facilities to be placed under special management, on 7 November 2012. Based on paragraph (2) of the said Article, the NRA required the relevant licensee to submit a plan to implement measures for the operational safety of the facilities (implementation plan), indicating "matters for which the measures should be taken" and the time limit therefor. The matters consist of those regarding the entire process and risk assessment, infrastructure, security, protection of nuclear material, fuel debris removal and decommissioning, planning of implementation, implementation, and inspection.

On 7 December 2012, the NRA received the implementation plan prepared by TEPCO based on Article 64-3, paragraph (1) of the Reactor Regulation Act. Subsequently, the NRA established the Supervision and Evaluation Committee for the Specified Nuclear Power Facilities to help investigate validity of the implementation plan and the status of the implemented measures through discussion with external experts. The Committee examined whether each facility described in the implementation plan, and each countermeasure, conforms to "the matters for which measures should be taken," or safety requirements, and other points using results of site inspection for reference. After a series of discussions of the Committee, it was confirmed that the submitted implementation plan conformed to the required goal. The NRA acknowledged that the implementation plan was sufficient to provide protection from nuclear fuel materials or contaminated objects as well as to prevent reactor-related disasters and protect the specified nuclear fuel materials, and approved the plan on 14 August 2013. The NRA also indicated precautions to be observed when implementing the plan, with regard to risk evaluation, monitoring reactors and equipment, fuel removal, storage of radioactive waste, countermeasures against contaminated water, radiation protection, emergency countermeasures, response to tsunamis, organisation structures, removal of fuel debris, and promoting public understanding to implement the plan.

Figure 2.41. **Implementation actions for specified nuclear facilities and progress**



Source: NRA (2013), www.nsr.go.jp/data/000067054.pdf.

Based on Article 64-4 of the Reactor Regulation Act, only part of the provisions of the said Act was decided to be applied to the Fukushima Daiichi NPP, as specified by the Cabinet Order established and enforced on 8 March 2013, as long as measures for operational safety are implemented in accordance with the implementation plan (Figure 2.41).

The provisions of the Reactor Regulation Act in English are available at www.nsr.go.jp/data/000067232.pdf.

Approaches to risk for long-term management

Risk mapping

In the "matter for which the measures should be taken" indicated by the NRA regarding the entire process and risk assessment, it is required to reduce and optimise risks of the specified nuclear power facilities as a whole, clarifying the entire process and evaluating each operation and stage for the entire process of measures towards completing decommissioning including removal and storage of melted or damaged fuel rods from units 1 to 4, and for the entire process to maintain cold shutdown stably for units 5 through 6. It is also required that risk assessment is applied to a large extent assessing the environmental impact in broad areas outside the plant, thereby ensuring that the reduction and optimisation of risk is sufficient for safety inside and outside the premises. The Supervision and Evaluation Committee for the Specified Nuclear Power Facilities has examined the plant status reported by the licensee regarding the completed/ongoing/planned measures against risks and the current situation. The committee has also asked the licensee to extract and rank risks in the whole facilities and to report on assessment of risk reduction.

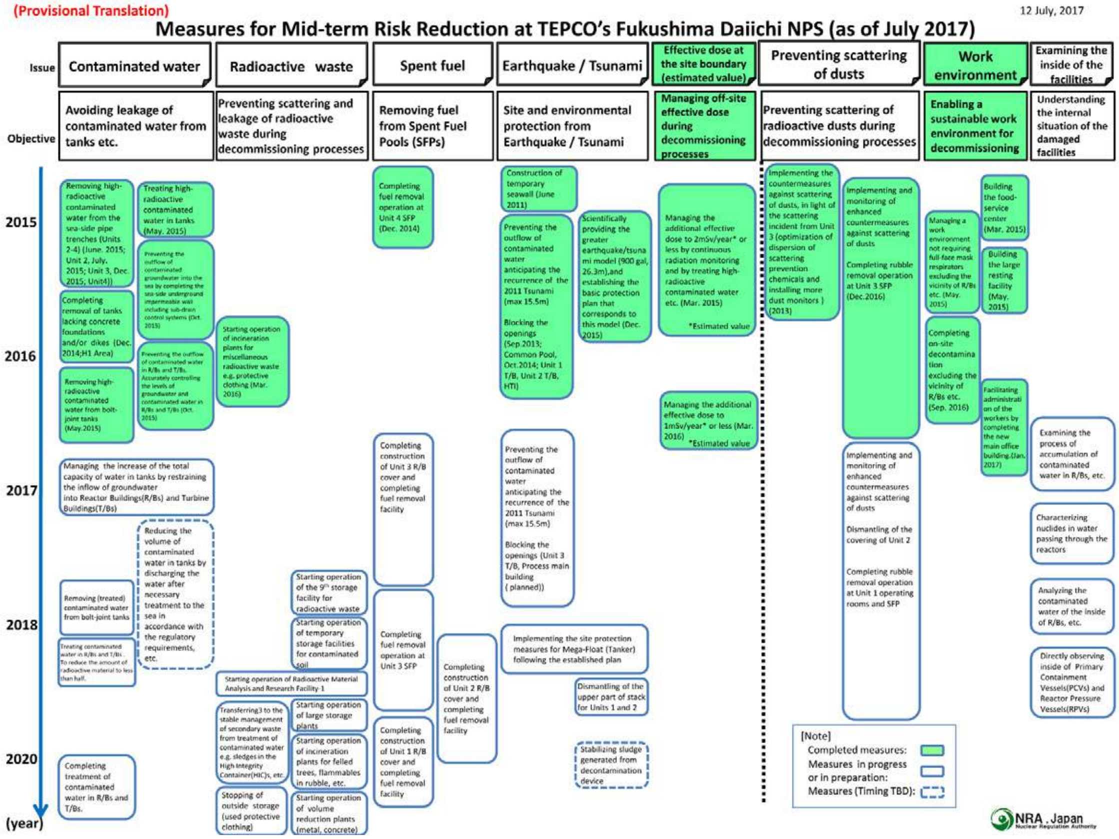
TEPCO has complied with this request conducting risk assessment to clarify objectives, identify risk matters, supposing accident scenarios or causing events, designing measures against major scenarios, validating the measures, and designing unimplemented measures against residual risk matters in the short and mid- and long term. The unimplemented measures were classified into five categories: (1) those for external events (e.g. earthquake/ tsunami or fire), (2) those for situations worsening in the near future (e.g. storage capacity for contaminated water and solid wastes), (3) those for situations triggered by implementing measures (e.g. groundwater levels or radioactive dust), (4) those for mid- and long-term management (e.g. spent fuel or radiation exposure), and (5) those for reliability of infrastructure (e.g. prevention or monitoring of leakage). TEPCO realised that the measures belonging to categories (1) to (3) should be implemented as soon as possible. The NRA accepted TEPCO's risk assessment requiring sustained efforts to reduce risks in varying situations, especially risks of contaminated water leakage.

Besides the mentioned risk assessment, the NRA suggested to compile a kind of risk map to visualise risk issues of the Fukushima Daiichi NPP affecting the environment. Such risk map should be designed to relate progress of safety operation to risk reduction clarifying completed measures rather than to quantify each risk source. Accordingly, the Secretariat of the NRA proposed a template for a map of measures for mid-term risk reduction indicating about five-year perspective regarding seven issues, five main issues plus two related ones: contaminated water, radioactive wastes, spent fuel pool, earthquakes and tsunamis, effective dose at the site boundary (estimated), work environment, and examining the inside of the facilities. The risk map, "Measures for Mid-term Risk Reduction at TEPCO's Fukushima Daiichi NPS", has been compiled with the following objectives:

- to present key priorities for safety identified by the NRA among the various measures undertaken by TEPCO for the decommissioning of the Fukushima Daiichi NPP;
- to clearly distinguish completed measures from ongoing and planned measures.

The progress made for each objective will be measured by the risk map, while this document will be regularly reviewed to reflect the state of risk reduction. The NRA has released the risk map five times until 12 July 2017, adding one issue on measures for preventing scattering of dust (Figure 2.42). The past editions of the risk map in English are available at www.nsr.go.jp/english/library/nraplans_01.html.

Figure 2.42. **Measures for mid-term risk reduction at TEPCO's Fukushima Daiichi NPP as of July 2017**



Source: NRA (2017), www.nsr.go.jp/data/000201934.pdf.

Risk assessment

The Decommissioning Office of the Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) is the organisation that was created to facilitate Fukushima Daiichi decommissioning. The strategic plan developed by NDF includes many aspects related to safe and prompt decommissioning of the plants: risk-driven decommissioning strategy, fuel debris retrieval strategy considering various options, radioactive wastes management strategy, research and development (R&D) plan for decommissioning, and organisational relationship between various actors (government, NDF, research institutes, TEPCO who is performing the on-site decommissioning).

NDF runs a fully comprehensive risk assessment, which can substantially be divided into two different approaches (NDF, 2016):

1. a risk reduction strategy aimed at prioritising LTM activities upon the level of risk presented by each type of radiation source;
2. a risk analytical approach based on splitting the LTM tasks into fuel debris or waste-related activities.

Risk reduction strategy

The first approach stems from adapting the UK Nuclear Decommissioning Authority's (NDA) so-called safety and environmental detriment (SED) score which is one of the scores used to prioritise activities at the 17 sites ruled by NDA (2011). Such score takes into account the radiation source in terms of stored radioactivity, how easily such radiation can be released in case of loss of containment function which depends on the physical phase, and the recovery time under generic threatening events (such as loss of cooling, loss of inerting, etc.). These three factors represent the consequence side of risk according to the following formula:

$$RHP = \frac{Inventory \cdot Form\ Factor}{Control\ Factor}$$

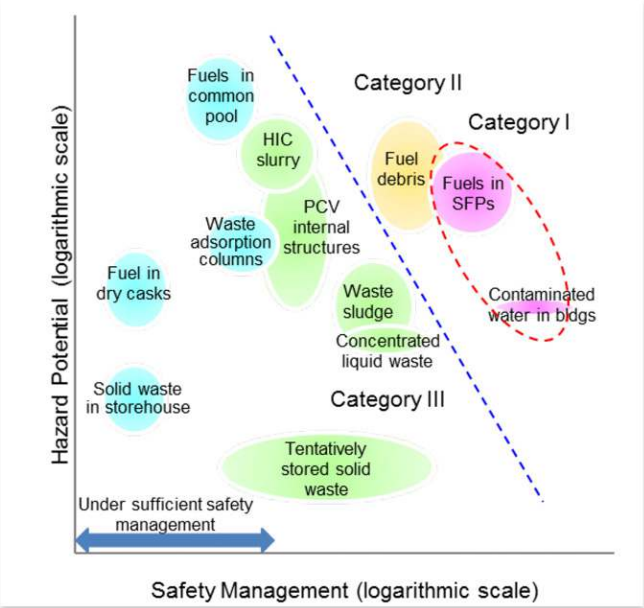
where *RHP* stands for *radiological hazard potential* and it is a measure of risk through the *inventory* variable which is proportional to the stored radioactivity by each source; *Form Factor* which depends on whether the radiation source is stored as a gas, liquid or types of solid; and *Control Factor* which attempts to give a measure of the available recovery time from a degraded situation.

On the likelihood side of risk, two factors help quantify the priority with regard to management to maintain safety: a first, *Facility Descriptor* or *FD* index categorises the suitability of containment by focusing on the status of barriers between the radioactive source and the environment. According to the status of the containment, a specific *FD* is assigned. A second, *Waste Uncertainty Descriptor* or *WUD* determines whether the radioactive source is degrading or chemically reactive and how monitored, managed and packaged it is in order to categorise the likelihood of deterioration of the source material such that future removal will be more onerous, ending up in the following SED formula:

$$SED = RHP \cdot (FD \cdot WUD)^4$$

where Chemical Hazard Potential similar to RHP is disregarded. Identified radioactive sources comprise the fuel debris inside the PCV of the three units, plus the fuel elements stored in all the SFPs including the common pool and dry casks, the contaminated water distinguishing the source upon a different location is identified, and all types of secondary and solid wastes. A SED score is then calculated for each of these sources for a series of major radionuclides and then compared in terms of the two main risk components of consequence or hazard *RHP*, and likelihood or safety management $(FD \cdot WUD)^4$ according to Figure 2.43.

Figure 2.43. **Ranking of hazard and likelihood risk components at Fukushima Daiichi**



Source: NDF (2016), www.dd.ndf.go.jp/en/strategic-plan/book/20170322_SP2016eFT.pdf.

Therefore, this comprehensive approach to risk allows facing different risk sources at once and revealing where higher risks are stored with the scope of prioritising LTM actions. In this sense, such approach does not yet enable the user identifying and quantifying the specific sources of risks specific of every risk source. This is why a second different strategy is necessary.

- Analytical approach to risk

The proposed analytical approach consists of identifying and categorising the different actions addressing the LTM within the two main broad types of issues of fuel debris and waste management. Such analytical task will be performed through a three-step approach:

1. A preliminary step based on analysing every action and identifying how to specifically address it at best.
2. Development of a so-called Logic Tree as a chart tool that easily allows identifying and arranging the needed tasks under a well-ordered structure.
3. Application of a generic risk method tool such as failure mode and effects analysis or IAEA-developed DRiMa (Decommissioning Risk Management) devoted specifically to decommissioning issues.

When making such analysis, it is essential to take into account all types of risk factors during the decommissioning activities not only dealing with risk reduction or safety, but also with workers' health, environment, security or financial affairs, as recommended by DRiMa or the NDA Value Framework.

- Long-term management for risks reduction in relation to plant damaged states

This section describes the progress of LTM implemented at the Fukushima Daiichi damaged site. In the early stage after the accident, the focus was primarily on recovery of critical safety functions and necessary systems and components. Further actions were then implemented to enhance robustness and monitoring of these critical safety functions, manage produced liquid and solid wastes, remove stored fuels from SFPs, implement countermeasures against external hazards, improve working environments, reduce dust dispersion and the site boundary dose rate, perform site inspections and prepare for decontamination and defuelling of the damaged RPVs and PCVs. LTM planning was driven by reductions in highly identified risks depending on the plant damaged state. Progresses in recovery actions have changed the plant damaged state by reduction of some risks and thus the LTM focus with time.

- Monitoring and strengthening of safety systems for long-term management

LTM management actions after reaching a stabilised controlled state are primarily designed to fulfil the three following "critical" safety functions:

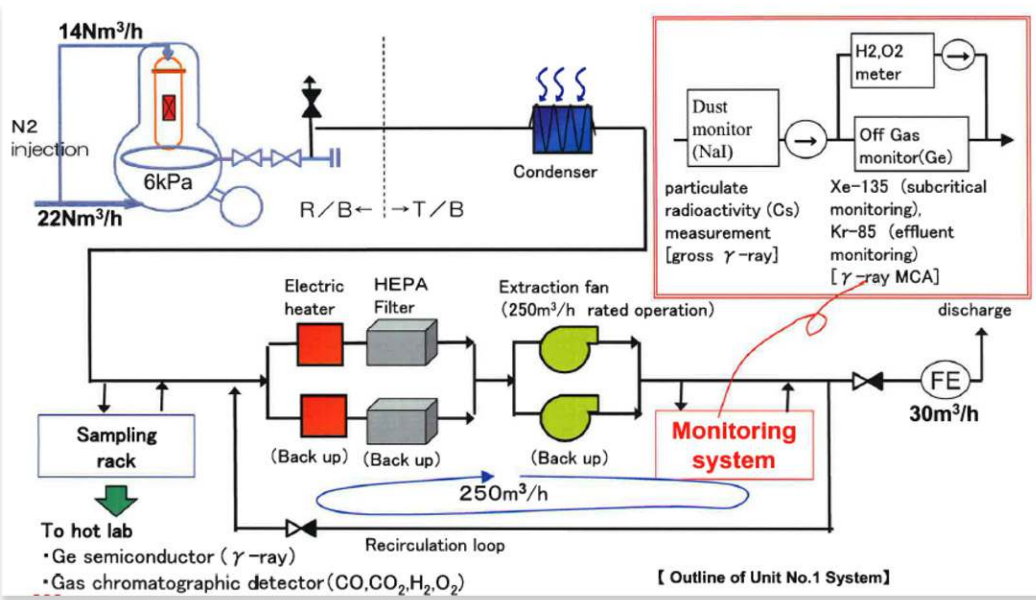
- maintain the sub-criticality;
- maintain the cooling in the RPV and in the PCV;
- maintain the controlled release.

The sub-criticality in the three units containing degraded fuel is monitored every hour by ^{135}Xe detection in the PCV gas control system (Figure 2.44), by temperature evolution measurements of the RPV lower head and by air dose rate measurements by monitoring posts. The criteria have been defined for checking sub-criticality. If the observed value deviates from the criteria, measures would be implemented to provide borate injection through the liquid control system and eventually, once the liquid control system is exhausted, by seawater injection. Up to this date, no deviation from sub-criticality has been detected.

The cooling of the RPV and the PCV in the three units containing degraded fuel is ensured by a closed loop, injecting water in the RPV, using either the core spray safety system or the feed water system. Out-coming water is treated and re-injected in the core to limit volumes of water wastes (see the next section). Cooling is monitored by temperatures of the RPV lower head and the PCV atmosphere, which should stay below 80°C. One issue in monitoring was the reliability of thermocouples after they had experienced severe accident conditions. Selection of reliable thermocouples had to be made based on analysis of their electric characteristics and response evolution. Monthly reliability analyses of thermocouples are performed. Temperatures and injected flowrates are checked every day. Flowrates for each reactor were reduced from 4.5 m^3/h to 3.0 m^3/h until early 2017.

The confinement of radioactivity remains a critical aspect as the containment leak tightness was lost during the accident in the three damaged units. As reported earlier, identifying all containment leakage paths is challenging. Atmospheric releases from the PCV are however reduced by high-efficiency particulate air (HEPA) filtration and monitored by γ-counting in the PCV gas monitoring system (Figure 2.44). N_2 injection to the PCV is done to prevent an H_2 explosion in the PCV; H_2 being produced by water radiolysis. H_2 and O_2 concentrations are measured in the gas monitoring system and it is checked that H_2 concentrations remain below limits that would permit combustion. N_2 flow and H_2, N_2 and O_2 contents are checked every day. Concerning waters used for RPV and PCV cooling, their volume is reduced as much as possible to reduce the amount of contaminated water to be treated (see the next section).

Figure 2.44. **PCV gas monitoring system at Fukushima Daiichi reactors**



Source: TEPCO (2018).

All units have been maintained in a cold shutdown state, with the cooling being maintained by a 0.8 km long closed water loop. SFP cooling is ensured via existing and newly installed facility.

As of April 2017, the temperatures are maintained between 15 and 35°C. Concentrations of radioactive gas (^{135}Xe) are not showing any significant changes and are close or below detection limits of the measuring devices. No sign of abnormality in cooling nor criticality was observed. These observations confirm that all three units are in a stable cold shutdown state.

Contaminated water management actions

The Fukushima Daiichi NPP is located downstream from abundant groundwater flowing towards the Pacific Ocean from mountainous areas on the west side of the power stations. Before the tsunami and the subsequent severe accident occurred, approximately 730 m^3 of groundwater had been pumped up every day through the sub-drain system, a group of wells, in order to control levels of groundwater around the reactor buildings and the turbine buildings of units 1 through 4. Since the sub-drain, this pumping equipment and the buildings were damaged, a large quantity of groundwater, which is combined with rainwater in rainy conditions, has entered the buildings and been subjected to contamination. Both the contaminated water and cooling water injected into the RPVs of units 1 through 4 are collected as a whole. About half of the contaminated water is injected back into the RPVs to cool the fuel debris, while the rest is stored in water storage tanks after getting treated to decrease concentrations of radioactive materials (the amount of added water was approximately 400 m^3/d as of January 2014; it is reduced to 220 m^3/d as of April 2017).

On 19 June 2013, TEPCO reported to the NRA that the concentration of tritium in groundwater near the revetment between seawater inlets of units 1 and 2 at the east side of the turbine buildings had increased to 500 kBq/L, a level ten times higher than that measured in December 2012, while that of strontium also had increased to 1 kBq/L which is much greater than the limit stipulated in the laws and regulations. TEPCO believed that previous fallouts were unlikely to have affected these concentrations and that the highly contaminated water in the trenches on the seaward side, which are underground tunnels designed to store pipes and cables, had leaked underground and mixed with the groundwater. The NRA established the Working Group on Contaminated Water Countermeasures under the Supervision and Evaluation Committee for the Specified Nuclear Power Facilities to examine the state of leakage at the site as well as TEPCO's implementation plan of countermeasures against leakage. The NRA instructed TEPCO to strengthen its monitoring of contaminated water, immediately implement countermeasures to prevent the contaminated water from flowing into the marine environment, immediately implement countermeasures to prevent leakage from the trenches on the seaward side containing highly contaminated water, which may have leaked underground, and implement any other actions deemed necessary.

TEPCO has three fundamental principles on contaminated water management: i) to remove sources of contamination, ii) to redirect fresh water far from contaminated areas, and iii) to retain contaminated water away from leakage to the environment. Several major management measures have been implemented as follows.

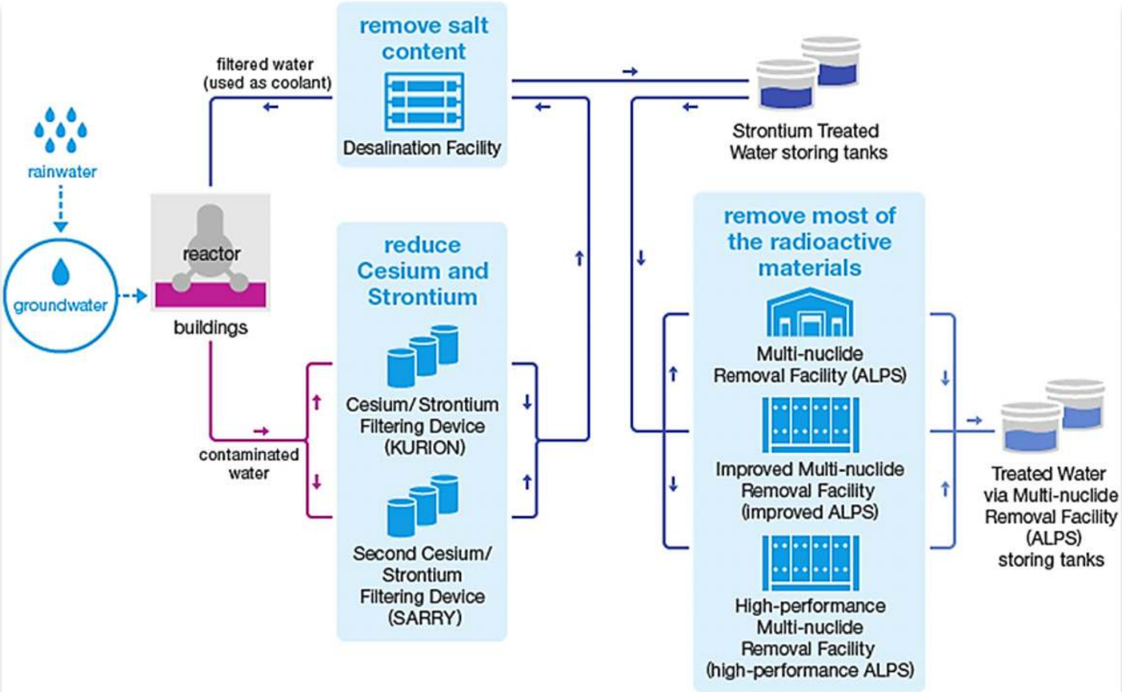
Multiple facilities were put into operation for sequential treatment processes of contaminated water accumulated in the premises (Figure 2.45). First, several removal systems including KURION, SARRY and mobile devices reduced caesium and strontium in contaminated water. After subsequent treatment for desalination, about half of the caesium/strontium treated water is injected back into the RPVs. The rest is treated to remove most of the radioactive materials except tritium at a multi-nuclide removal facility with an advanced liquid processing system (ALPS). The ALPS treated water is finally stored in storage tanks. ALPS was confirmed to have the capability to reduce concentrations of 62 nuclides under the designated concentrations limits. On 27 May 2015, treatment of most of the highly contaminated salt water stored in storage tanks was completed. From this point, ALPS treated water and caesium/strontium treated water have been stored in the station and they have been accumulated up to approximately 1 020 000 m^3 as of October 2017. TEPCO has been expanding the storage capacity replacing existing flange type tanks with welded type ones to realise higher reliability.

In December 2015, removal of highly contaminated water accumulated inside the seawater piping trench connected to the seaside of the turbine buildings of units 2 through 4 was completed. Considering the quality and quantity of water contained in the tunnels or vertical shafts of the trench, contaminated water was transferred to the turbine buildings. The inside of the trench was also filled with grout materials. The operation was completed in March 2017.

TEPCO has implemented countermeasures against contamination of the marine environment: groundwater bypass, sub-drain and groundwater drain system, and seaside impermeable wall (Figure 2.46).

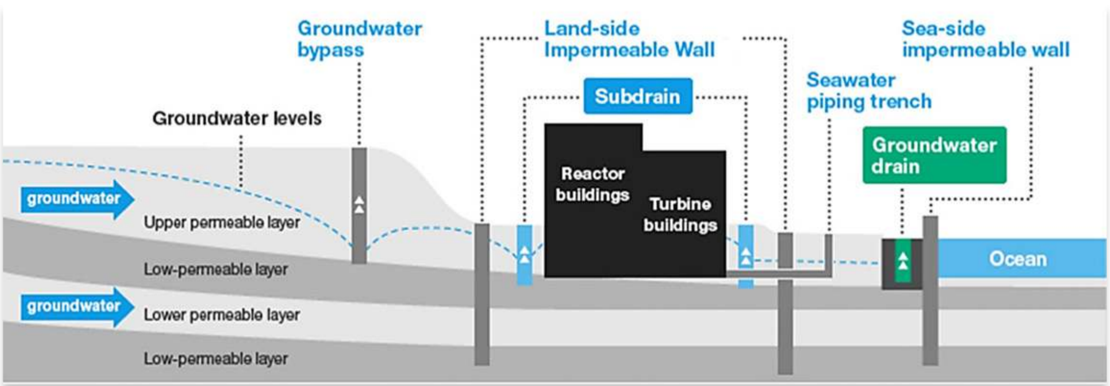
The groundwater bypass was implemented in order to direct clean groundwater to flow downhill towards the sea, bypassing the facilities. The water quality is monitored regularly to confirm satisfaction of the discharge criteria (less than 1 Bq L^{-1}, 1 Bq L^{-1}, 5 Bq L^{-1}, and 1 500 Bq L^{-1} for ^{134}Cs, ^{137}Cs, gross beta, and ^{3}H, respectively). The groundwater bypass has been in operation since May 2014, and was estimated to reduce groundwater flowing into the building basements by up to 100 m^3/d.

Figure 2.45. **Flow diagram of the contaminated water treatment**



Source: TEPCO (2018)..

Figure 2.46. **Countermeasures against contamination of the marine environment**



Source: TEPCO (2018)..

About a half of the damaged pre-existing pits had been restored and new ones had been added to the sub-drain system around the buildings of units 1 through 4. Groundwater has been pumped up since September 2015 in order to reduce the flow into both the buildings and the sea. The amount of groundwater pumped up, which is approximately 510 m^3/d through 42 pits as of 2017, has been decided considering the treatment capacity and keeping levels of the highly contaminated water inside of the buildings lower than those of surrounding groundwater, aiming to prevent the former from flowing into the latter. The collected groundwater has been purified at a treatment facility specifically designed for low-level contaminated water and stored in temporal storage tanks. After checking quality, the purified water has been discharged to the port area based on agreement with the stakeholders.

The seaside impermeable wall had been constructed along the coastline on 26 October 2015, aiming to reduce risks of contaminated water flowing into the ocean if any contaminated water leakage would occur in the premises. The wall is composed of 594 steel pipe sheet-piles driven into the ground deeper than the lower permeable layer across a length of about 780 m alongside units 1 through 4, and is expected to hold groundwater flowing into the port area from the premises. The accumulated water behind the wall has been pumped up through the groundwater drain, which consists of five pits, preventing overflow into the port, and treated together with the water collected through the sub-drain system.

A landside impermeable wall (ice wall) started operation in March 2016. The total length of the wall is about 1 500 m and the amount of frozen soil is about 70 000 m^3. So far, the amount of underground water including rainwater flowing into the buildings has been reduced to approximately 100-200 m^3/d.

- Solid waste management actions

A huge amount of waste was generated by recovering works from the emergency situation in a short time. Due to the limited capacity in waste storage buildings, a large part of waste was stored outside temporarily. The waste stored in the temporary storage will have its volume reduced as much as possible before moving to the solid waste building.

Solid waste generated in the course of decommissioning work is classified based on the surface dose rate. Rubble exceeding 30 mSv/h is stored in the waste storage buildings. Fallen trees, used protective clothing and rubble below 30 mSv/h are stored in temporary areas. The secondary wastes from the water treatment facility such as high integrity containers from the multi-nuclide removal facility (ALPS) are classified into categories like adsorption towers, sludge and concentrated waste liquid then temporarily stored. The solid waste incinerator started operation in March 2016 (300 kg/h x 2 trains) and used protective clothing is being incinerated.

The solid waste storage management plan was developed in March 2016 (revised in June 2017) as scheduled in the mid-long-term roadmap. The amount of wastes in the coming ten years is estimated at about 754 000 m^3 and it turned out that the estimated amount would exceed the current storage capacity. Construction of an additional solid wastes incinerator, volume reduction facility and storage building are planned. With these enhanced capacities, rubble volumes will be reduced as small as possible then it will be stored in storage buildings so that temporary storage will no longer be needed.

The secondary wastes from water treatment are temporarily stored outdoors now. It is planned to shift to storage in the buildings and decrease the temporary outdoor storage in the future. The studies for the stable storage will be also planned. Considering future changes in the decommissioning work plan which is the basis of quantity estimation of wastes, the solid waste storage plan will be revised once a year.

Details of radioactive waste are summarised in *Management of Radioactive Waste after a Nuclear Power Plant Accident* (NEA, 2016).

- Management of spent fuel pools

Spent fuel pools at units 1 to 4 are maintained under desalination and recirculation cooling. Water temperature is maintained stably and reliability of the cooling system has been enhanced. Currently, SFP cooling systems are installed at each unit but a plan has been developed to integrate them into a single system that will start operation in 2018.

Unit 4

Spent fuel removal at unit 4 was started in November 2013 and all the 1 533 fuel bundles were removed from the SFP in December 2014 (2 new fuel assemblies were earlier removed in 2012). 1 331 spent fuel bundles were transferred to the common pool and 202 new fuel bundles were transferred to SFP at unit 6. These removal operations were completed safely.

Unit 3

The unit 3 reactor building was damaged due to a hydrogen explosion. The fuel handling machine had fallen in the SFP and other equipment was scattered on the refuelling floor.

The dose rate on the refuelling floor was as high as several hundred mSv/h because of scattered rubble. Then, the rubble including the fuel handling machine and the other equipment were removed and decontamination was conducted. In April 2016, the dose rate has been reduced to less than 5 mSv/h in the majority of the refuelling floor by shields installation.

A fuel removal cover for the SFP was pre-assembled into several large components at Onahama, located about 60 km south of the Fukushima Daiichi NPP in order to reduce the on-site exposure dose by the construction work. These components were transported to the site and are being assembled on-site as of January 2018. The fuel removal work is planned starting from mid-2018.

Unit 1

The reactor building cover was dismantled to prepare for rubble removal on the refuelling floor. The dismantling work started in May 2015 with spraying of an anti-dust agent. Roof panels were removed and investigation was conducted on the refuelling floor. In April 2016, the water spray equipment was installed to prevent possible radioactive dust dispersion.

Investigation on the refuelling floor is being conducted. The status of fallen ceilings, the reactor well plug and the overhead crane has been established. This information will be used in planning future rubble removal works on the refuelling floor.

Unit 2

To facilitate removal of fuel assemblies and debris from the spent fuel pool, it was decided to dismantle the whole rooftop above the highest floor of the reactor building. The objective is to remove the fuel as fast as possible to reduce radioactive release risks during the removal process.

▪ Countermeasures against external hazards

External hazards considered at the Fukushima Daiichi NPP are earthquakes, tsunamis, fires, severe rain, typhoons and tornadoes. It is important to note that the decay heat of spent fuels and fuel debris has decreased significantly and the risk associated with radioactive material release to the environment has also decreased. In particular, most of the volatile radioactive gases and iodine have already decayed. The time margin in the case of suspended cooling has significantly increased.

In the process of decommissioning, risk sources will be removed and reduced. On the other hand, some risks associated with those factors such as spent fuels, fuel debris, waste storage facilities, contaminated water stored in tanks and buildings will remain on-site.

Earthquakes and tsunamis

There are two sets of seismic accelerations and tsunami heights which were revised after the accident. One is the design basis earthquake and tsunami: 600 gal and 15 m. The other is the beyond design basis earthquake and tsunami: 900 gal and 26.3 m.

Structural integrity of the reactor buildings containing spent fuels in SFP and fuel debris is evaluated to be maintained against the design basis earthquake and the beyond design basis earthquake (600 gal and 900 gal). At the same time, reliability of flexible emergency measures with portable equipment needs to be enhanced.

Outflow of contaminated water accumulated in the buildings needs to be addressed in case of the design basis tsunami of 15 m based on the size of opening and inventory with consideration on the progress of water treatment and relative risks to the environment. Since measures against the tsunami of 26.3 m will take long time, contaminated water treatment and studies on safe transferring/treatment of radioactive sludge are underway in order to further reduce risks.

Fires

In order to protect the site from external fire events, a firebreak is established in the vicinity of important facilities. The firebreak is 30 m in width taking into account the results of "Impact evaluation on forest fire to Fukushima Daiichi NPS" (JNES-RC-2012-0002). Also, the on-site fire brigade will battle the fire on the site using fire trucks and sprinkler trucks as an initial response in case it is anticipated that the fire may spread into the site.

Severe rain and typhoons

High-level radioactive contaminated water is contained inside reactor buildings, turbine buildings, a rad waste building, a process main building, control buildings, a high temperature incineration building, an incinerator and machine shop building and spent fuel pool building. These buildings are designed in accordance with the Building Standards Act which sets conservative assumptions on loads acting on the building from the past severe wind records. Although it is unlikely that the buildings would lose their function due to severe rains and typhoons, level-raising of accumulated contaminated water at the underground of these buildings was evaluated as follows:

- Observation records near the Fukushima Daiichi NPP show the maximum annual precipitation is below 2 400 mm, maximum daily precipitation is 285 mm and maximum monthly precipitation is 634 mm at Namie town.
- Assuming conservatively that daily precipitation would be 1 000 mm and the water level in the units 1 through 4 buildings is evaluated to raise at T.P. 2 311 mm that remains below the level (T.P. 2 564 mm) at which the water might leak outside.

In conclusion, even under hypothetically conservative maximum precipitation, it is confirmed that the contaminated water in the buildings would not leak out.

Tornadoes

Since the buildings containing contaminated water are constructed with reinforced concrete structures, direct damage to the buildings caused by tornadoes is unlikely.

Pumps for water injection into the RPV and the PCV are dispersedly located on the high land, in the turbine buildings and the high temperature incineration building. A risk to lose functions of all these pumps by a single tornado is considered to be small. However, if the time to restore water injection is judged to be long, fire trucks are to be deployed to inject water. Fire trucks and materials necessary for temporary water injection are also dispersedly located so that they will not be lost at one occasion.

As for the SFP, it will be protected by practical measures such as installing a cover on the surface. In the case of pool water leakage, mitigation measures will be taken with emergency motor driven pumps, fire trucks and concrete pump trucks.

Water treatment equipment and electrical equipment are located in the reinforced concrete buildings and the risk of direct damage caused by a single tornado is also unlikely. If all power buses become unavailable, dedicated generators installed near each facility will provide necessary power. If power trucks are available, they will be deployed to supply power to important equipment.

▪ Site boundary dose rate, mitigation of dust dispersion, improving working environments

Monitoring results

TEPCO has enhanced monitoring of radioactive concentration in the sea water in and out of the port since March 2011. Although the reported concentration exceeded legal limits just after the accident, it has decreased over time and currently it is maintained below the legal limits even inside the port. The concentration level monitored in the sea water intake in front of units 1 through 4 showed significant decrease after completion of the seaside impermeable wall.

Dose evaluation at the site boundary

TEPCO reported that the effective dose attributed to radiation from rubble stored on-site, contaminated water in tanks and radioactive materials released to the environment (liquid and gas) decreased below 1 mSv/y at the site boundary in March 2016.

At the end of March 2016, TEPCO estimated the maximum total site boundary dose rate as 0.96 mSv/y. The breakdown is as follows: radioactive gases contributed 0.03 mSv/y, direct and sky shine from various on-site facilities contributed 0.65 mSv/y, liquid waste discharge contributed 0.22 mSv/y, and water spray using treated rain water from dyke surrounding tank areas contributed 0.066 mSv/y.

Mitigation of dust dispersion

Multiple measures were in place to protect the environment and worker health for the unit 1 building cover removal work by preventing the scattering of contaminated dust. Anti-scattering agents were used to keep dust down, small pieces of rubble that can create dust were vacuumed, and mist sprinklers were also used.

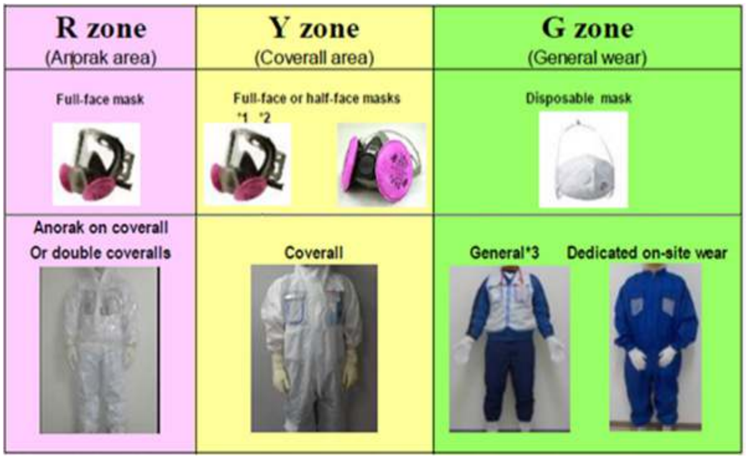
Work environment improvement

Since July 2014, the average number of daily workers ranges from about 5 500 to 7 000. As of March 2017, the local hiring ratio is about 55%. It was reported that the worker dose (including TEPCO and contractors) was high just after the accident but it decreased rapidly. It was reported that the monthly average dose was 0.30 mSv in August 2017 that is significantly lower than the monthly limit of 1.7 mSv (equivalent to the regulatory requirement of 20 mSv per year averaged over five years).

After removal of high dose rubble, surface soil and paving, as for the on-site radiation level (except for units 1 through 4 area and waste storage areas), it was reported that the target of 5 μSv/h has been achieved in a majority of the on-site areas (it was about 3 mSv/h before these works).

TEPCO reorganised on-site zoning of radiation controlled areas and optimised personal protective equipment as of 8 March 2016 (Figure 2.47). As a result, general wear or dedicated wear for on-site works is adopted in about 90% of the site area. On-site work environment improvement continues to further optimise personal protective equipment requirement.

Figure 2.47. **Optimised personal protective equipment**



Notes: *1. For works in buildings including water treatment facilities (multi-nuclide removal equipment, etc.) (excluding site visits), wear a full face mask; *2. For works in tank areas containing concentrated salt water or Sr-treated water (excluding works not handling concentrated salt water, etc., patrol, on-site investigation for work planning, and site visits) and works related to tank transfer lines, wear a full face mask; *3. Specified light works (patrol, monitoring, delivery of goods brought from outside, etc.).

Source: TEPCO (2018).

In order for workers to easily check the radiation dose rate at the work field, 86 radiation monitors were installed (as of March 2016) and real-time measurement can be observed on large screen monitors in the seismic isolation building and access control building.

Welfare facilities are also being improved. Worker surveys are periodically conducted and improvements have been made in reflection of needs from workers.

A large rest facility with a cafeteria with a capacity of 1 200 workers started operation in May 2015, and a convenience store opened in March 2016. A meal service centre in Ohkuma town started operation in June 2015 and provides foods to the on-site cafeteria. A new office building opened in October 2016 and about 1 100 employees are working in this building.

- Site inspection

Based on Article 64-3, paragraph (7) of the Reactor Regulation Act, TEPCO is required to undergo inspections conducted by the NRA to confirm whether measures for the operational safety of specified nuclear facilities or for the physical protection of specified nuclear materials are implemented in compliance with the implementation plan. The NRA has often conducted inspections with regard to starting-up of systems, checking of infrastructure, and security.

In addition to the said usual inspections, the NRA conducted special ones when necessary. On 19 August 2013, TEPCO reported to the Secretariat of the NRA that contaminated water, which was estimated to be up to 300 m^3, had leaked out of the weirs installed around a storage tank area for contaminated water, and that the contaminated water seemed to be released from a flange type water tank but the exact leakage path could not be identified. The NRA carried out a site inspection on 23 August to observe the situation as well as TEPCO's response to the incident. Based on the results, the NRA instructed TEPCO to assess the contamination outside the weirs, to identify the leakage paths as quickly as possible, to strengthen monitoring of the perimeter of the said tank area, to remove the soil contaminated by the leaked water, and to investigate outflow of the contaminated water into the ocean.

TEPCO subsequently removed the contaminated soil and undertook an environmental assessment through close monitoring, and also implemented countermeasures to prevent spread of leakage, such as replacing the old tanks with welded tanks, storing water flooded inside the weirs including rainwater in temporal storage tanks, raising the weirs, and coating the concrete floors under storage tanks. Implementation of these countermeasures was confirmed by the Fukushima Daiichi nuclear safety inspector's office and by the Working Group on Contaminated Water Countermeasures.

- Preparation of defuelling actions

Many efforts and resources are now placed on the preparation of defuelling actions with many actors and initiatives in Japan.

In order to address the profound challenges arising from the accident, tasks involving interconnected, diverse and evolving conditions were embodied in waste management and decommissioning and a new branch of TEPCO, called the Fukushima Daiichi Decontamination and Decommissioning Engineering Company (FDEC), has been created; this is focusing on contaminated water countermeasures, compilation of international knowledge on debris and fuel removal, and progression towards long-term resources for the decommissioning. Much information is being published on a ministry website (see www.meti.go.jp/english/earthquake/nuclear/decommissioning/index.html).

The establishment of the NDF strategic plan faced challenges as the access to contaminated areas and in particular PCV and RPV is difficult and information on the PCV and RPV situation is still too uncertain. A few more years will probably elapse before fuel debris retrieval will start, with significant R&D on debris characterisation and on retrieval methods and tools in progress. Figure 2.48 summarises, for instance, the present strategy proposed by NDF for investigation of degraded fuel in-vessel and how it interacts with the decision and planning of retrieval operations.

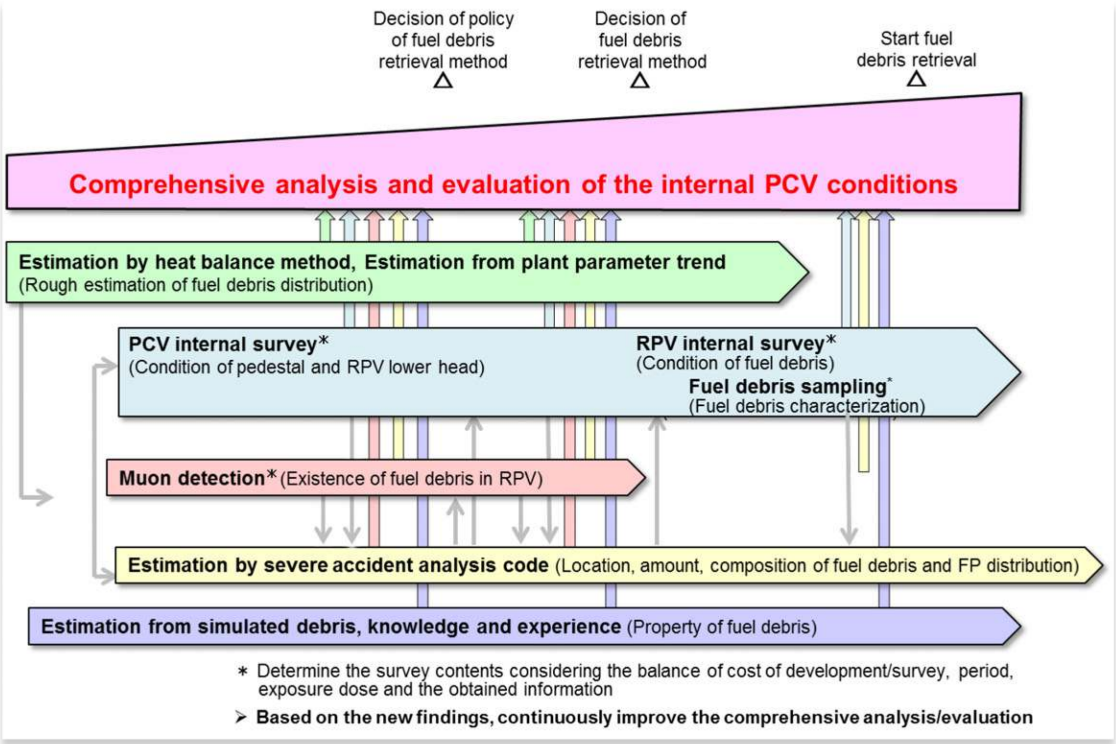
Two main options facing different challenges have been discussed for fuel debris retrieval:

- Degraded fuel retrieval under water: this would ease fuel cooling, prevent dispersion of radioactive material during retrieval operations and provide shielding against high dose rate but would face major challenges such as repairing the PCV leaks for submersion, maintaining sub-criticality during submersion, ensuring the submerged structures mechanical resistance (in particular to seism), and managing the contaminated waters after flooding.
- Degraded fuel retrieval in partial submersion conditions: the approach would face major challenges such as less-shielded high dose rates, risks of radioactive dust dispersion at the time of degraded fuel retrieval, and device survivability to high dose rates.

A risk-informed approach is used for safe fuel debris retrieval incorporating in-core inspection results and up-to-date R&D results as illustrated in Figure 2.49. Current knowledge and technology may recommend that the efforts be focused on the partial submersion method which is more feasible than the submersion method due to difficulty in repair of the top of the PCV.

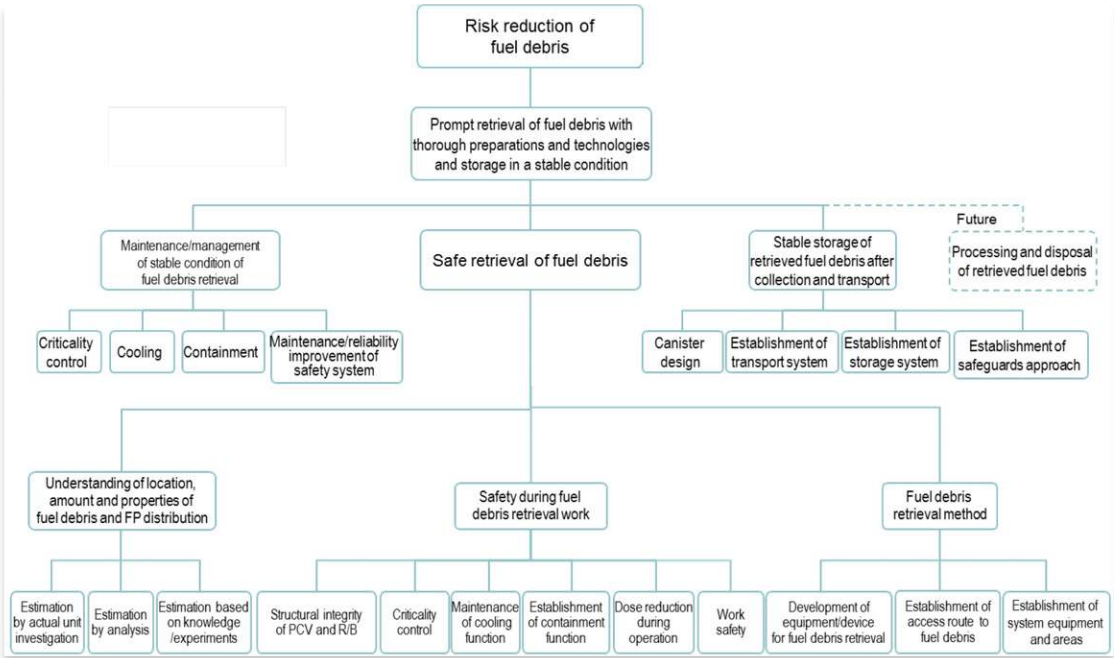
The Supervision and Evaluation Committee for the Specified Nuclear Power Facilities established by the NRA has also investigated precautions for safety involved in defuelling actions examining TEPCO's specific implementation plans and assessing required safety measures.

Figure 2.48. **Strategy for debris characterisation**



Source: NDF (2016), www.dd.ndf.go.jp/en/strategic-plan/book/20170322_SP2016eFT.pdf.

Figure 2.49. **Logic tree diagram for reduction of risks related to fuel debris retrieval**



Source: NDF (2016), www.dd.ndf.go.jp/en/strategic-plan/book/20170322_SP2016eFT.pdf.

▪ Summary of main implemented countermeasures

Reactor cooling

Multiple and diverse cooling facilities were installed shortly after the earthquake and the facilities have been upgraded since then so that stable cooling conditions could be maintained.

The decay heat in the reactor fell down to 1/100 or less of that at the earthquake time and it is evaluated that even if the cooling facilities are damaged due to a large-scale earthquake or tsunami and re-cooling requires 12 hours, the effective dose at the site boundary would not exceed 1.5 x 10^{-6} mSv/y. This dose is 500 000 times less than that measured six months after the earthquake.

The cooling status of the spent fuel pools (SFPs) has been maintained in a stable manner through a long period of time, and it is evaluated that even assuming that the cooling facilities are temporarily lost, the time delay until the water temperature of the spent fuel pool would reach 100℃ is more than 200 hours, even in the most severe case of the common pool.

Spent fuel removal

Spent fuels from unit 4 were removed in December 2014. Currently, equipment operations are being carried out for the cover and fuel handling system for removal of unit 3 spent fuel. A plan is being carried forward so that the fuel can be removed safely and steadily for units 1 and 2.

Contaminated water treatment

In the early stages after the earthquake, measures were taken against contaminated water occurring due to the inflow of groundwater, but multiple contaminated water treatment facilities were put in service after the autumn of 2014, and in May 2015, the treatment of all concentrated seawater was completed. Furthermore, for the highly-concentrated contaminated water remaining in the seawater piping trench of units 2 and 3, the accumulated water was transferred by August 2015, and the trench filling work was finished.

During contaminated water treatment, various measures were taken based on the three policies: i) to remove sources of contamination, ii) to redirect fresh water far from contaminated areas, and iii) to retain contaminated water away from leakage to the environment, and it was confirmed that results, such as reduction in the amount of contaminated water generated everyday due to the groundwater, were steadily achieved.

Radioactive waste

The waste, such as scattered rubble, generated after the earthquake and tsunami, is temporarily stored in accordance with the dose rate, and the secondary waste generated during contaminated water treatment, such as sludge, slurry and adsorption towers, are stored in appropriate storage facilities taking into account their characteristics.

The amount of waste is reduced by incineration and volume reduction and is consolidated by storage inside the building.

A "Waste Storage Management Plan" was established in March 2016 (revised in June 2017) in which the amount of solid waste that would be generated in the next ten years was estimated and the necessary waste-related facilities were studied. Hereafter, the progress of work will be incorporated into the plan, and the estimation of the generated amount of solid waste will be reviewed and updated once a year. More details on waste management planning are provided in *Management of Radioactive Waste after a Nuclear Power Plant Accident* (NEA, 2016).

Radiation protection

By implementing various measures, effective dose on the site, except for fallout (radioactive materials that had been released at the time of the Fukushima Daiichi accident and that are still in the environment), has reached less than 1 mSv/y at the end of 2015. After that, the dose has been maintained below 1 mSv/y.

Inside the port, the radioactive concentration has been reduced with time since the earthquake, and after the installation of the seaside impermeable wall in October 2015, a further reduction has been measured.

To control the dispersion of radioactive dust outside the site, measures such as spraying of anti-scattering agents in work areas are being taken and in addition to the existing monitoring posts, monitoring has been enhanced by the installation of continuous dust monitors on the site and on the refuelling floor where there is a possibility of dust dispersion.

Earthquake and tsunami countermeasures

Measures including the application of flexible response to the design basis earthquake ground motion 600 Gal, and earthquake ground motion for study 900 Gal, are taken.

For outer rise tsunamis, protection was implemented through a temporary seawall. Measures such as aperture sealing are being taken for the highest-recorded 15.5 m-level tsunami.

Regarding the beyond design basis tsunami (26.3 m), measures for the treatment of water accumulated in the buildings will be undertaken in priority.

Improvement in work environment

With the beginning of usage of the main building and contractors building, the working environment of the employees and the contractor workers has been improved.

The on-site rest areas are also being expanded and added in a planned manner.

With the decontamination on the site progressing in a planned manner, TEPCO's monitoring data indicated that the target dose rate (5 µSv/h) was achieved in a wide area on-site except for the vicinity of units 1 through 4. The controlled areas have been classified according to the status of contamination and the radiation protection equipment has been optimised.

Summary of main issues and challenges at Fukushima Daiichi

The accident that occurred at TEPCO's Fukushima Daiichi NPP on 11 March 2011, is different from TMI-2 and Chernobyl as it evolved into multiple units.

Regulatory framework

After the accident, the regulatory framework was revised to regulate facilities which were specified as "specified nuclear power facilities" and needed special management towards recovery or decommissioning. With regard to risk reduction and operational safety, the NRA confirms the validity of implementation plans and the status of the implemented and ongoing measures.

Approaches to risk

The NRA has constructed a risk map, "Measures for Mid-term Risk Reduction at TEPCO's Fukushima Daiichi NPS", to present key issues for risk reduction identified by the NRA among the various measures being undertaken by TEPCO, and to indicate the progress of measures for risk reduction.

NDF has run a comprehensive risk assessment composed of a risk reduction strategy and a risk analysis approach.

Management of contaminated water

In order to continuously reduce the risk of water leakage from the buildings and the storage tanks to the environment caused by some future potential event such as an earthquake, tsunami or ageing, some countermeasures are necessary.

It is necessary to make untiring efforts for risk reduction measures such as replacing flanged-type water tanks with welded type (high reliable type) ones and suppressing groundwater ingress into the buildings by a combination of various water management measures such as groundwater bypass, the sub-drain system, and impermeable walls.

Fuel removal from spent fuel pools

The spent fuel removal work is planned by TEPCO in the order of unit 4, unit 3 and units 1 and 2 taking into account the risks and difficulties.

The spent fuel removal work at unit 4 was completed in December 2014. The knowhow obtained through the work at unit 4 will be reflected in the safe work at unit 3 and the following units.

At unit 3, rubble removal and decontamination/shielding installation works have been performed by November 2017. The fuel removal building cover is now being constructed and the preparation works for the removal operation and maintenance are ongoing in parallel.

As for unit 1, the building cover, which had been built in 2011 for preventing radiation material from dispersing to the environment, was dismantled.

Rubble removal works are necessary before starting the fuel removal work.

Major challenges of the work at unit 1 are countermeasures to prevent dust dispersion during the rubble removal works, careful handling of large rubble such as the destroyed overhead crane over the pool and careful handling of spent fuel bundles.

As for unit 2, the integrity of the reactor building structure remains relatively intact because no hydrogen explosion occurred there. However the radiation dose rate in the reactor building including the refuelling floor is still high and the overhead crane and the fuel handling machine are considered to be useless taking into account the conditions after the accident such as long-term exposure to high temperatures and humidity.

So, TEPCO plans to dismantle the upper floors of the reactor building and will study some basic concept designs of the fuel removal cover in the future.

Fuel debris retrieval

The fuel debris removal work mainly consists of the following steps: i) investigations of RPV/PCV internal conditions, ii) installation of removal work equipment, iii) removal of obstacles to access, iv) debris cutting, v) transferring debris into containers and transportation, vi) analysis, and vii) storage.

Even though various investigations in the PCVs have been conducted and rough distributions of fuel debris were assumed so far, detailed locations and the conditions have not been confirmed as of January 2018.

So, the challenge for the time being is to survey and grasp more detailed and precise distributions and the conditions of the fuel debris by carrying out further investigations.

In addition to the major challenges of the fuel debris retrieval works themselves, various safety measures such as securing sub-criticality, cooling, shielding, anti-scattering measures and appropriate storage will be required as well.

Solid waste management

It is evaluated that a huge amount of waste will be generated in the next decade. So, the countermeasures such as waste volume reduction and increasing waste storage capacity are the key issues for future waste management.

With regard to the volume reduction measures, construction of incineration facilities and volume reduction facilities are planned. And additional waste storage buildings are also planned for the increase in the storage capacity.

Work environments

The decommissioning work will continue for a long period of time and it is necessary to reduce the workers' inconveniences and anxieties through continuous improvements of the work environments in order to secure a stable workforce in the future.

2.2. Status of long-term post-accident management and actions in NEA member countries

This Section was established based on answers received from utilities and safety authorities or their technical support organisations through a questionnaire that was distributed by the work group (provided in Appendix A). The questionnaire objectives were to review planned, envisaged or existing regulatory requirements, guidance and practices and to identify critical issues with respect to long-term management and actions for a severe accident. This section provides a summary of the collected information and is based on contributions from Belgium, Canada, Czech Republic, Finland, France, Germany, Japan, Korea, Slovenia, Spain, Switzerland, and United States. The summary starts with the summary of the existing regulation concerning the long-term management of the severe accidents in different countries. The following subsections summarise the identified critical issues and knowledge related to the long-term management.

Existing, envisaged or planned regulation for long-term management in NEA member countries

It is generally considered that the requirements and guidance that apply to emergency operating procedures/severe accident management guidelines (SAMGs), to emergency preparedness, to NPP siting, and decommissioning should also apply to post-SAMG long-term management phase. Following gives a short summary of the state of the regulation for LTM in the different countries which provided response to the questionnaire.

Belgium

There are no specific regulatory requirements for the long-term management of severe accidents, and currently, no regulatory requirements or guidance from the safety authority specifically for long-term management and actions for a severe accident are planned or anticipated. In the context of the action plan in response to the accident in Fukushima Daiichi, the licensee has, on demand of the Belgian safety authority, developed several measures to manage the accident in the long term, and to minimise the effects of the radioactivity. These measures include guidance on radiation protection, setting up a logistic support centre with a mobile unit, confinement of the radioactivity both as air and water releases, and management of water waste.

Canada

Canadian Nuclear Safety Commission (CNSC) has issued REGDOC-2.3.2 Version 2, Accident Management, in September 2015. In this regulatory document, although post-SAMG long-term accident management is not defined, general requirements and guidance are stipulated without a time frame. Implicitly, those requirements and guidance apply to the emergency operating procedure/SAMG phase as well as the post-SAMG long-term accident management phase. The following specific goals of accident management shall be achieved for any accident management phase including long-term accident management: i) minimise the release of radioactive materials into the environment, ii) achieve a long-term safe stable state of the reactor core or spent fuel storage.

Czech Republic

There are no specific regulatory requirements for long-term accident management. Regulatory requirements for long-term recovery post-accident phase follow from regulatory degrees on plant siting (SÚJB, 1997), on emergency preparedness (SÚJB, 2002), and from the degree on decommissioning of a nuclear installation (SÚJB, 2003). Current severe accident management is aimed to recover the damaged unit into controlled safe state, to verify coolability and non-criticality of the degraded fuel and to assess the extent of the damage.

Finland

In the Finnish regulations, most of the severe accident management requirements are not given for a certain time period, but are applied as long as the condition addressed by the requirement persists. Monitoring of the plant status and the environmental releases, molten core material cooling and containment integrity must be ensured under all conditions. As the overall safety goal, the Nuclear Energy Decree (1988/161) stipulates that a severe accident shall not cause acute harmful health effects to the population in the vicinity of the nuclear power plant, nor any long-term restrictions on the use of extensive areas of land and water. This requirement is in effect regardless of the accident duration, and it is satisfied if the probability of the total atmospheric release of ^{137}Cs exceeding 100 tera-Becquerel (TBq) is extremely small. Regulations of decommissioning and handling of nuclear fuel and waste, and the radiation dose limits for these activities, do not include separate requirements for decommissioning a damaged NPP after a severe accident. Review of the regulations which has been performed after accidents at Fukushima Daiichi has not resulted in new requirements for long-term management.

France

Arrêté INB (Nuclear Installation Decree), Article 3.07 states that the radioactive releases out of the installation as a result of an incident or an accident should be evaluated in a pessimistic way. To evaluate the radiological consequences of the radioactive releases, the demonstration of nuclear safety addresses the consequences in the short, medium and long terms, by considering the various ways of transfer of radioactive substances to the environment and people. This is translated to an objective of maintaining the containment in the long term, and an objective of limiting the long-term effects (permanent relocation, crop losses). The French Safety Authority (ASN) also imposed a set of requirements in response to the post-Fukushima Daiichi stress tests with the ambitious objective to reach for generation II reactors the safety objectives of generation III ones. LTM should benefit from important countermeasures implemented or in the process of being implemented by Electricité De France (EDF) in response to these requirements, e.g. the fast action force (FARN) with its mobile materials or ultimate electrical (bunkerised diesel generators) and cooling emergency systems.

Germany

There are no specific requirements with regard to the management of long-term severe accident situation on-site, but there are general requirements in the latest German nuclear rules for the management of severe accidents and there are documents for off-site management and protection of public. Preparations and measures are indicated to assess the situation, to alert the involved parties and to bring the plant back into a “safe and stable state”. In response to the accidents in Fukushima Daiichi, improvements have been made with regard to on-site accident management measures, mainly by additional mobile equipment and implementation of SAMG. Both German safety commissions, the “reactor safety commission” and the “radiation protection commission (SSK)” have issued in 2014 and 2015, respectively, two reports on updated requirements on accident management, emergency preparedness, radiation protection, protection of people in the vicinity of the plant, etc. but with no explicit requirements for long-term management.

Japan

Nuclear Reactor Regulation Law has a chapter about a special nuclear facility that needs special treatment for nuclear safety (Chapters 64-2). The NRA of Japan published long-term management requirements to the Fukushima Daiichi NPP based on the above law on 7 November 2012 including requirements for monitoring of RPV, residual heat removal, monitoring of containment atmosphere, keeping RPV and containment inert, defuelling, storage and management of fuels from SFP, securing power sources, designing measures for loss of power, treatment, storage and management of solid, gas and liquid wastes, radiation protection measures, emergency measures, design requirements for safety functions, and others.

Korea

There are no existing or planned regulatory requirements or guidance for the long-term management and actions in Korea. The general requirements and guidance are comprehensively and implicitly included within a framework of SAMG (implemented before the Fukushima Daiichi accidents) and accident management plan (to be implemented after the Fukushima Daiichi accidents) without a differentiation of time frame for a severe accident. The overall safety goal, regardless of the accident duration, is that the ^{137}Cs release of more than 100 TBq has a probability lower than 10^{-6}/yr.

Slovenia

National Emergency Response Plan for Nuclear and Radiological Accidents is in the process of revision. The Slovenian regulations are given in SNSA, 2009a, 2009b, 2010, 2011, 2015, and Government of Slovenia, 2010 and are all derived from the WENRA RL (2008). The requirements will be updated according to new WENRA RL (2014) that include design extended condition (DEC) A and B criteria. Currently, the Slovenian legislation does not distinguish between short-term and long-term management and actions for a severe accident except in the Act ZVISJV, where the requirements for the mitigation of the consequences of an emergency are set (after the emergency is brought under control).

Spain

Consejo de Seguridad Nuclear (CSN) has not issued any specific regulatory requirements for long-term management of severe accidents. After the accidents at Fukushima Daiichi, CSN requested provisions for management of contaminated water. This was addressed by implementing measures to store contaminated water and to safely transport the water from the site. In addition, a number of requirements for radiological protection of the workers were requested. These requirements are valid for any time after the occurrence of a severe accident. There are no immediate plans to implement new guidance of requirements for long-term management.

Switzerland

The Nuclear Energy Act states that licensees must make provisions against undue releases of radioactivity and radiation exposure of people. A regulatory requirement specifies that SAMG needs to cover all phases of a severe accident. In addition, long-term management will benefit from measures taken by the licensees after the accidents at Fukushima Daiichi: commissioning of an off-site storage facility and storage of additional severe accident management resources on-site.

United States

Long-term management of accidents in commercial nuclear power plants is the responsibility of plant owners and licensees. The scope of licensee actions include: i) bringing a plant to a stable state; ii) performing site clean-up and decontamination operations; iii) performing defuelling operation; iv) providing facilities and infrastructure for temporary and permanent storage of radiological waste forms; and v) providing protection of health and safety of plant personnel. The US Nuclear Regulatory Commission has the role of providing regulatory oversight to ensure public health and safety and that of the environment. Administration of this oversight function is codified in Title 10 of Code of Federal Regulations (10 CFR) including 10 CFR Part 20 (Standards for Protection against Radiation), 10 CFR Part 50 (Licensing power reactors), 10 CFR Part 50.47 (Emergency Plans), 10 CFR Part 51 (Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions), 10 CFR Part 71 (Packaging and Transportation of Radioactive Material), and 10 CFR Part 73 (Physical Protection of Plants and Materials). The lessons learnt from TMI-2 for long-term management have been factored in by the industry in its operational practices and by the NRC in its reactor oversight programme (ROP). In light of the Fukushima Daiichi accidents, the ROP is being examined for potential updates in areas related to mitigation of beyond design basis events.

Identified critical safety functions for long-term management

It was generally agreed among the contributors that the critical safety functions important to long-term accident management are:

- to control the reactivity;
- to remove heat from the degraded fuel, and to achieve and to maintain a long-term stabilised controlled state of the reactor core or spent fuel storage;
- to confine the radioactivity by controlling the operational discharges, as well as limiting accidental releases;
- to shield against radiation;
- to monitor the safety important parameters to guide operator actions.

Assessment of the plant damaged state and contamination after the event is necessary to allow planning of recovery actions. The main priority of recovery actions on-site is to maintain a controlled state for all units at the affected location during all LTM phases while minimising the staff occupational dose and minimising additional releases possibly related to recovery actions.

Identified measures and actions for long-term management

The measures and actions identified by the participants are divided into three subsections: procedures and guidelines, human and organisational resources, and technical resources.

Procedures and guidelines

The first procedural requirement is to ensure that accident management measures cover all modes of reactor operation including the shutdown states, and that all events that could cause damage to the fuel in a reactor core, in transport to storage, or stored in a spent fuel pool are included in the management measures also in the long term.

The foremost feature of the response by the participants was the recognition that a systematic identification of long-term accident management challenges and measures is missing, and the necessity for long-term management measures has not been assessed. Consequently, the following issues were identified:

- Most of the countries do not have any specific guidelines or procedures relevant to the long-term management of severe accidents.
- In most countries, such procedures are not foreseen to be implemented.
- In the cases in which measures are required or foreseen for the long-term management, the measures have not been properly analysed, and confirmed by verification, evaluation, and independent review.
- In general, factors and criteria that should be considered for the design and implementation of measures and actions applicable to long-term accident management have not been reviewed.
- The effect of the present SAMGs on the long-term management has not been evaluated.
- The transition from the accident management activities to accident recovery is not always defined. Some generic SAMGs include specific criteria to define a controlled stable state after a severe accident and to leave the SAMGs. These criteria are usually core exit temperature, hydrogen content and pressure in containment, radiation level on plant area and in SFP, and the water level in SFP.

Even though there are not many requirements for LTM, recommendations were given for the planning of LTM by the participants:

- LTM should be done in phases progressing from decontamination to defuelling and then decommissioning.
- Each phase needs detailed assessment of the plant damaged state. Data can be collected in consecutive steps depending on accessibility which will impact planning of recovery actions.
- It is recommended to do periodic evaluation of the plant damaged state using monitored data and numerical simulations of the accident scenario to support LTM planning.
- Importance of visual observation means was highlighted.
- Multi-unit arrangement of operated plants should be taken into consideration for long-term post-accident management.
- Contamination monitoring protocols and locations during the recovery phase should be defined, and a national strategy regarding solutions for post-accident contamination should be prepared.
- Criteria for returning to the evacuated area, and criteria for return to normal from the emergency state should be established.

Human and organisational resources

A long-term support to an NPP may be a real challenge for plant operators and state organisations, especially when there are aggravating factors such as multi-unit accidents, and degraded regional infrastructure.

The following issues were mentioned by the participants to be considered regarding the human and organisational resources:

- The roles and responsibilities of the personnel responsible for accident management should be clearly identified and communicated.
- The personnel involved in managing an accident should have all the necessary information, procedures and human and material resources to carry out accident management actions.

- Technical support centres for severe accident management and emergency crisis management teams can be provided in addition to the human resources at the plant.
- Solidarity between nuclear sites can provide additional personnel to the affected site.
- External contractors may be employed for specific tasks, e.g. mobile radiological measuring units, medical units, and additional fire men.
- Training of the personnel for severe accident management should be conducted regularly, and could also include elements of long-term management, such as the use of mobile equipment.

External conditions, e.g. extreme heat, cold, high radiation level, pose an additional challenge to the personnel at the affected site, and should be properly accounted for.

Equipment, infrastructure and instrumentation

After the degraded core becomes stabilised, the long-term management of the accident is expected to be carried out using the existing long-term plant monitoring capability, restored equipment, infrastructure and instrumentation, all the available and applicable procedures and guidelines, and available human and organisational resources. Measures required for the long-term accident management functions and goals are generally expected to exist in NPPs. These measures include plans, procedures and guidelines, equipment and infrastructure, plant monitoring instrumentation, and human and materiel resources.

All safety systems and components of the plant maybe useful for LTM and should be maintained operable as much as possible. Accessibility of the plant should be ensured. Measures and systems to monitor, control and maintain safety critical functions (sub-criticality, degraded fuel cooling, radioactive release limitation) on the long term should be provided as well as their support equipment.

Most countries mention the possibility of using specific equipment and measures developed for emergency phases also in LTM either on-site or off-site, e.g. central protected storage facilities for mobile equipment. Equipment such as large fire water pumps to supply cooling water to more than one unit or SFP and diesel generators to supply power to main plant safety systems are envisaged or already implemented in most countries.

Several systems of special importance to the long-term management were mentioned by participants:

- alternative water sources and injection capabilities, including several connections for them, e.g. reactor, steam generators, containment spray, and containment sumps;
- a dedicated system to reflood the corium and evacuate the residual heat from the containment;
- a core catcher, sacrificial material, or other system to arrest the concrete erosion by MCCI;
- a filtered containment venting system for containment pressure reduction and methods for long-term passive cooling of the containment;
- passive autocatalytic recombiners for hydrogen control;
- dedicated sampling systems for containment gas phase (dose rate, H_2 and O_2 concentrations) and sump coolant, and systems that can tackle high activity inventory samples;
- provisions for safe handling of samples and capability of analysis of high activity samples;
- on-site monitoring of activity release and effluent sampling;
- the storage for highly contaminated liquid and solid wastes;
- proper radiation protection measures, staffing and technical support centres;

- infrastructure needed to support at least three shifts on-site;
- accessibility of the site for emergency material and deliveries from outside.

Some more advanced technologies which were mentioned and considered to be useful for the LTM were given as:

- water treatment technology including decontamination and potentially desalination;
- remote control monitoring and investigation using robots;
- acoustic monitoring for passive detection of boiling behaviour;
- muon tomography for detection of core debris location.

Approaches, methods and tools for long-term management and actions assessment

In general, no established and well-thought approach to deal with LTM issues has been developed. The following tools were identified by the participants to be useful in the LTM:

- The existing integral severe accident codes, such as Accident Source Term Evaluation Code (ASTEC), MELCOR, and MAAP, can be used to assess the status of the affected plant, the possible damage to RPV and PCV, degraded fuel characteristics and location, as well as to support preparation of the fuel recovery actions. The analysis results should be compared with the available plant data regularly.
- Dedicated codes for certain phenomena, e.g. core-concrete interaction, may provide a more detailed assessment of the phenomenon under inspection than what is possible with the integral codes.
- A solid level 2 probabilistic safety assessment associated with detailed SAMG procedures can provide helpful guidance and understanding appropriate for long-term management, for instance, the identification of the most probable failure mode of RPV could provide valuable insights regarding the time windows available to the operators to flood the containment.
- Instrumentation and measurements may be applied to follow the evolution of the accident on the site.
- Monitoring radioactivity on-site and in the vicinity of the plant is important in the long-term management phase to detect possible leakage of radioactivity as a result of the core degradation or as a consequence of the decontamination activities.

Some challenges were identified in application of the existing tools for LTM:

- No tools or methods exist which have been developed and validated for analysis of long-term behaviour of a damaged nuclear power plant.
- The severe accident codes cannot predict the damage state of the plant in sufficient detail:
  - uncertainties in the prediction of the status of the damaged RPV and PCV, degraded fuel characteristics, and location for all conceivable accident scenarios;
  - uncertainties in the long-term accident phases such as MCCI termination and delayed releases of fission products;
  - fission product and actinide distribution in the plant cannot be predicted in sufficient detail;
  - uncertainties in the location of possible air or liquid leakages of radioactivity.
- Absence of data and models to assess risks that are specific for the LTM phases, e.g. safety system failure risks due to harsh conditions from the accident transient and from the long-term operation, risks related to degraded fuel leaching, combustion risks due to effects of corrosion reactions, potential iodine release from liquids or filtered containment venting systems, risks related to fuel recovery actions and water waste management.

- Application of probabilistic safety assessment to all events and all modes of operation is incomplete.
- Probabilistic safety assessment considered as not necessarily relevant to support long-term management and actions (LTMA) as the focus shifts from plant operation to the safe performance of recovery and waste management actions.

It is recognised that risk-informed approaches should be used to select the safest approach for LTMA incorporating observations and analyses results enlightening the plant damaged state. The plant damaged state will be changing with time with the progress of recovery actions. The uncertainty in the plant status decreases with time as more data from analysis and observations become available to assess the accident evolution. The evolution has to be considered for LTMA planning.

Chapter 3. **Approach to long-term post-accident management and actions**

The main goal of the present chapter is to describe fundamental aspects of long-term management (LTM) of a severe accident and to provide useful tools to address it. The definition and scope of LTM, as defined by the group, is given in Section 3.1. The main long-term controlled state functions as well as their monitoring for a safe LTM are stated in Section 3.2.

Then, an approach is described to identify challenges, issues and risks for LTM, covering a larger diversity of accident scenarios (reactor or spent fuel pool (SFP) severe accidents) in order not to limit the scope only to Three Mile Island unit 2 (TMI-2), Chernobyl and Fukushima Daiichi accidents scenarios.

Section 3.3 provides a generic classification of accident scenarios as a function of accident progression events (e.g. reactor pressure vessel (RPV) and/or containment failed or intact) and level and extent of radiological consequences in the plant which should largely influence the LTM scope. LTM challenges and issues are then identified for each generic scenario.

Section 3.4 presents a structured approach aimed at identifying and categorising main issues and risks for LTM. Within this approach developed to guide LTM, a risk-informed, plant-specific classification method of the events, alternate to the one discussed in Section 3.3, is used.

Finally, Section 3.5 presents an action identification and ranking table (AIRT) exercise that was conducted to identify challenges, open issues and knowledge and technological gaps related to main LTM actions: maintaining a coolable configuration and managing confinement, water waste, solid waste and effluent management, site clean-up and decontamination, defuelling of damaged reactors, damaged fuel/fuel debris and radioactive waste disposal, and spent fuel removal.

It is worth noting that the plant damaged state (PDS) classification (Section 3.3) and AIRT exercises (Section 3.5) were performed to support the formulation of generic recommendations (Chapter 5).

3.1. **Definition and scope**

In this report, long-term management[1] of severe accidents refers to *accident management actions* implemented after a plant has reached a stabilised and controlled state following a degraded core accident and up to and including fuel and debris retrieval from the damaged plant, temporary on-site storage of the fuel and debris, and eventual transportation to off-site permanent storage, and removal of spent fuel from the pool and transportation to off-site permanent storage. The timing of a plant reaching a stabilised and controlled state depends on, among other things, the accident initiating event, any and all prevention, mitigation and emergency response measures taken to reach a stabilised controlled state and the resulting plant damaged state and operating conditions.

1. The definition of "long-term management" in the context of this project is not to be interpreted as the so-called long-term management action traditionally considered by the industry as part of the severe-accident management. The latter definition still relates to bringing a plant to a controlled state following a degraded core accident, and involves, besides use or restoration of on-site equipment, implementation of mitigation measures employing mobile equipment, among other things, brought from off-site locations.

Preparation of some long-term management actions may be initiated before reaching a stabilised controlled state. However, in a severe accident situation, the priority should logically go to actions to terminate the emergency response phase and to stabilise the plant.

A plant is considered in a stabilised and controlled state when i) all parts of the degraded core, either still in place and/or relocated in-vessel and/or ex-vessel, are coolable and subcritical, and any stored spent fuel is also coolable and subcritical; ii) dispositions have been taken to limit as far as possible any further radioactive products dispersion and release to the environment; iii) there is no apparent nor urgent risk of combustible gas explosions.

The degraded core, if retained in-vessel, is considered to have reached a coolable configuration when there is no further hydrogen production from water-metal (clad and structural materials) interaction; the release of fission products from the degraded core is exceedingly small; there is no risk of corium (molten composition of core materials, structural materials, and other control materials) rupturing the vessel because of thermal attack (though other risks, e.g. loss of integrity because of fragility, can keep being developed further); and the degraded core remains subcritical. Similarly, the degraded core, if ex-vessel, is considered to have reached a coolable configuration when there is no further non-condensable gas generation either from metal oxidation or from molten core-concrete interaction; the release of fission products from core-concrete interactions is exceedingly small; the ex-vessel core debris is retained in the containment or, in the event of containment breach, environmental release of fission products is exceedingly small; and finally, the debris remains subcritical. The spent fuel inventory is considered in a coolable configuration if all the spent fuel rods, degraded or not, are confined in the pool without the risk of a runaway oxidation reaction; there is no significant production of hydrogen; release of fission products from degraded spent fuel is small; and spent fuel remains subcritical.

Accident management actions for long-term management of severe accidents comprise the actions taken to: i) ensure the plant remains in a stabilised controlled state as defined above till it reaches a long-term safe stable state; ii) prevent and mitigate unintended on-site and off-site consequences such as uncontrolled releases that could result from actions to maintain the plant in a stabilised controlled state as well as recovery actions; and iii) prepare and complete fuel and debris retrieval ensuring that there are no unintended consequences during the retrieval process and transfer of the same as well as spent fuel to temporary on-site and permanent off-site storage facilities.

Generic long-term on-site accident management actions include: i) maintaining and monitoring a controlled state; ii) cleaning and decontamination of work environment; iii) clean-up of liquid and solid wastes; iv) on-site processing and temporary storage of wastes; and v) fuel and debris retrieval, and transfer to temporary on-site storage facilities, and eventual transfer to permanent off-site facilities; and vi) radiological protection of plant personnel and emergency management. Likewise, generic off-site actions, not discussed in this report, include: i) monitoring and controlling effluent discharge; ii) monitoring environmental contamination and airborne activities; iii) decontamination of land mass and water bodies; iv) radiological protection of population; v) preparation for rehabilitation and food-chain recovery; and vi) preparation for re-establishing commerce and other normal civic functions.

The on-site and off-site actions related to radiological protection and public health are consistent with those defined in various International Commission on Radiological Protection (ICRP) documents as well as various Committee for Radiological Protection and Public Health (CRPPH) documents. The long-term accident management phase defined in the current report maps to the recovery phase in the ICRP documents. It is characterised by planned or existing exposure situations and recovery actions in areas contaminated during early and intermediate phases of an accident. Note the topic of radiological protection of general population is not addressed further within the scope of the current report as the subject is more suitable for the CRPPH mission. Also, the topics of rehabilitation, re-establishment of commerce, and other normal civic functions are not considered in the scope of this report.

It is important to emphasise that the precise nature and scope of long-term accident management actions are heavily dictated by the entry conditions to long-term which, as already pointed out above, depend, among other factors, on PDSs. For example, the PDS at TMI-2 reactor (i.e. severely degraded core but no failure of reactor pressure vessel) charted the course of long-term accident management actions in a certain way. In contrast, the accidents at the Chernobyl unit 4 and Fukushima Daiichi units 1, 2 and 3 resulted in PDSs that are far more challenging than that at TMI-2 and consequently, charted a different course of specific long-term management actions.

It is also important to emphasise that the PDS when entering long-term management depends on the preventive and mitigative actions taken during the short and intermediate terms of accident management to stabilise a plant. Thus, it is imperative to have a systematic analysis of entry PDSs. In Section 3.3, a pragmatic classification for entry plant damaged states is proposed. The classification is intended to be as generic as possible and highlights how such classification impacts recovery issues and long-term management actions. A second complementary approach, more detailed and systematic and based on the foundation of level 2 probabilistic safety assessment, is also provided in Section 3.4. This second approach not only complements the first one but also provides a sound technical basis to confirm the pragmatic classification.

3.2. Long-term controlled state functions and monitoring

LTM comprises a set of different goals which may be arranged under three categories:

- maintaining the damaged plant in a stabilised controlled state;
- decontamination and waste management;
- defuelling.

Each of these LTM top goals requires different technical and organisational approaches.

Goals aimed at maintaining a controlled state for the damaged plant are called functions.

This section is focused on the stabilised controlled state functions which are necessary to manage safely and successfully the full spectrum of long-term actions.

LTM functions aim at maintaining a controlled state at all units and SFPs of a site when a severe accident occurred in core and/or in SFP. Successful achievement of these functions relies on knowledge of the PDS plus continuous monitoring of critical parameters as an essential precondition.

These top-level functions, each of which is necessary yet not sufficient to reach a controlled state, concern:

- reactivity control;
- decay heat removal;
- mitigation of radioactive releases;
- prevention of combustible gas explosions.

Ensuring these functions during all LTM actions presents specific challenges partly linked to the difficulty to establish the plant damaged state (e.g. extent of fuel damage, barriers status). The severity of the accident (e.g. degraded fuel in and ex-vessel, eventually outside confinement, significant radioactive releases outside confinement) will also affect strongly the extent of countermeasures and actions and increase the complexity to comply with these functions. Finally, depending on the focus of the LTM actions (e.g. maintaining a controlled state in the damaged plant, decontaminating, defuelling, waste processing), complying with the functions will require different technical and organisational approaches.

A brief introduction to these functions is provided in Section 3.2, where the fundamental means to accomplish them, together with highlighting of critical aspects, are presented. Section 3.2 also gives an overview on monitoring of parameters needed to meet these functions.

Long-term management functions to maintain a controlled state

Reactivity control

Assessing risks of criticality in various configurations (in-vessel, ex-vessel, in SFP, during fuel retrieval, waste processing) is of special importance to implement, if necessary, proper countermeasures (e.g. use of borated water in some cases, use of absorber materials in others). Fuel type should be considered in the risk evaluation (enriched fuel, mixed oxide fuel and, eventually, new fuels [ATF]). Some configurations may present more risks than others, e.g.: in-vessel when control rod or plates degrade and relocate, in-vessel when fuel is fragmented forming porous debris beds, ex-vessel after long time when solidified corium has fragmented under radiation-induced damages.

Decay heat removal

Damaged fuel/fuel debris cooling on the LT may be provided by active or passive cooling using liquid or air coolant, and by means of a single or two separated coolant circuits.

Several support systems to accomplish this function are needed, namely a reliable and accessible – even under the presence of certain radiation levels – heat transfer mechanical and control equipment and ultimate heat sink.

Mitigation of radioactive releases

Limitation of gaseous and liquid radioactive releases entails a wide spectrum of sub-functions mainly dealing with the preservation of defence in depth barriers:

- Prevention of further RPV deterioration. If in-vessel melt retention (IVMR) succeeded – whether through in-vessel or ex-vessel flooding-, the remaining strength of the RPV after the accident, if it had been weakened by fuel relocation to the lower plenum, may not be known implying risk of failure at any time during LT. The mechanical resistance of the steam generator tubes, hot legs and main steam lines might also constitute an issue since high pressure and temperature differences during the course of the severe accident (by flowing of overheated gases) may have significantly weaken high sensitive components of the reactor coolant system (RCS).
- Containment pressure control (avoiding under and overpressure).
- Identification of release paths (including that after molten corium-concrete interaction [MCCI]).

Prevention of combustible gas explosions

Even if decay heat removal is achieved and radioactive releases are kept below certain threshold, the existence of potential flammable clouds whether inside containment, SFP building (if applicable) or any other building, should be identified and duly measures taken to avoid risk of explosion.

Monitoring quantities needed for LTM controlled state functions

Monitoring, control bands and trends of the safety important parameters should provide information about the state of key functions described above, together with the characterisation of the existing and evolving PDS.

The facility operator should, as far as achievable, monitor and control the following parameters (either through installed or portable/movable equipment and instrumentation, availability depending on physical plant conditions) to ensure that the stabilising functions are met:

- dose maps and environmental conditions to allow manipulating critical equipment;
- RPV water level and temperature;

- RPV pressure;
- RPV/RCS water injection flow rate;
- for pressurised water reactor (PWRs), steam generator pressure and water level (if RCS integrity preserved);
- containment water injection flow rate and flooded level;
- containment temperature (in dry well and wet well for boiling water reactors [BWRs]);
- containment pressure;
- containment combustible gas concentrations (hydrogen, carbon monoxide, oxygen);
- containment ^{135}Xe gas concentration (detection of criticality events);
- containment radiation level;
- for PWRs, auxiliary building temperature and radiation level;
- for PWRs, combustible gas concentrations in buildings connected to the containment;
- for BWRs, secondary containment temperatures, radiation levels, water levels, integrity, hydrogen concentrations;
- spent fuel pool water level;
- spent fuel pool temperature;
- water chemistry of reactor and SFP;
- radioactivity in waters around the facility.

3.3. Possible accident/plant damaged states classification for long-term management

In this section, the objective is to propose a classification that is as generic as possible for nuclear power plant (NPP) damaged states[2] in terms of their different impact on LTM and recovery issues and actions. It should serve to identify main LTM and recovery technical issues and actions as a function of main type of stabilised plant damaged states following a severe accident.

The goal is not to discuss here systematic and methodological step approaches that could provide additional information but would probably require some development since they were not applied to LTM up to this date. Such approaches are discussed in some detail in Section 3.4.

Further, human, organisation and relations to the public aspects are not discussed here even if their importance has been recognised.

Also, it is acknowledged that the proposed classification uses a very simplified approach for the building of operational plans for LTM and recovery actions. The establishment of operational plans for LTM and recovery actions would have to consider in much more detail:

- plant and site specificities, availability of in-site and off-site resources (installed and portable systems, equipment and instrumentation) and capability to install them;
- possible aggravating factors (e.g. strongly damaged infrastructures, multi-units accidents, etc.);
- progress with time in the diagnosis of the situation knowing that it may be highly challenging depending on the accident severity and available investigation tools to know about the degraded situation (e.g. damaged fuel distribution in the vessel and the containment, containment leak paths);

2. Plant damaged states discussed here are different from traditional PDS classification that is used in level 2 probabilistic safety assessment.

- the evolution of the situation with time (e.g. advancement of recovery actions, status of mitigating systems, possible cliff-edge effects on structures and systems, change in degraded fuel configuration, etc.);
- identification and prioritisation of risks challenging the damaged installation safety;
- the organisation of various actors of LTM.

The TMI-2, Chernobyl and Fukushima Daiichi accidents experience (Section 2.1) should help providing some general recommendations related to LTM as discussed further in Chapter 5.

The classification considers as first main criteria the extent of radioactive releases covering situations without and with containment failure and as second main criteria the type of accident (with or without RPV failure, with or without containment failure considering main modes of containment failures based on the Rasmussen classification [Rasmussen et al., 1975], SFP accidents). A summary of main discussed issues is provided in Table 3.1.

Situations with very limited release[3] to the environment during the emergency phase

Plant damaged state 1 (PDS-1): severe accident with intact RPV and containment

The accident results in-core degradation with partially degraded fuel assemblies; some debris formation and some fuel melting have occurred. Core cooling has been recovered before extended fuel melting and relocation could induce vessel failure. There is very small radioactivity release to the environment (less or comparable to limits for design base accidents). The plant state may be described by:

- active or passive cooling of degraded core in-vessel has been reached;
- partially degraded fuel assemblies present mechanical fragilities, presence of debris and solidified corium in the vessel;
- contaminated water circulate in RCS with forced or natural circulation and lixiviate exposed degraded fuel;
- H_2 combustion in the containment due to hydrogen production by cladding oxidation has been prevented (by dedicated severe accident management measures, e.g. passive autocatalytic recombiners or igniters, inerting);
- the containment building is contaminated. Some contaminated water may be present in circuits and tanks that were connected to the sump and/or RCS during the accident.

For LTM, despite preservation of RCS and containment integrity, such a situation is already rather challenging for ensuring main safety functions:

- maintaining degraded core cooling in-vessel and avoid any re-criticality on the LT considering the issue of damaged RPV resistance on the LT;
- avoid any further containment leakage and radioactive release on the LT;
- manage hydrogen in RCS, containment and circuits containing radioactive liquids on the LT (produced by radiolysis and corrosion reactions).

Generally, the systems used to provide the required functions such as cooling the corium and evacuating the residual power out of the containment are not qualified under severe accident (SA) conditions and are often degraded. It is therefore important to make these functions reliable by restoring the failing trains of these systems or by providing mobile means capable of replacing the means in service in the event of failure of the latter. This is valid for all the PDS situations discussed in the present section. We however consider that in the PDS-1 the stabilised state was established by the recovery of plant cooling systems (provided by original or enhanced design).

3. Limited to the level of design basis leakages.

Also, the challenge will be to determine what will be the best options for recovery actions in such a situation knowing that one will have to face recovery of partially degraded fuel assemblies with some fragility (which may be a challenge for handling) of fuel debris and of solidified corium depending on the extent of core degradation, recovery and treatment of highly contaminated waters from RCS and containment. These recovery actions will have to be conducted with a highly contaminated environment in the containment and will have to be conducted minimising dispersion of active material outside containment.

Learnings from the LTM of the TMI-2 accident are relevant for such situations (Section 2.1). At TMI-2, in addition to the highly contaminated containment, some contamination was also transferred to the fuel handling and auxiliary buildings. The approach used shows that first recovery actions dealt with removal or stabilisation of contamination to maintain access and operability of systems used to ensure main safety functions. Then, removal or stabilisation of contamination of large areas of the site's building started in the fuel handling building and the auxiliary building, then in the containment to prepare access to the RPV and defuelling operations. During this phase the main issues were:

- The decontamination of highly contaminated non-coated concrete floors and walls where contamination penetrated at variable depth. In particular, the extremely contaminated building interior required the removal of the concrete top layer.
- The treatment of large amounts of contaminated waters with in particular the elimination of Cs and Sr fission products.
- The monitoring of radiation levels and plant state.

These actions required the development of specific tools, equipment, facilities and technics.

The next recovery actions dealt with degraded fuel and debris retrieval from the reactor vessel. Planning, development of specific equipment and tools and training of operators for these actions could only be designed after sufficient knowledge of the degraded configuration in the vessel was obtained and risks and challenges associated to damaged fuel retrieval were identified and analysed. These operations were particularly challenging but were conducted safely and successfully at TMI-2.

The other challenging issue at TMI-2 was the radioactive waste management since the accident created unique radiological waste characteristics and some waste did not fit into established regulatory waste classification categories for transportation and disposal but this was also managed successfully.

Situations with limited or controlled releases to the site and environment during the emergency phase

PDS-2: SA with failed RPV and containment maintained without venting

The main difference with respect to the previous situation is that fuel assemblies degradation went further with large fuel melting. Degraded core cooling is recovered too late to avoid lower head vessel rupture and debris and corium have spread in the containment building. RCS depressurisation has been obtained either by RCS discharge or has resulted from the vessel rupture (without containment failure due, e.g. from energetic direct containment heating). The state is described by:

- active or passive cooling of the degraded core in-vessel and of the debris and corium ex-vessel have been reached through fixed or mobile equipment;

- some partially degraded fuel assemblies debris and solidified corium can remain in the core[4] (depending on e.g.: radial core power profile, the extent of core degradation, RPV breach location and extent and pressure at vessel rupture);
- debris and solidified corium are present in the containment, mainly in the reactor pit;
- the RPV failure under high pressure should be eliminated by severe accident management, still there is residual risk of a high pressure melt ejection scenario (which would lead to dispersed corium in the containment);
- contaminated water circulate through the breached vessel and lixiviate some exposed degraded fuel;
- contaminated water cool the corium and debris ex-vessel and lixiviate them;
- H_2 and CO (from oxidation and MCCI) combustion risk has been supressed by dedicated severe accident management measures;
- pressure build-up in the containment has been managed by heat evacuation means other than containment venting system;
- the containment building is contaminated, containment leakage might be increased but still sufficient to allow termination of uncontrolled release by reduction of the containment overpressure.

For LTM, despite preservation of containment, this state is more challenging than state 1 for:

- maintaining degraded fuel cooling in and ex-vessel on the LT with increasing risks of clogging or failure of cooling water loops and/or avoiding any re-criticality on the LT;
- avoiding any containment leakage, including that from any additional loop that would be required for ex-vessel corium cooling and containment heat evacuation;
- managing hydrogen production (by radiolysis and eventual corrosion reactions).

The degraded fuel and debris retrieval will be more challenging than in PDS-1 with larger dispersion in the containment and possible accumulation in areas where accessibility is very limited. Again, planning, development of specific equipment and tools and training of operators for these actions should only be designed after sufficient knowledge of the degraded configuration in the vessel and the containment is obtained and risks and challenges associated to damaged fuel retrieval are identified and analysed. These operations are expected to be particularly challenging as illustrated by the situations at the damaged units at Fukushima Daiichi.

The other challenging issue will be the radioactive waste management considering higher volumes of liquid and solid wastes than for PDS-1.

PDS-3: SA with failed RPV and limited or controlled releases (leakage of the containment or filtered venting)

The main difference with respect to the PDS-2 is that limited atmospheric radioactive releases have occurred and may affect the site accessibility. Such limited radioactive release may occur due to intended containment venting or leakage by containment overpressure. The state may be described as follows:

- active or passive cooling of the degraded core in-vessel and of the debris and corium ex-vessel have been reached through fixed or mobile equipment;

4. Some scenarios may result in a nearly complete transfer of degraded fuel to the containment notably when large fuel melting has occurred in the RPV and when the RPV failure favours melt transfer to the containment, see e.g. the situation at the Fukushima Daiichi unit 1. High pressure melt ejection scenarios are also expected to result in large corium transfer to the containment.

- some partially degraded fuel assemblies debris and solidified corium can remain in-vessel;
- debris and solidified corium are present in the containment;
- highly contaminated water circulate through the breached vessel with forced or natural circulation and lixiviate some exposed degraded fuel (large volumes may be required due to the breached vessel);
- highly contaminated water cool the corium and debris ex-vessel and lixiviate them;
- H_2 combustion risk due to hydrogen produced by cladding oxidation and by MCCI has been suppressed by dedicated severe accident management measures;
- pressure in the containment has been managed by heat evacuation means or containment venting system (CVS);
- the containment building is highly contaminated.

For LTM and recovery actions, such a state presents similar challenges like PDS-2 with added constraints resulting from the radioactive release from the containment.

Situations with significant uncontrolled releases to the environment during the emergency phase

We assume here that all preventive measures and actions provisioned to avoid such severe situations have failed.

PDS-4: SA with failed RPV and failed containment

The main difference with respect to the previous situations is that containment has either failed by over pressurisation (progressive pressure increase resulting from heat-up and condensable gas build-up or rapid pressure increase resulting from energetic events [direct containment heating], steam explosion, hydrogen combustion) or due to failure of containment isolation. The state is described by:

- active or passive cooling of the degraded core in-vessel and of the debris and corium ex-vessel have been reached through fixed or mobile equipment;
- some partially degraded fuel assemblies, debris and solidified corium can remain in-vessel;
- debris and solidified corium are present in the containment;
- contaminated water circulate through the breached vessel and lixiviate some exposed degraded fuel;
- contaminated water cool the corium and debris ex-vessel with forced or natural circulation and lixiviate some exposed degraded fuel;
- containment failed;
- the containment building and the attached buildings–are contaminated;
- the site is contaminated.

For LTM and recovery actions, such a situation presents much higher challenge than previous situations with added constraints resulting from the large on-site contamination and the failed containment. The emphasis of LTM could be placed first in decreasing dose levels on-site to restore site accessibility and recovering containment functions.

PDS-5: SA with failed RPV and failed containment by MCCI (for some designs)

The main difference with respect to the previous states is that containment has failed by MCCI and that significant radioactive releases occurred through the soil below the reactor building (for most designs). Ex-vessel core cooling, if applicable before the basemat melt-through, has been inefficient in avoiding concrete containment basemat melt-through by corium. The containment integrity is lost and cannot be recovered. The situation is described by:

- Cooling of the corium has occurred in the soil below the containment concrete basemat, its progression in the soil has stopped. For such a situation, it will be highly challenging to diagnose the situation and establish with confidence that the corium progression has stopped. Also, defining an entry to LTM is challenging as the degraded fuel in the soil can theoretically release by leaching radioactive material for an indefinite period of time and contaminate groundwaters. Containing contaminated waters would be very challenging in such a situation.
- Debris and solidified corium are present in the soil below the containment basemat.
- Large volumes of highly contaminated waters used to attempt to cool the corium have been released through the soil after basemat failure and have contaminated groundwaters.
- The containment building and underlying soil are highly contaminated, the containment integrity is lost and cannot be easily recovered.
- When the containment fails at high internal pressure then the release to the atmosphere may occur for certain reactor designs.

For this type of situation, it is rather challenging to specify what could be the best strategies to cool the corium and reduce radioactive releases, notably liquid releases.

Also the challenge will be to determine what will be the best options for recovery actions in such a situation knowing that one will have to face soil and groundwater contamination, then will have to re-establish some sort of containment, then possibly recover corium in the soil and treat large volumes of highly contaminated soils and waters which are not easily accessible.

PDS-6: SA with containment bypass (e.g. steam generator tube rupture in PWR and liner melt-through in BWR Mark I and Mark II containments[5])

The main difference with respect to previous states is that the containment is bypassed and significant early atmospheric radioactive release may occur. Unmitigated scenario with containment bypass leads to large release, site contamination and limited site accessibility.

The state may be described by:

- active or passive cooling of degraded core has been reached;
- extend of core damage depends on time of cooling recovery;
- the leakage has been isolated eliminating the containment bypass or the internal pressure was decreased to terminate release;
- the site is contaminated.

For LTM, that type of situation is rather challenging for:

- maintaining degraded core cooling in-vessel on the LT, avoid any re-criticality[6];
- avoid any further leakage and radioactive release;
- manage hydrogen combustion risk on the release path (with severe accident management provisions for hydrogen mitigation bypassed).

5. BWRs liner melt-through could as well be classified in PDS-4.
6. For BWRs liner melt-through scenarios, re-criticality may not be an issue if corium is outside the containment.

PDS-7: SFP accident

The main difference with respect to the previous situations is that the accident is occurring in the SFP. Depending on the reactor design, the SFP is either in the containment or in a specific building outside the containment.

We consider a situation resulting in severe fuel assembly damage and significant radioactive releases in the environment. With containment, the situation could be less challenging with notably reduced contamination on-site but the containment building would be highly contaminated. The situation is described by:

- sufficient water inventory and level in the pool have been recovered;
- active or passive cooling of severely degraded assemblies have been reached;
- the pool contains severely damaged fuel assemblies and fuel debris in water;
- the pool contains highly contaminated water;
- H_2 combustion risk due to hydrogen production by cladding oxidation has been suppressed by severe accident management provisions in the containment (when pool is in the containment).

The situation could be highly challenging due to large on-site and SFP building contamination and severe fuel degradation. Another challenge will be radioactive releases and waste volumes minimisations. Eventually, degraded fuel ageing and leaching in SFP may become issues as cooling will have to be maintained for a long time before final defuelling.

For SFP outside containment the risk of severe fuel degradation should be practically eliminated by preventive measures. The same requirement applies to plant operational states with open containment and unavailable safety systems (e.g. during refuelling). It should be mentioned that SFP outside the containment may cause high radiation level and site accessibility problem just due to water level decrease in the pool.

Aggravating factors

In nearly all the situations described earlier, some aggravating factors may render the LTM and recovery actions more challenging. These aggravating factors are:

- unreliability of equipment, instrumentation and systems (permanent or portable) for LTM diagnosis and actions;
- necessity to use other cooling sources than permanent ones (other external or portable sources) with problems which may arise from the use of raw water during the emergency phase (e.g. necessary chlorine elimination from wastes) and of larger volumes, the use of added cooling loops, etc.;
- damaged infrastructures resulting from e.g. seism, flooding, explosions, structural collapses. Risk of structural collapse may pose additional LTM challenge (cf. Chernobyl accident);
- some site specificity such as hydrology (cf. Fukushima Daiichi with groundwaters flowing in damaged facilities);
- meteorology during the release phases which may concentrate aerosol deposition on-site (e.g. in the case of heavy precipitations), aggravating the situation from the on-site point of view;
- multi-unit accidents that may result from external hazards (cf. Fukushima Daiichi).

Summary and conclusion

Using a simplified plant damaged states classification, summarised in Table 3.1, functions and some issues and challenges associated to LTM have been identified and discussed. They are rather generic in nature for all considered entry plant damages states for LTM; identified issues and challenges generally being of increasing complexity with increasing extent and level of on-site contamination, with increasing damaged fuel dispersion in the facility or eventually in the environment and with increasing generated wastes volumes. SA prevention and mitigation measures aim at reducing releases and contamination and damaged fuel dispersion. Aggravating factors may however strongly affect the extent of LTM.

Table 3.1. **Summary table of simplified classification of SA plant damaged states and corresponding long-term management functions and challenges**

Plant damaged state	Long-term management functions and challenges				
	Maintain stabilised controlled state	Recovery and decontamination	Defuelling	Wastes management	Worker and environment protection
PDS-1 • Core partially degraded • RPV damaged but not failed • No containment bypass • Containment intact but heavily contaminated (gas and liquid releases from RCS) • Very limited fission product release outside containment (design basis leakages)	• Avoid re-criticality in-vessel (e.g. boron injection) • Maintain core coolability in-vessel (using restored RCS cooling) on the LT (issue: RPV LT resistance) • Maintain the containment • Manage hydrogen combustion risk on the LT (production by radiolysis and corrosion) in RCS, containment and in all waste management processes	• Remove contamination from buildings (if containment leaked during the accident) and from the highly contaminated containment (issue: removal of incrusted contamination) • Recover highly contaminated waters for treatment • Minimise airborne and liquid releases during cleaning operations	• Establish as much as possible knowledge of fuel distribution and on various forms in-vessel • Design-specific tools and procedures for retrieval considering potential risks (re-criticality, exposure dose, dispersion, etc.) and challenges (accessibility, fragility of damaged fuel) • Defuel damaged reactor core	• Treat highly contaminated waters (particularly β-emitters elimination) and store processed waters (issue: tritium) • Handle and store fuel debris • Handle and store other wastes forms • Consider H_2 risk due to radiolysis in wastes • Minimise generated wastes volumes	• Monitor doses on-site and off-site (incl. in complex and contaminated environment) • Check efficiency of cleaning operations • Check adequacy and efficiency of wastes treatment and management
PDS-2 and PDS-3 • Core severely degraded, cooling recovered too late • RPV failed at low/high pressure • No containment bypass • Containment intact but heavily contaminated (releases from RCS plus debris and corium from RPV, possibly dispersed) • Moderate fission product release in environment (SA leakages, filtered venting)	• Avoid re-criticality in and ex-vessel • Maintain debris coolability on the LT in and ex-vessel after cooling is restored (issue: resistance of fixed/mobile cooling loop) • Minimise radioactive releases (gas, liquid) to the environment • Manage hydrogen combustion risk on the LT in containment and connected volumes and in all waste management processes	• Clean contaminated buildings and the highly contaminated containment (see defuelling) • Treat localised high contamination outside containment (e.g. leak paths, vent lines) • Recover highly contaminated waters for treatment • Minimise airborne and liquid releases (particularly groundwater contamination) during cleaning operations	• Idem to PDS-1 with higher complexity due to presence of corium/debris in containment (particular challenge if fuel is highly dispersed): • Establish as much as possible knowledge of fuel distribution and on various forms in and ex-vessel • Design-specific tools and procedures for retrieval considering potential risks and challenges • Retrieve remaining damaged fuel in reactor core and in containment	• Idem to PDS-1 with additional complexity due to presence of corium/debris in containment (particular challenge if fuel is highly dispersed) • Minimisation of waste volumes more challenging	• Idem to PDS-1 with additional complexity due to on-site contamination and presence of corium/debris in containment

Table 3.1. **Summary table of simplified classification of SA plant damaged states and corresponding long-term management functions and challenges** (cont'd)

Plant damaged state	Long-term management functions and challenges				
	Maintain stabilised controlled state	Recovery and decontamination	Defuelling	Wastes management	Worker and environment protection
PDS-4 and PDS-5 • Core severely degraded, cooling recovered too late • RPV failed at low/high pressure • Containment failed (open hatch, early failure, failure by MCCI) • Significant and uncontrolled fission product release in environment	• Idem to PDS-2 and PDS-3 with additional challenges linked to high on-site doses (delayed accessibility, cooling to be maintained much longer) and in addition: • Reduce, as much as feasible, containment leakages • Minimise, as much as feasible, radioactive releases (gas, liquid) to the environment • Manage hydrogen combustion risk on the LT in all volumes where H_2 is produced or transferred	• Idem to PDS-2 and PDS-3 with additional challenges linked to high on-site doses: • Treat highly contaminated zones on-site (particularly challenging for extended contamination and containment breached by MCCI) • Recover highly contaminated waters for treatment • Minimise, as much as feasible, airborne and liquid releases during cleaning operations	• Idem to PDS-2 and PDS-3 with an additional challenge if debris and corium are outside containment (containment failed by MCCI) • Ageing processes may have to be considered for material behaviour if defuelling cannot be performed in less than about 20 years after the accident	• Idem to PDS-2 and PDS-3 with higher complexity due to failed containment • Minimisation of waste volumes more challenging • Fuel leaching may become an issue if cooling has to be maintained for a long time	• Idem to PDS-2 and PDS-3 with additional challenges linked to failed containment and high on-site doses Monitor radioactivity in liquid and gas releases to the environment
PDS-6 • Core severely degraded, cooling recovered too late • RPV possibly damaged but not failed (if failed, see PDS-4) • Containment bypass • Significant and uncontrolled fission product release in environment through bypass	• Idem to PDS-1 with additional challenges linked to bypassed containment and high on-site doses (delayed accessibility) and in addition: • Isolate, as much as feasible, the bypass • Minimise, as much as feasible, radioactive releases to the environment • Manage hydrogen combustion risk on the LT in all volumes where H_2 is produced or transferred	• Treat highly contaminated zones on-site (particularly the bypass zone) • Recover highly contaminated waters for treatment (liquid releases by the bypass) • Minimise, as much as feasible, airborne and liquid releases during cleaning operations	• Idem PDS-1	• Idem to PDS-1 with additional challenges linked to bypassed containment and treatment of released contamination at bypass	• Idem to PDS-4 and PDS-5
PDS-7 • Fuel assembly severely degraded, cooling recovered too late • Full drain-down not avoided • Partial drain-down to level not adequate for fuel assembly cooling Significant fission product release to the environment	• Specific challenges related to high on-site contamination (accessibility, habitability) • Avoid SFP re-criticality Maintain coolability of spent fuel • Prevent hydrogen explosion in SFP and adjacent structures Mitigate environmental release • Specific challenges related to high on-site contamination (accessibility, habitability)	• Treatment of contaminated water • Decontaminate SFP structure and atmosphere • Minimise airborne and liquid releases during cleaning operations	• Establish knowledge of damaged assemblies and retrieval plan • Design-specific tools and procedures for retrieval considering potential risks and challenges (fragility of damaged fuel) • Clean-up and storage of damaged fuel • Handling and storage of other waste forms	• Idem PDS-1 with larger amount of damaged fuel	• Dose monitoring in SFP • Airborne radioactivity monitoring

3.4. Methods and tools for risks and issues ranking for long-term management

Introduction to a structured approach to risks ranking and issues identification

Risk-informed decisions play a significant role in many nuclear safety applications. Identification of the relative contribution of each piece of the system to plant safety response has been proven to be relevant to improve system performance by orienting efforts in safety arrangements.

Risk identification, whether quantitative or qualitative, is carried out through application of standard methods fitting well with the available information and nature of the system, that is to say, type and interrelation of the underlying components.

Contrary to plant normal operation conditions or even accident conditions before core damage, the long-term phase management of a severe accident addresses many different systems each of them entailing a set of different components or structures, among which RPV and containment, but also the areas of the site affected by spread radioactive contamination, solid and liquid waste storage, SFP and SFP building, etc.

Due (i) to the different nature of the systems involved in the LTM, it is not possible to assign one only risk identification method as it suitability depends, as stressed above, on a fitting between the method and the nature of the system itself. In addition (ii), the entry conditions to the long-term phase range among a wide range of possible scenarios.

As a consequence, a comprehensive approach to risk ranking requires as a preliminary step the identification of the different components involved in the performance of each different system. Only once such issue identification has been carried out, identification of the available and most suitable risk ranking tools to each of the components of each different system might be performed.

This method has been here addressed as structured approach to risk ranking and issues identification[7].

After briefly detailing the scope and some specific notes on the contents of the proposed method, the tasks to be followed are afterwards described, among which the need for scenario categorisation. The main section presents a comprehensive analysis of the main issues to be addressed under the LTM as the preliminary, fundamental step for risk method identification. The conducted structuring and arrangement of the different LTM goals and derived tasks is not mandatory in its details and just aims at providing an example of how to perform issue identification as a precondition for risk method assignation. The section is concluded in by linking suitable, available methods to each issue identified as gathered in Table 3.2.

The appropriateness of such structured approach for LTM has been recently confirmed by the recent efforts currently undertaken for LTM of Fukushima Daiichi on following up, monitoring and reducing risks as briefly described in Section 2.1. The method developed goes very much in line and shares many aspects in common with the methodology presented hereafter in this section.

Several preliminary remarks to clarify key aspects in driving identification of risk and issues ranking methods follow:

- Risks/issues and related ranking/identification methods depend on the scenario characterisation, i.e. risks/issues are accident sequence and configuration-alignment dependent.
- Therefore, a PDS categorisation process should be applied before identifying risks/ issues and related ranks/methods.

7. Rather than at a system level, the LTM will be structured upon different LTM goals, where each goal comprises a set of different, independent activities therefore constituting an independent system.

- There is a traced-back relationship linking identification methods with risks and issues, and risks and issues with PDS categorisation (Fukushima Daiichi challenges largely differ from those of TMI-2).
- Even if different analysers can handle different own specific definitions of risk, all of them encompasses key elements of frequency and consequences, i.e. potential for things to change and the magnitude of consequences if they do change (IAEA, 2001).
- As for the consequence, a reference situation will therefore have to be first taken from which deviations can be measured and, hence, the risk.

Since the current Section only deals with issues identification methods and not issues identification itself, and such methods depend on the type of issue but not on the LTM PDS, the identification methods will only be linked with the issue categorisation process.

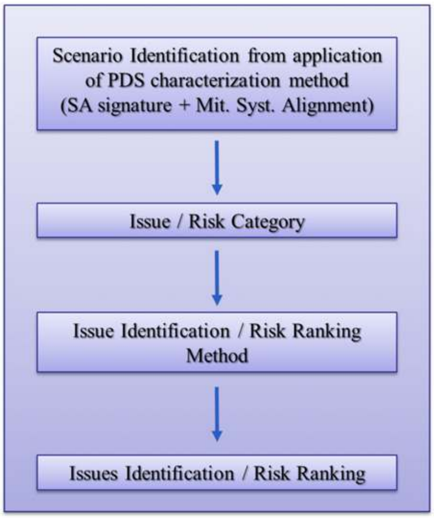
First subsections will describe the basis for a general structured, traceable, and practical identification of LTM risks and issues, starting from critical safety functions down to types of potentials for reference situations to change, where the latter will constitute the *joint* with suitable issues identification methods.

General approach to issues and risks identification

Long-term management of the severe accident covers a wide range of issues and risks along a wide range of scenarios. Identification of such scenario-dependent issues and risks relies also on the identification method itself so that each method's strengths and limitations can highlight and shadow issues and risks accordingly.

Therefore, in order to provide with practical suggestions for the full spectrum of long-term management on which method most suitably fits with issues identification and risk ranking, a generic approach will be applied as shown in Figure 3.1.

Figure 3.1. **Generic approach proposed for long-term management issues identification and risk ranking**



Mit. Syst. = mitigation systems.

Source: OECD/NEA (2018).

I. Scenario identification. Severe accidents can differ because of many aspects among which initiating event, performed operator actions or survivability of safety barriers. To handle such derived large number of scenarios, categorisation based on aspects leading to different long-term driving actions should be applied. Criteria underlying categorisation might be limited to those sensitive events and situations influencing long-term critical safety functions and derived actions as defined. Each scenario is considered as the single combination of a certain PDS in terms of severe accident phenomena having taken place and safety systems configuration to meet with long-term entry conditions. An example giving suggestions to conduct scenarios categorisation is discussed later in this section.

II. The second fundamental aspect addressed to suitably identify the most appropriate method for risks ranking and issues identification consists of performing a structured issue categorisation wherein scenario identification should be applied. Identification of issues and risks should be derived from suggested methods application for each category once adapted according to the scenario identified. An instance giving suggestions to conduct issue categorisation is discussed later in this section.

Scenario categorisation and analysis for conducting risk assessment and issues ranking

Addressing plant state categorisation contributes to the risk assessment process because of the following reasons:

- tasks addressing long-term issues are sequence-specific hence should be derived from particular PDS;
- risk ranking and issues identification methods and tools apply to specific accident conditions;
- each severe accident evolution features a different signature in terms of thermodynamic and radiological characterisation.

A detailed PDS categorisation upon (i) main aspects characterising severe accident signature and affecting long-term controlled state functions and actions, together with (ii) safety systems configuration needed to meet with those critical safety functions, helps revealing significant challenges to the long-term management.

Unlike design basis accidents and – to a lesser extent – the short-term phase of a severe accident, initial and boundary conditions in the accident long-term phase can widely vary:

- plant configuration: multiple mitigating system lineups meeting with long-term controlled state functions, in addition, with the possibility of using non-conventional and non-safety systems, on-site but also off-site equipment, fixed but also portable, etc.;
- location of fuel debris: depending on RPV failure mode, significant masses of fuel debris might be located outside the area below the RPV lower head;
- flooding levels both for the primary containment vessel (PCV) and attached buildings to PCV (including the reactor building for BWRs) that in the very long term might in the end damage mitigating equipment;
- environmental conditions (temperature, humidity, radiation) in rooms – or open spaces – hosting mitigating systems where human actions are foreseen;
- RPV/PCV physical boundaries, e.g. whether existing breaches communicate with attached buildings or the environment.

A widely used tool to arrange the entire spectrum of severe accident sequences, currently available in the majority of NPPs, is level 2 probabilistic risk assessment within which release category classification constitutes one of the mandatory steps of the methodology. Release categories are groups of core-damage accident sequences featuring similar source term release characterisation. Underlying criteria in containment event tree design indeed embeds aforementioned SA-LT categorisation criterion of making distinction upon the three accident management actions: (action 1) to reach a cooled configuration aimed at achieving a long-term controlled state, (action 2) to limit radioactive releases (with different source terms: location, type of failure and magnitude of the release) and (action 3) to switch to defuelling (given that from the accident categorisation standpoint, defuelling strategies depend on fuel debris distribution).

Appendix B provides a simplified example of SA-LT categorisation from level 2 probabilistic risk assessment release category figures of merit. It should be highlighted that this risk-informed approach to PDS classification differs from the one presented in Section 3.3.

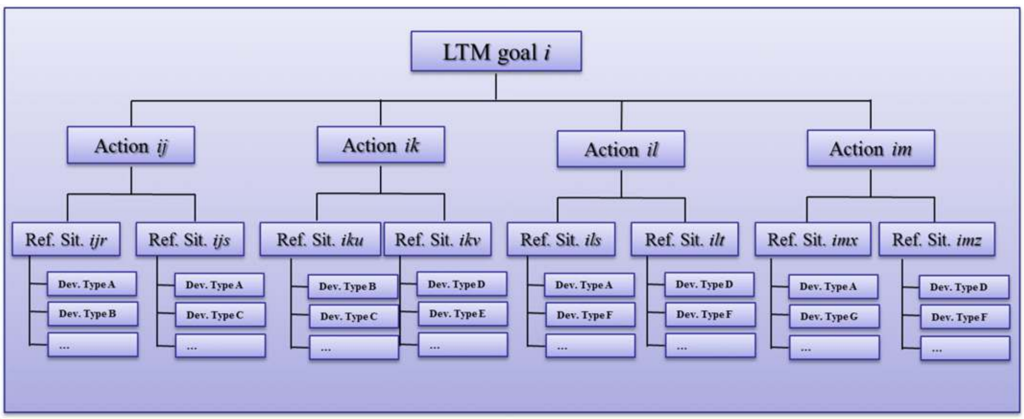
Issue categorisation and analysis for conducting risk assessment and issues ranking

Introduction and main components of a general structured layout approach

In order to generically address issues identification and risks ranking, a structured layout of long-term scenario category of issues must be undertaken. To help simplify the process and ensure the comprehensive nature of the resulting categorisation, such layout should be developed in a top-down approach stating i) LTM pursued goals, ii) means and actions in meeting with such goals, iii) reference situation/equipment needed and configuration to perform such means or actions, and iv) potential types of deviations in failing to maintain such reference situation. Suitable issues identification and risks ranking tools and methods will then be assigned according to the nature of the issue/risk and the type of deviation[8].

Figure 3.2 depicts the general structured layout for issues identification according to the aforementioned components in a top-down approach.

Figure 3.2. **General structured layout for issues identification**



Source: OECD/NEA (2018).

From a non-analytical view, such approach starts by distinguishing and reducing LTM objectives into categories of objectives which are independent of one another, i.e. each objective can be met without partly or fully correlatively meeting with the rest of the objectives. From an analytical view, each objective stands as a category as long as it constitutes an independent set of mandatory taken actions/measures, i.e. the carried out actions within each objective should be different of one another.

Stepping one level down, each action or measure is categorised as an independent means of accomplishing the overarching objective. At the action/measure-level, independency is established by featuring differences regarding some of the following criteria:

- addressed object which undergoes the action;
- addressed subject who performs the action;

8. Deviations stand for situations departing from reference situations where conditions are met for maintaining a controlled state. For instance, a situation where the hydrogen concentration in the containment increases above safety threshold limits is a deviation. The reference situations and related deviation scenarios within each LTM goal discussed in the following sections attempt to cover the most relevant aspects for LTM but the lists of items are not necessarily fully comprehensive.

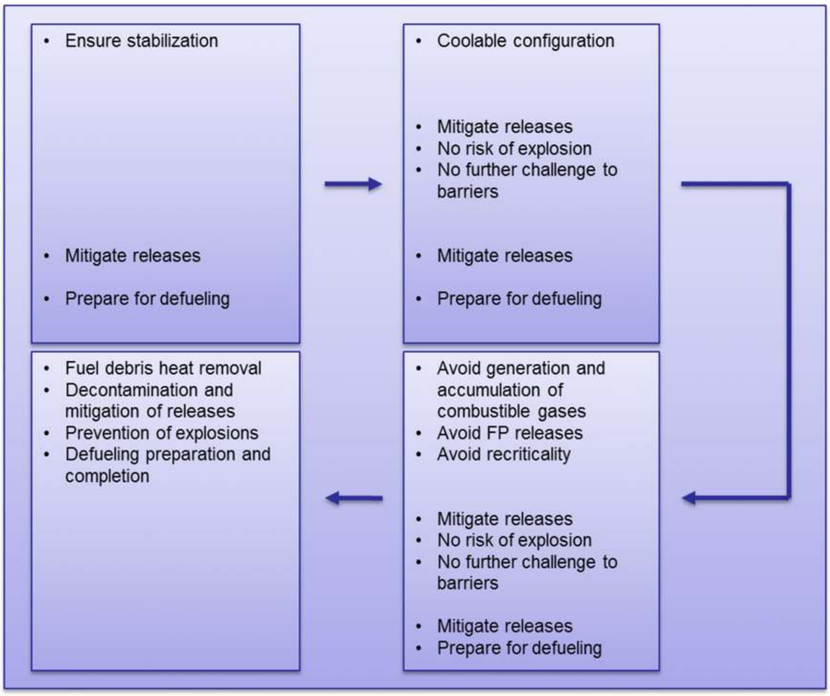
- area or component where the action takes place;
- constraints the action is subject to.

Each measure or action is in turn materialised into a specific set of supporting systems, configuration, scenarios, etc., each one constituting a so-called reference situation. Finally, the lowest level corresponds to the different causes making each action to depart from each reference situation.

Long-term management goals

According to the LTM definition (Section 3.1), accident management actions for long-term management are taken to reach the top goals of i) maintaining a stabilised controlled state; ii) prevent and mitigate uncontrolled releases; and iii) prepare and perform defuelling.[9] Each of these three top goals comprises in turn a set of related measures or actions. Since meeting with several of these goals depend on similar actions, LTM goals may be reclassified as depicted in Figure 3.3.

Figure 3.3. **LTM goals for the risks and issues identification as derived from the LTM definition**



Source: OECD/NEA (2018).

Following the LTM definition, the first goal of ensuring a stabilised controlled state can be broken down into ensuring coolability, mitigating radioactive releases, avoiding explosions and preventing any further defence in depth barrier degradation regarding their state at the time of entering into LTM phase. In turn, coolability condition is broken down into avoiding generation and accumulation of combustible gases, fission product releases and re-criticality either in-vessel, ex-vessel and in the spent fuel pool.

9. The followed structure is arranged upon LTM goals as specified within the definition in Section 3.1. An alternative, more exhaustive and detailed structure might rely on the generic long-term on-site and off-site actions as indicated also in the definition.

The resultant simplifying transformations shown in Figure 3.3 lead to the following combined top goals:

- Heat removal: fission product releases and flammable gas generation are kept under control in case of effective cooling of the fuel debris no matter their location. Even if the fuel debris is cooled down effectively, further failure of RPV and containment could occur on the LT, potentially challenging fuel debris heat removal. Thus, we opted for including prevention of further challenges to barriers under this current function/goal[10].
- Decontamination and mitigation of radioactive releases: actions involved in meeting with this objective can substantially differ from the rest of long-term actions since cooling can be achieved and barriers can maintain their status and at the same time significant leakages can result in radioactive releases to the outside environment or attached buildings (provided containment leak tightness has been lost).
- Prevention of explosion: risk of combustion of gases generated during the emergency phase has been as far as achievable suppressed. Moreover, since combustible gases might be further generated on the LT even though at very low rates once effective fuel debris heat removal is achieved, devoted actions should be considered in the LT in this respect.
- Defuelling preparation and completion: actions looking at removing fuel debris from containment, RPV and SFP belong to a separate set of actions.

With respect to the sub-criticality top-level issue, it initially addresses a goal different than removing heat from the affected buildings so it should be classified accordingly. Sub-criticality is ensured through monitoring and backup system aimed at keeping the fission reactions as low as possible, for instance by rapidly injecting boric acid.

Nonetheless, since both actions share their reference situations hence related deviations with some of the actions subsumed into the other categorised goals, sub-criticality as a top objective will not be further analysed. As an instance dealing with the boric acid injection, the reference situation will be a system injecting boron solution into the compartment where fuel debris are located, e.g. RPV or reactor pedestal, which fully coincides with the *maintaining LTM entry conditions* reference situation within the *plant performance* action within the *corium heat removal* top goal. And the same is valid for the monitoring action which can be found under different goals.

Fuel debris heat removal

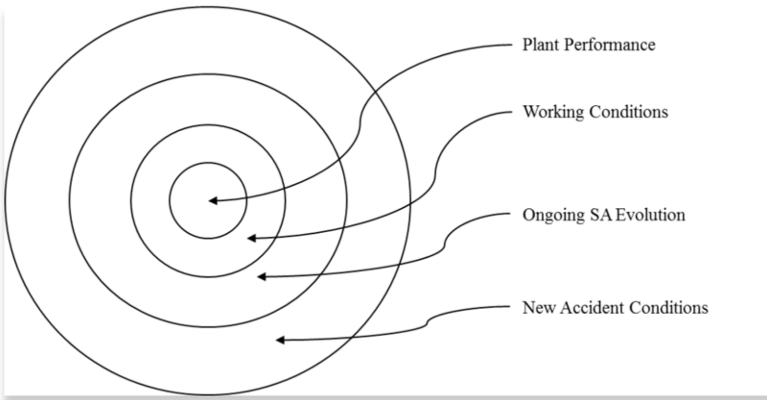
Fuel debris heat removal top-level function/goal comprises the following fundamental actions which can be figured out as different defence in depth levels as depicted in Figure 3.4:

- Plant performance: plant configuration is likely drastically modified so that systems in charge of accomplishing with the long-term controlled state functions should perform well along the LTM phase.
- Working conditions: actions performed on working mitigating systems, recovery actions, managing actions conducting the process, etc.
- Ongoing severe accident evolution: *synchronically*, entry conditions do not constitute the full list of relevant magnitudes to ensure keeping with a stabilised and controlled state through the entire long-term phase of the accident. *Diachronically*, entry-condition related magnitudes can evolve along time, whether in a slow degraded process or by abruptly coming across so-called cliff-edge effects so that critical quantities, so far kept below safety levels, would depart from a stabilised and controlled state.

10. Sub-criticality issues will be subsumed under the fuel debris heat removal category because related actions share reference situations and derived deviations.

- New accident conditions: measures against the onset of initiating events and evolution potentially leading to the loss of corium heat removal or any other long-term controlled state function capacity should be carried out moreover when plant configuration performance will likely lack of preventive safety measures other than working mitigating systems themselves[11].

Figure 3.4. **Fundamental actions affecting the long-term management goal of fuel debris heat removal**



Fuel debris heat removal: Plant performance

Source: OECD/NEA (2018).

Brief description of the issue

Mitigating systems meeting long-term controlled state functions are subject to fail intrinsically either because of internal component failures or so-called cliff-edge effects related to backup systems depletion. According to IAEA (2016), a cliff-edge effect "*in a nuclear power plant is an instance of severely abnormal plant behaviour caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in-plant conditions in response to a small variation in an input*".

Provisions for maintenance actions dealing with depletion are advisable in order to face specific long-term supplying type of problems since mission times usually managed in normal and accident conditions do not go far beyond 72-96 hours. Recovery actions dealing with internal failures should also be preaddressed and included as preventive measures to avoid dramatic equipment replacement gaps and minimise the corresponding corrective human actions.

Within the action of *plant performance*, ensuring that the working systems perform well, together with the recovery actions and the status of available guidelines to drive the related processes are analysed.

Reference situations and deviation types

Overall plant performance in meeting with corium heat removal can be analysed through the following reference situations each of which subject to specific potential deviations:

- Maintaining LTM entry conditions: Mitigating systems ensuring plant stabilisation should be kept under control.

11. The issue on new accident conditions actually affects also the "mitigation of releases" LTM goal insofar existing structural material, e.g. deposits of stored contaminated water, might be challenged because of the onset of an accident. For the sake of simplification, it would only be addressed within the "fuel debris heat removal" LTM goal.

- Mitigating systems alignment: Intrinsic deviations associated to mitigating system alignment meeting with fuel debris heat removal stem from internal failures that can either be mechanical, electrical or hydraulic of the safety system itself or supporting systems.
- Mitigating systems can also deviate from normal performance because of occurring conditions arising from slowly-increasing performance degradation going beyond equipment design technical specifications as a result of environmental and non-environmental burdens typical of unfolded severe accidents in the long term. Even though system suppliers might confirm that equipment performs well under certain test profile conditions through equipment qualification testing or survivability analysis, high uncertainty might arise when severe accident spans over long periods of time. For instance, sump blockage (highly reducing the available net positive suction head), heat transfer degradation by aerosol deposition (for instance, in the containment heat removal system cooling batteries), or mechanical fatigue (for instance, in the filtered venting system), might cause losing some of the safety critical functions such as containment overpressure or MCCI mitigation for some specific NPP designs. Ageing as a cause for conditions of the facilities getting moderately worse in the future also falls under this deviation type.
- Actions dealing with depletion: In dealing with depletion actions, as listed in *Safety of Nuclear Power Plants: Design, Specific Safety Requirements* (IAEA, 2016), the following examples illustrate the possible sources of internal cliff-edges:
  - depletion of batteries in blackout conditions;
  - depletion of water sources for primary circuit for loss-of-coolant accidents when recirculation is not possible;
  - depletion of water sources for secondary make-up in case of unavailability of closed circuit for heat removal through steam generators (this is applicable mainly for PWRs for transient events and events with secondary side line ruptures or secondary side valves failing to reclose).
- Certain short-term actions aimed at driving the plant to a stabilised and controlled state might negatively impact on the long-term phase (NEA, 2000). Corrosion problems derived from acid solutions present in the containment and coming from chemical additives, high dose rates in the auxiliary buildings as a consequence of containment hard venting, or thermal shocks because of high injection rates of cold water into the RPV, are instances of actions that should be analysed in advance to find a compromise solution between the short-term and long-term management.

- Recovery actions on equipment that can play a safety role in the long term.
  - Instrumentation capability to monitor main parameters evolution. Instrumentation availability deserves special attention since almost all actions and overarching functions/goals rely on instrumentation reading and feedback. In order to transit to LTM phase, a minimum set of instrumentation to check compliance with LTM entry conditions must be available, e.g. flammable gas concentration monitoring system, containment and RPV pressure and temperature instrumentation, etc. However, as noted above, the plant can depart from a stabilised and controlled state whether synchronically or diachronically. In order to track such safety challenge, instrumentation should be ensured beyond minimal requirements in meeting with LTM entry conditions.
  - Standard safety equipment: recovery actions focused on standard safety systems might contribute to give support to fuel debris heat removal. Allocated resources, planning of activities, people involved, etc., will dedicatedly address this topic. With respect to recovered standard safety equipment not being used in achieving safe stable state condition, special attention should duly take to ensure that no new challenging issues arise from such equipment as for instance damaged piping whose integrity has been lost or radioactive products spreading to areas hosting that

equipment. Therefore, deviations in this topic stem from worsening the current situation by putting in place standard safety equipment to reinforce fuel debris heat removal capabilities.

- Back-up guidelines supporting operator actions.
  - Existing procedures and guidelines, e.g. severe accident management guidelines (SAMGs). Once LTM entry conditions are met, current existing SAMGs, in the most favourable case, will only give instructions to follow-up generic actions. Therefore, challenges in this topic deal with gaps in managing and conducting corium heat removal related actions as specified above.

Fuel debris heat removal: Working conditions

Brief description of the issue

Human actions have traditionally played a very significant role as risk contributors to safety figures of merit, staying usually at the top of minimum cut sets contribution to core damage frequency in level 1 probabilistic risk assessment applications. Since SA-LT human actions largely differ from those characterising standard normal and abnormal operational conditions (i.e. normal operation and accidents with baseline plant equipment and configuration), it is highly advisable to adequately characterise human actions along the SA-LT phase from the perspective of so-called performance shaping factors (PSFs) and equivalent or related formulations.

Reaching a stabilised and controlled state as defined does not ensure that environmental conditions will long be kept under safe, sustained values – as already mentioned when distinguishing between synchronic and diachronic reasons to departing from a stabilised and controlled state. Variables dealing with meeting LTM entry conditions might not fully satisfy the entire set of variables affecting long-term human actions, e.g. radiation values, flooding levels, habitability (humidity and temperature) conditions, or existence of structural obstacles, or they might be satisfied only on a temporary basis.

It is worth noticing that the reason why human performance is analysed under the fuel debris heat removal issue is because the remaining actions, i.e. *plant performance, ongoing accident evolution* and *new accident conditions*, constitute the supporting foundations of the entire LTM building so that without keeping the reactor under a stabilised and controlled state no further action such as mitigations of releases, decontamination or defuelling might take place. Moreover, those actions are not only relevant from a safety standpoint but also because they are likely the most demanding ones in terms of human error precursor factors such as available time or stressful environments.

Reference situations and deviation types

Human performance is carried out by means of the following reference situations each of which subject to specific potential deviations:

- Environment conditions: This topic refers to environmental conditions affecting human action performance since harsh conditions in terms of high temperature, humidity or radiation will severely impact the success of the human action. Contribution of local actions in the SA-LT phase will likely take a far-reaching predominant role than standard normal and accident conditions so that new environments will have to be faced by field operators. Actions dealing with manipulating equipment, replacement, replenishment, or simply isolating/connecting areas by closing/opening safety gates will require a set of minimum working conditions to be able to perform that action:
  - Challenges to working conditions come from unbearable environments resulting from the ongoing accident itself or as a consequence of taken actions to reach and maintain LTM stabilised and controlled state.
  - Physical impediment to get access to specific areas and rooms and freely move throughout the plant as necessary.

- Human action performance: Deviations in human actions taken as a whole from expected performance in carrying out fundamental tasks under fuel debris heat removal deserve dedicated treatment:
  - Aside from environmental constraints, human actions can depart from their expected performance because of so-called categories of omission or commission, under which precursors such as lack of conducting guidance, ambiguous information, environmental stress, and many other factors are gathered upon.

Fuel debris heat removal, ongoing accident evolution (non-system-related cliff-edge effects)

Brief description of the issue

Loss of stabilised and controlled state can also be caused by cliff-edge effects leading to sudden modification of thermal-hydraulic conditions. Prolongation of previously under-control variables coming from short-term severe accident phase whose natural evolution encounters a cliff-edge effect can dramatically overturn its safe time trend.

Reference situations and deviation types

Potential issues coming from ongoing SA evolution affect the following reference situations each of which subject to specific potential deviations:

- Status of the environment in the primary/secondary system and attached buildings to containment.
  - Equipment performance can be challenged if the environment conditions go beyond design specifications. Such modifications might derive from the accident evolution in the long term and, alongside the containment, it can affect the auxiliary buildings hosting safety equipment. For instance, for PWR designs provided with passive autocatalytic recombiners (PARs) as the only available means to cope with flammable gases, if the containment runs out of oxygen, the PARs would not be able to keep the hydrogen and carbon monoxide concentrations under a certain limit. This C-E effect phenomenon might be relevant in case the non-flammability state of the containment still relied on PARs (even in the absence of MCCI, some hydrogen and carbon monoxide might continue being generated which, together with a continuous steam condensation, might in the end increase the flammable gases concentration). Progressively lower decay heats resulting in a negative pressure evolution might also challenge the containment pressure design limits and/or the flammability limits. Such pressure evolution might cause a decrease in the saturation temperature, and as a consequence, large steam condensation rates that in the end might result in flammable conditions.
- Prevention of further challenges to barriers.
  - Even if the fuel debris has been effectively cooled down, there might be a risk for undergoing leakage in the containment or the RPV. Contributors to this risk might come from unexpected pressure increases – for instance, related to continuously higher flooding levels pressurising a particular compartment –, or from a slow yet continuous increasing in the flooding level until reaching a specific mitigating system elevation.

Fuel debris heat removal, new accident conditions

Brief description of the issue

Contrary to standard normal and accident initial and boundary conditions, SA-LT phase is characterised by plant configurations where internal initiating events can only be derived from mitigating systems themselves, namely plant complexity degree has substantially been reduced since the number of plant systems is limited to only those ones in charge of achieving and maintaining the long-term controlled state functions (together with their support systems as well). Therefore, both in case of external and internal events, initiating events will always lead to direct loss of long-term controlled state functions (as there is no other plant working system performing functions different than safety functions).

Moreover, since the plant might have shifted to an almost entirely new configuration where most of previous existing systems to avoid initiating event propagation are no more available, preventive actions[12] should be accordingly undertaken to replace such systems.

External and internal events should be analysed and weaknesses that could lead to loss of long-term controlled state functions.

Reference situations and deviation types

Potential issues coming from new accidents affect the following reference situations each of which subject to specific potential deviations:

- External safety barriers:
  - Protection against new external hazards such as earthquake, flooding or extreme temperature should be analysed from the view of portable mitigating systems directly exposed to outside environments thereby to all type of extreme climate events without any kind of protection.
  - Consequence on the status of defence in depth barriers by external hazards should also be analysed.
- Internal safety barriers stemming from systems different than mitigating systems:
  - As a result of a drastic modification in the available systems, the plant might face an upcoming initiating event in a degraded situation. The lack in the long term of several standard safety measures operable in normal operation and limited accident conditions might result in the available structure (safety systems, but also including existing emergency procedures) falling short to correctly respond to the accident. Among them, the lack of the engineered safeguards actuation and sequence system, or of any redundancy should a system failure occur, might significantly contribute to the overall risk of reaching new unstable conditions. Therefore, the identification of the most vulnerable failures directly propagating to a sudden loss of a top-level function/goal turns to be a fundamental issue.

Mitigate radioactive releases

Mitigate radioactive releases top-level function/goal comprises the following fundamental actions:

- decrease radioactive gas emissions;
- decrease radioactive liquid effluent emissions;
- decontamination of highly contaminated floors and walls;
- radiological protection[13].

▪ Mitigate radioactive releases: Gas emissions

Brief description of the issue

LTM entry conditions ensure that generated radioactive gas source term has decreased to exceedingly small values. Nonetheless, radioactive gas could still be present in containment and attached buildings so that duly measures should be taken in an attempt of reducing their releases to outside environment. Moreover, if containment bypass leakage exists (beyond

12. Preventive regarding the new stabilised situation in the long term taken as a reference.
13. Disregarded here since not fully treated in the current report.

allowable leakage), or there is a breach in containment integrity or containment isolation system has failed, and the containment or RPV is under saturated conditions, radioactive gas will be continuously generated and released with steam.

Reference situations and deviation types

Treatment of radioactive gas releases is carried out through the following reference situations each of which presents particular deviation types:

- Identification of release locations: Radiation area monitors placed at each potential source term release point can help pinpointing radiological release locations. Deterministic and probabilistic analysis conducted in the plant might highlight those several equipment and containment interface locations more prone to fail and act as source of radioactive release. Actions should therefore duly be taken to check if those points have been challenged, e.g. whether they have undergone mechanical integrity failure, stuck-open positions, uncovered emissions (non-water-sealed or water-scrubbed releases), etc.:
  - Radiation monitor failure can hinder the process of identification.
  - In case of not having access to the monitor reading, it will also be difficult to establish a radiological map of the plant detailed enough to identify potential source term releases.
- Filtering of containment gases and radionuclides: Auxiliary buildings filtered ventilation system together with in-containment filters, e.g. those belonging to filtered containment venting (FCV), can have a strong impact in decreasing radiation values both inside the plant and released to the outside environment, furthermore increasing habitability:
  - slowly decreasing filters performance (reaching threshold performance limits);
  - lack of feedback through appropriate available instrumentation to check effectiveness of carried out actions, non-reliable readings.

Mitigate radioactive releases. Liquid effluent emissions

Brief description of the issue

Release of radioactive liquid effluent may occur under the special circumstances of containment flooding level reaching failed-to-close penetrations, because of containment bypass or containment failure.

Reference situations and deviation types

Control and minimisation of liquid effluent emissions are carried out through the following reference situations each of which presents particular deviation types:

- Separate treatment and storage of high and medium-low contaminated waters:
  - stable storage threatened by external hazards;
  - radioactive spills derived from logistics issues such as short-term temporary storage during the first instants after the accident, sufficient storage space or free area to install due connections with containment and treatment equipment;
  - failures in water treatment and disposal (including fission product control, monitor and sampling).
- Prevention of liquid effluents entering/exiting containment:
  - drawbacks from building additional barriers for liquid effluent confinement, both on-site (e.g. by isolating different contaminated areas of auxiliary buildings attached to containment) or off-site;

    drawbacks from temporary one-through liquid treatment and non-treatment water released to outside environment.

- Mitigate radioactive releases: Decontamination of highly contaminated floors and walls

Brief description of the issue

Decontamination efforts pursue maintaining access and operability of plant systems, in the first place, and removal of any remaining contamination for long-term storage in the second place.

Reference situations and deviation types

The following reference situations share common deviation types all of them related to issues stemming from monitoring and control of radiation spreading.

- identification and division of plant contaminated buildings into different levels of contamination;
- site buildings clean-up: sludge removal and surface decontamination;
- debris and sludge characterisation;
- soil and groundwater monitor and decontamination;

Prevention of risk of explosions

Prevention of risk of explosions as top-level function/goal comprises the following fundamental actions:

- monitoring;
- clearing;
- removal.

- Prevent risk of explosions: Monitoring

Brief description of the issue

One of the LTM entry conditions consists of keeping combustible gas generation under threshold limits either from metal oxidation or from molten core-concrete interaction. Therefore, in order to transit from short-term to long-term phase of the severe accident management, monitoring of combustible gases should be performed.

Reference situations and deviation types

Monitoring of combustible gases is carried out through the following reference situations each of which presents particular deviation types:

- Analysis, monitoring and filtering of containment combustible gases, especially in those locations more prone to hydrogen and carbon monoxide build-up:
  - loss of monitoring devices (including non-reliable readings and samplings);
  - inaccessibility issues to monitor readings.
- Environmental analysis of attached buildings to containment including reactor building, especially in those locations subject to hydrogen and carbon monoxide accumulation:
  - inaccessibility to rooms hosting safety equipment or other areas susceptible to explosion.

Prevent risk of explosions: Clearing

Brief description of the issue

Even if combustible gas generation has been drastically reduced, flammable clouds may still be present in containment and attached buildings, e.g. small confined compartments or even inside pipes, so that duly measures should be taken in an attempt of preventing risk of explosions in the long term.

Reference situations and deviation types

Clearing of combustible gases is carried out through the following reference situations each of which presents particular deviation types:

- Venting, whether filtered venting through a chimney or manual venting through manually opening of hatches located at top floors of attached buildings to containment, as well as accessible doors increasing natural circulation thereby flammable gas mixing and dilution with air:
  - inaccessibility due to radiation;
  - loss of automatic opening system;
  - slowly-increasing filters performance degradation (reaching threshold performance limits).
- Switch-on of heating, ventilation, and air conditioning system (or equivalent) to favour flammable gases dilution with air:
  - lack of supporting systems (power, actuation);
  - inaccessibility to locations from where the system must be activated.

Prevent risk of explosions: Removal

Brief description of the issue

Even if corium has been quenched, flammable gases may be continuously generated for instance through water radiolysis.

Reference situations and deviation types

Removal of combustible gases is carried out through the following reference situations each of which presents particular deviation types:

- recombiner device performance:
  - oxygen depletion would interrupt hydrogen and carbon monoxide oxidation;
  - recombiners may only face low flammable gases generation rates.
- injection of inert gases:
  - loss of injection of inert gases such as nitrogen may conduct to flammable gas escalation.
- igniters:
  - Loss of igniters supporting systems would prevent igniters performance. Moreover, after significant elapsed times along which igniters have not been working, thereby flammable gases have been building-up, turning on igniters might drive to challenging explosions.

- venting[14].

Defuelling preparation and completion

Contrary to the other three main areas of activities, no urgent issues should be immediately addressed related to the topic of preparation and completion of defuelling.

As top-level function/goal, it comprises the following fundamental actions:

- collect information from in-vessel/ex-vessel and SFP inspections and observations of the status, location and characterisation of fuel debris, including analysis performed with code simulations;
- training of defuelling operators and development of new tools and equipment;
- defuelling and disposal operations;
- safety requirements for defuelling.

Preparation for defuelling: Collection of information

Brief description of the issue

First issue to tackle with when planning defuelling activities consists of achieving thorough understanding on the extent of the events in terms of status, location and characterisation of degraded core, e.g. whether RPV failure occurred, corium relocation to other parts of the containment aside from the area placed right below the vessel, etc.

Reference situations and deviation types

Information gathering is carried out through the following reference situations each of which presents particular deviation types:

- available information about the accident follow-up in terms of status, location and characterisation (e.g. mass) of the degraded fuel:
  - ambiguity in operators/technical support centre in diagnosing the accident/ interpreting the available information;
  - absence of information on RPV/SFP state and fuel debris distribution and characteristics.
- inspection by means of dedicated devices such as visual cameras, muon techniques, etc.:
  - impossibility of introducing devices into containment/RPV/SFP;
  - deficient provided information.
- analysis with dedicated severe accident simulation codes:
  - discrepancy in the results when using different codes;
  - uncertainty in the results whether because of a deficient the state of the art or because of a lack of information.

Preparation for defuelling: Training

Brief description of the issue

In order to address training on defuelling and disposal, personnel should be trained in using dedicated new tools and equipment.

14. Already included within the previous clearing action.

Reference situations and deviation types

Training activities may include simulations in real mock-ups and development and handling of ad hoc tools, e.g. to visually inspect containment, RPV or SFP, or remotely load debris canisters.

No generic deviation type in this action deserving issues identification and risks ranking methods and tools application is foreseen.

▪ Preparation for defuelling: Defuelling operations

Brief description of the issue

The area of activities focused on defuelling operations puts an end to the long-term management of the severe accident. It includes a wide series of ad hoc activities performed by trained personnel and spanning along a large time frame that can last several months.

Reference situations and deviation types

Defuelling operations are carried out through the following reference situations each of which presents particular deviation types:

- Provide the plant with suitable temporary defuelling platforms:
  - No generic deviation type in this action deserving issues identification and risks ranking methods and tools application is foreseen.
- Collection and characterisation of debris samples:
  - worsening scenario by related handling operations;
  - extrapolation feasibility of sample characterisation.
- Collect corium as deposited during accident evolution distinguishing between debris containing degraded fuel and additional debris material such as cladding, structural and control materials:
  - Hazards involved in related operations such radioactive dust dispersion, high dose rates, modification of plant scenario resulting from containment flooding, radiation contamination, re-criticality, etc.
- Place all collected kind of high activity inside-containment debris into debris containers or canisters:
  - Deviations match those ones already indicated for collection of corium reference situation.
- Disposal of solid waste by-products used in the clean-up and treatment of effluents phase including filters, demineralisation devices, clothing, tools and equipment:
  - Deviations match those ones already indicated for collection of corium reference situation.
- Transport all collected kind of high activity inside-containment debris, together with aforementioned by-products, to an on-site safe, temporary storage location:
  - Deviations match those ones already indicated for collection of corium reference situation.
- Final transportation to off-site dry, safe storage:
  - Deviations match those ones already indicated for collection of corium reference situation.

- Different provisions for site modification dealing with following topics: containment isolation, full area decontamination, tracking of nuclear materials, establishment of a regular monitoring programme to ensure continued safety of the plant:
  - Deviations match those ones already indicated for collection of corium reference situation.

Preparation for defuelling. Safety requirements

Brief description of the issue

When developing new tools and equipment and planning defuelling operations, safety requirements should be clarified and shared among the relevant stakeholders. This is important not only to protect people, including public and workers, and the environment, but also to facilitate the defuelling through avoiding unnecessary rework for preparation based on unclarified requirements.

Reference situations and deviation types

These safety requirements should build on the reality of the facilities, e.g. availability of existing facilities and effectiveness of human actions as safety functions, and consequently may be different from those for the operational power reactors.

Deviation could come from too much conservatism due to lack of information. This may result in extended period of time and increased work for preparation, which in turn would extend time at risk and increase occupational exposure.

Issue identification and risk ranking methods

Once each LTM goal has been analysed by having referred to their actions and related reference situations, within which a set of main potential deviations have been anticipated, issues identification and risks ranking methods will be applied based on the nature of the identified deviations. In order to identify such methods, the main source of information will come from industry, and more specifically, from nuclear safety and risk applications.

Identification and ranking tools might be easily categorised in qualitative and quantitative methods. Since LTM has not long been one main focus of attention within nuclear safety, some of the suggested methods have not ever been applied to this particular field. Nonetheless, extrapolation might be carried out provided the method applies to the same category of treated events. As an instance, this is the case of fault tree analysis (Appendix C) addressing portable equipment rather than traditional equipment, or SA system codes in extending the calculations at long-term time frames typical from LTM, e.g. in the order of weeks.

The suggested methods are linked to each identified deviation and they are followed by either an “I” or “R” letter standing for methods which address issues identification or risk ranking.

For each goal/action/reference situation, possible issue identification and risk ranking methods are listed in Table 3.2.

Table 3.2. **Issue Identification and risk ranking methods applicable to long-term management and actions (LTMA)**

Goal/action/reference situation	Deviation	Issue identification/risk ranking methods
Corium heat removal		
Plant performance		
Maintaining LTM entry conditions	Deviations from intrinsic equipment failures	• Single-system fault tree analysis (See Appendix C) (I & R) • *Failure mode and effect analysis* (I & R) • SA system codes (I) • Extended fault tree analysis (See Appendix C) (I & R) • Engineering analysis to test the reliability of the equipment operating in extreme conditions during long-term periods (I)
	Slowly-increasing performance degradation	• Hazard and Operability Study (I & R) • SA system codes (I) • Phenomena identification ranking table (I & R) • Engineering analysis to test the reliability of the equipment operating in extreme conditions during long-term periods (I)
	Depletion-related issues	• *Failure mode and effect analysis* (I & R) • Extended fault tree analysis (I & R)
	Impact of short-term actions in the long term	• Research survey (I)
Recovery actions	Instrumentation capability	• Redundancy (I) • Analytical calculations (I)
	Standard safety equipment (including worsening consequences)	• Pipe and instrumentation and instrumentation and control drawings (I), systems performance procedures (I)
Back-up guidelines supporting operator actions	Once LTM entry conditions are met, current existing SAMGs, in the most favourable case, will only give instructions to follow-up generic actions	• Probabilistic risk assessment • SA system codes • (See Appendix IV) (I)
Working conditions		
Environment conditions	Unbearable environments resulting from the ongoing accident itself or as a consequence of taken actions to reach and maintain LTM safe stable state	• Human Reliability Analysis techniques (identification stage) (I & R) • Walkdown (I) • SA system codes (I)
	Physical impediment to access to specific areas and rooms or freely move throughout the site	• Walkdown (I)
Human action performance (holistically estimated)	Human actions can depart from their expected performance because of so-called categories of omission or commission, under which precursors such as lack of conducting guidance, ambiguous information, etc.	• HRA techniques (I & R) • Operating experience (I & R) • SA-LT drills (I)
Ongoing SA evolution		
Environment in primary/secondary system and attached buildings	Environmental characterisation on safety systems constraints and equipment survivability as a consequence of sharp modifications in accident evolution in rooms and buildings hosting safety equipment	• SA system code simulations (I) • Risk Oriented Accident Analysis Methodology (I & R)
Prevention of further challenges to defence in depth barriers	Containment and RPV leak tightness could be jeopardised even if corium is effectively cooled down	
New accident conditions		
External safety barriers	Protection against new external hazards and outside environmental threatens against directly exposed portable equipment	• Probabilistic risk assessment event trees dealing with external events (See Appendix II) (I & R) • Risk Oriented Accident Analysis Methodology (I & R)
Internal safety barriers	Internal failures not stemming from mitigating systems directly propagating to sudden loss of top-level function/goal	• *Failure mode and effect analysis* (I & R) • Fault tree analysis (I & R) • Risk Oriented Accident Analysis Methodology (I & R)

Table 3.2. **Issue Identification and risk ranking methods applicable to LTMA** (cont'd)

Goal/action/reference situation	Deviation	Issue identification/risk ranking methods
Mitigation of radioactive releases		
Gas emissions		
Identification of release locations	Radiation monitor failure	• Dedicated movable radiation monitor devices (I) • SA system code sequence results fed by available online accident evolution information (I & R) • Radiation modelling and tracking codes for auxiliary buildings to containment for indirect confirmation (I) • Risk Oriented Accident Analysis Methodology (I & R)
	Inaccessibility to monitor readings	
Filtering of containment gases and radionuclides	Slowly-increasing performance degradation (reaching threshold performance limits), both in containment and in attached buildings to containment where radioactive cloud has leaked into	• Hazard and Operability Study (I & R) • Phenomena identification ranking table (I & R) • Engineering Tests fed by Long-Term SA system code simulations (I) • Risk Oriented Accident Analysis Methodology (I & R)
	Lack of feedback from available instrumentation to check impact of carried out actions	• Indirect measures (e.g. pressure evolution in potential released points, radiation meter evolution on-site or off-site close to the release point, etc.) (I)
Liquid effluent emissions		
Treatment and storage of highly containment waters	Stable storage threatened by external hazards	• *Failure mode and effect analysis* applied to water treatment new equipment coupled with containment/auxiliary buildings (I & R)
	Logistic issues	• Preliminary design of water treatment and disposal facility (I) • Operating experience (I)
	Failures in water treatment (including fission product control, monitor and sampling)	• *Failure mode and effect analysis* (I & R) • Operating experience (I)
Minimisation/prevention of liquid effluents entering/existing containment	Drawbacks in building additional barriers for liquid effluents confinement	• Engineering and manufacturing analysis (I) • Operating experience (I)
	Drawbacks in temporary one-through liquid treatment or non-treatment water released to outside environment	• Preventive measures (I) • Operating experience (I)
Decontamination of highly contaminated floors and walls		
Identification and division of plant contaminated buildings into different levels of contamination	Wrong schedule of activities	• Dedicated movable radiation monitor devices (I) • SA system code sequence results fed by available online accident evolution information (I & R) • Radiation modelling and tracking codes for auxiliary buildings to containment and off-site area for indirect confirmation (I)
	Wrong diagnosis	
Site buildings clean-up	Inaccessibility due to environmental constraints in case of radiation monitor failure	
Debris and sludge characterisation		
Off-site soil and groundwater monitor and decontamination		
Prevention of risk of explosions		
Monitoring		
Analysis, monitoring and filtering of flammable gases	Loss of monitoring devices	• Redundancy (I) • Analytical calculations (I) • Indirect measures (I) • SA sequence simulations (I) • Risk Oriented Accident Analysis Methodology (I & R)
	Inaccessibility issues to monitor readings	
Environmental analysis of attached buildings to containment	Inaccessibility to rooms hosting safety equipment or other areas where explosion might be propagated	• Flammable gas clouds tracking and build-up in auxiliary buildings simulations resulting from bounding SA sequences to detect whether safety equipment hosting rooms might be affected by leaked flammable gas from containment. (I & R) • Risk Oriented Accident Analysis Methodology (I & R)

Table 3.2. **Issue Identification and risk ranking methods applicable to LTMA** (cont'd)

Goal/action/reference situation	Deviation	Issue identification/risk ranking methods
Clearing		
Venting	Slowly-increasing filters performance degradation	• Hazard and Operability Study (I & R) • Phenomena identification ranking table (I & R) • Engineering Tests fed by Long-Term SA system code simulations (I) • Risk Oriented Accident Analysis Methodology (I & R)
	Inaccessibility due to radiation	• Application and plant-specific analysis, e.g. to instruct personnel to promptly ventilate auxiliary buildings through operation of HVAC or opening of doors/hatches (I)
	Loss of automatic opening systems	
Heating, ventilation, and air conditioning (HVAC) system switch-on	Lack of supporting systems (power; actuation)	
	Inaccessibility to locations where systems can be activated	
Removal		
Recombiners	Oxygen depletion	• SA sequence simulations (I)
	Flammable gases generation peaks	
Injection of inert gases	Loss of injection of inert gases conducting to flammable gas escalation	• Engineering analysis providing countermeasures (e.g. redundancy) • Dedicated procedures instructing operators to follow suitable alternatives (I) • Fault tree analysis (I & R) • SA sequence simulations (I)
Igniters	Loss of supporting systems	
Venting	(already identified)	(already identified)
Preparation for defuelling		
Collect information		
Provided information during the accident follow-up	Ambiguity/disagreement in operators/technical support centre diagnosis	• Reaching a clear, unified picture by contrasting and solving contradictory pieces of information between all sources of information (available monitor readings – including radioactivity area measures), analysis of code results and visual inspections (I) • Use of alternative best-estimate stand-alone codes specific for ex-vessel corium relocation and deposition (I) • Operating experience (I & R)
	Absence of information on RPV/SFP state and fuel debris distribution and characteristics	
Inspection by means of technical devices	Impossibility of introducing devices into containment/RPV	
	Deficient provided information	
Analysis with dedicated ex-vessel phase simulation codes	Discrepancy in the results	• International code benchmark exercises (I) • Research and development efforts in uncertainty assessment, modelling and experimental analysis (I)
	Uncertainty in the results	
Training		
Training activities	N/A	• Operating experience (I & R) • Dedicated facilities (I & R) • Virtual reality devices (I)
Defuelling operations		
Suitable temporary defuelling platforms	N/A	N/A
Collection and characterisation of debris samples	Worsening scenario by related handling operations	• Human performance (I & R) • Operating experience (I & R)
	Extrapolation feasibility of sample characterisation	• Best-estimate stand-alone codes specific for debris characterisation (I) • Operating experience (I)
Collect deposited corium	Hazards derived from related operations (radioactive dust dispersion, high dose rates, worsening scenarios, etc.)	• Usual human performance tools (e.g. documentation evidence field walk-down track, pre-job brief, problem statement analysis causes testing solution, etc.) (I) • Operating experience (I)
Place all collected kind of high activity inside-containment debris into debris containers or canisters		• Human performance (I & R) • Operating experience (I) • Current existing dedicated guidelines (from affected plant and beyond) (I)
Disposal of solid waste by-products used in the clean-up and treatment of effluents phase		
Transport of high activity debris		
Final transportation to off-site dry safe storage		
Different provisions for site modification		

3.5. **Action identification and ranking table exercise**

Introduction, scope and objectives

The phenomena identification and ranking table (PIRT) is a structured elicitation process used to achieve consensus on subject matters of interest. In the context of Long-Term Management of Actions for a Severe Accident in a Nuclear Power Plant (LTMNPP), the process entails reaching consensus on commendable LTM practices based on lessons learnt from past nuclear power plant accidents. The management actions are informed by the state of phenomenological knowledge of severe accident progression and consequences in the long term as well as any gaps therein. The actions are also informed by guidance, procedures, and availability and reliability of prevention/mitigation systems. For brevity, the long-term management PIRT will henceforth be referred to as the AIRT.

The scope of the AIRT is focused on all necessary accident management actions to maintain a plant in a stable post-accident state and prepare for plant decommissioning. The decommissioning action itself is not in the scope. Specifically, the scope focuses on continued cooling of debris, managing confinement of radioactivity and managing risk from combustible gaseous products, water and solid wastes and effluent management, site clean-up and decontamination mitigating in-plant dust dispersion, defuelling, safe disposal of fuel debris and radioactive waste, and on-site emergency management. The last of these items covers hazard assessment, occupational health protection, countermeasures taken, and any emergency management and administrative actions that may arise during the recovery process. The topic of public health is not included in the scope even though being an important element of LTM. Other important elements of LTM, not covered in the AIRT and in consistency with the scope of the LTMNPP project, are regulatory oversight, public interactions, environmental remediation, rehabilitation, and economic considerations.

Acknowledging the wide types of actions addressed under the long-term management phase, a LTM-top-goal-driven approach is taken so that rather than placing all actions together, each of them will be binned according to the objective pursued by the action.

A total of six tables have been considered, the first five corresponding to different LTM top goals, whereas the sixth table addresses management cross-sectional issues thus affecting actions gathered under the other five tables:

1. maintaining coolable configuration and managing confinement;
2. water waste, solid waste, and effluent management;
3. site clean-up and decontamination;
4. defuelling of damaged reactors;
5. fuel debris and radioactive waste disposal, SFP fuel removal;
6. long-term management.

Given the aforementioned scope, the AIRT process consists of taking each major LTM goal and dissecting it into several actions (some inter-dependent) and determining what each action is with regard to scope; practical means of achieving the scope; and gaps in technology, procedure, and other factors that may pose a challenge in accomplishing the objectives. In parallel, each action is evaluated with regard to their role and the degree of success in accomplishing the overall particular LTM objective. In the final step, gaps in technology, procedure, and other factors are ranked and prioritised, leading to a set of recommendations for improving the LTM practices to achieve enhanced safety.

Procedural aspects of conducting the AIRT are as follows. For each major long-term accident management action identified within the scope, the AIRT members were provided a template containing information about the scope of actions associated with each major goal, and tools and technology as well as procedures to perform each action. The members were asked to rank, based on their knowledge and expertise, the likelihood of success of each action in accomplishing the LTM objectives. Individual ranking was performed in a qualitative scale of high (H), medium (M), and low (L) with regard to how, in the individual's opinion, a particular management action is likely to succeed in accomplishing the LTM objectives. The members were also asked to provide individual input on gaps in technology, procedures, and other factors, as well as rank them in a similar qualitative manner. The individual scores in both the likelihood of success and the technology gaps were then discussed among the AIRT members, at the conclusion of which a consensus score was assigned in each category.

The objective of the AIRT is to identify open issues and gaps whether stemming from challenges dealing with the likelihood of success, whether from any gap such as technological, procedural, knowledge-based, etc., linked to the main actions carried out during the long-term phase of the severe accident as currently defined in this project. In this regard, *the most relevant AIRT outcomes to pay attention to will be those actions ranked as low (L) consensus score with regard to their likelihood of success and high (H) consensus score with regard to gaps*.

Notwithstanding the above, the entire spectrum covered by the long-term phase of the severe accident presents to a more or less extent certain degree of uncertainty. That is to say, issues or gaps can be identified for nearly the whole list of related actions.

This is why additional relevant issues should also be lifted up from those actions ranked as medium in terms of likelihood of success and/or gaps, since such “medium” means that limited yet existing challenges and/or gaps for the action to be accomplished have been identified.

Therefore, and even if not as highly relevant as the former ones categorised with L/H combinations, medium ranked actions should be analysed to highlight the associated open issues and/or gaps. In this way, a threshold to focus the attention to will be fixed between those actions presenting a medium or high value against those ones only ranked with low significant related marks. And within the selected actions, conclusions derived from those ranked as L/H should be prioritised.

It is worth noting that unlike traditional PIRT where actions are ranked upon their importance, all accident management actions are deemed important and necessary by definition; thus their importance will be neither discussed nor ranked.

Elicitation process

Individual process of assessment

Besides the AIRT panel member’s expertise, any kind of ranking activity largely relies on subjective cognitive thinking processes taking place along the time interval an issue is being taken into consideration. Factors such as scenario characterisation and forecast, or assumed hypotheses taken as initial and boundary conditions, might contribute to discrepancies in judging the success likelihood of an action.

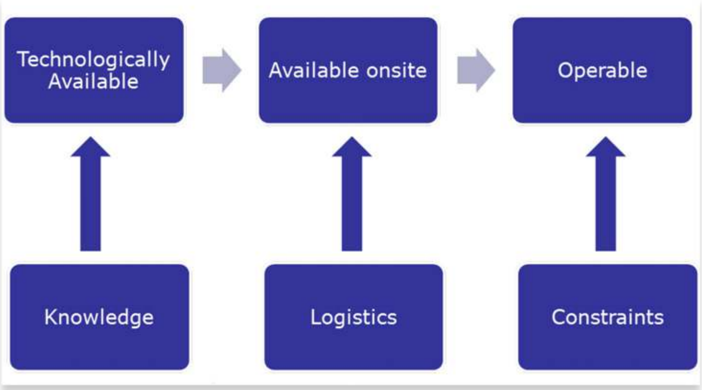
To address such lack of transparency and provide with an as-objective-as-possible process, the success likelihood ranking – together with the technology gaps ranking – will be further described based on the particular aspects characterising the long-term management actions as included in the tables.

The idea is to provide with a traceable path whose application, on one hand, allows readers to have a clear understanding on the reasons behind the given rankings, and on the other hand, allows panel members ranking the actions based on the same rationale.

Likelihood of success

When looking at the actions developed along the LTM phase, their likelihood of success comprises the different challenges found throughout the arrangement settings, disposition and performance of an action in accomplishing its goals. This means that in order for an action to carry out and fulfil its functions, several barriers of different nature must be overcome. Such barriers cover the entire process from the very beginning – at the design or knowledge phase – to the very end – at the time of actual implementation of the action on-site. Figure 3.5 below graphically represents such process where the main items are afterwards explained.

Figure 3.5. **Aspects considered for deciding on the likelihood of success of a given long-term management action**



Technologically available/knowledge issue: Tools

An action usually relies on devices[15], and these devices may be available worldwide and technologically mature enough to perform its intended functions.

Harsh environment conditions typical of a severe accident might challenge LTM via two different ways: by jeopardising the equipment performance because of going beyond design specifications (addressed under this issue), or by preventing the implementation or handling of the equipment (addressed under the accessibility issue).

The success of an action will also largely depend on the knowledge of i) the environmental conditions for the use of the device ii) the device functioning in the environmental conditions iii) challenges associated to the action to perform (e.g. characteristics of matter to treat for fuel retrieval, for waste management, etc.). One has thus to consider uncertainties associated to the detailed knowledge of plant damaged state.

Available on-site/logistics: Provisions

While the device can be found in the industry, i.e. it has been designed and manufactured, such device must be found on-site in order to be implemented. Depending on the issue, accommodating the physical needs underlying every action can be more or less demanding. For instance, water contamination handling may need large free areas to accommodate the storage tanks and filter equipment that might not be easily available at the plant site.

Operable/constraints: Accessibility

Even if the device exists in the industry and is found on-site, the device must feature operability. Such operability may require performing local actions to connect the device or just place it where it is supposed to perform its function. Uncertainties on the scenario where such device must fulfil its goal, or on how well the device copes with the harsh environmental conditions, can lead to a device which is technologically available yet where there is reasonable doubt on whether the device can be placed and implemented exactly when needed. For instance, devices to remove the core from the containment may exist and be technologically demonstrated and even available on-site yet they cannot perform their function because of high radiation level preventing operators to handle the device inside containment.

15. In this context, the word "device'" plays the equivalent role than the acronym SSC (structures, systems and components), that is to say, it refers to any kind of equipment, device, system or even entire installation whose correct performance is necessary to fulfil the pursued function.

Going through the three fundamental challenges jeopardising the implementation of an action, each of them may be qualitatively ranked into three different categories according to Table 3.3a.

Table 3.3a. **Classification of challenges associated with implementation of long-term management actions**

State	Challenge		
	Tools	Provisions	Accessibility
True (T)	The device is properly designed and qualified to tackle with environmental conditions and challenges linked to action fulfilment.	The necessary tools, related human actions, logistics, allocated surface, etc., do exist on-site and a dedicated plan to implement the action is available or the action is simple enough so that no dedicated plan should be followed up.	No constraint of any type preventing the action to be implemented exists dealing with context concerns such as physical impediment, high-level dose, etc.
Partly true (PT)	Uncertainties stemming from coping well with any type of LTM scenario at the technology level exists.	The necessary equipment (as listed above) does not exist on-site but it is likely (including time constraints) to bring it into the plant and accommodate it when needed.	Reasonable doubts due to aleatory or epistemic uncertainty in the accident evolution – before and after entering into the LT phase – exist to access to the locations where the equipment must be placed to perform its function.
False (F)	The technology to build up the device used to perform the action has not been designed or manufactured.	The equipment (as listed above) is not available on-site and it cannot be brought to the plant easily (including time constraints).	Environmental constraints are likely too demanding to avoid getting closer to the location where the equipment is intended to perform its function.

Technology gaps

Regarding the second ranking process of each action, namely the "technology gaps", this element covers a broader scope than the above mentioned challenge of "tools" affecting the likelihood of success. In particular, any gap in technology but also in procedures or necessary knowledge to fulfil the action should be estimated here. Therefore, a false condition in tools will lead to a low ranking in technology gaps but not necessarily the other way around.

Table 3.3b collects the material conditional logical connectives between the challenges to the likelihood of success and the two ranked elements for each action.

Table 3.3b. **Logical connectives used to fill in Table 3.4**

Tools	Provisions	Accessibility	Likelihood of success	Technology gaps
True				Low
Partly true				Medium
False			Low	High
	False		Medium/low	
True	Partly true		Medium	
Partly true	True		Medium	
		False	Low	

It is worth noting that in contrast with a false value in tools or accessibility, a false value in the provisions side does not necessarily lead to a low value in the likelihood of success since depending on the infrastructure status and the LTM goal evolution, e.g. available time for the operators to maintain the plant under a safe stable state by only using the existing resources on-site, the function might be successfully met or not. On the contrary, a false value either in tools or accessibility directly leads to a low likelihood of success for obvious reasons, i.e. whether the equipment has not even been designed and manufactured or – even if available on-site – the plant conditions do not allow the action being implemented, the device will not be operative so that the likelihood of success must be necessarily low.[16.]

The likelihood of success and the technology gap will then be ranked as a set of particular combinations between states and challenges according to Table 3.4, where the most relevant outcomes, i.e. L/H combinations, have been highlighted within a black-frame box. Therefore, the likelihood of success is derived from the three aforementioned issues of tools, provisions and accessibility, whereas the state of the tools issue directly relates to the technology and knowledge gaps.

Table 3.4. **Combination of classification of long-term management challenges and derived connection with "likelihood of success" and "technology gaps"**

Challenge			Likelihood of success	Technology gaps
Tools	Provisions	Accessibility		
T	T	T	High	Low
T	T	PT	Medium/high	Low
T	T	F	Low	Low
T	PT	T	Medium/high	Low
T	PT	PT	Medium	Low
T	PT	F	Low	Low
T	F	T	Medium	Low
T	F	PT	Medium/low	Low
T	F	F	Low	Low
PT	T	T	Medium/high	Medium
PT	T	PT	Medium	Medium
PT	T	F	Low	Medium
PT	PT	T	Medium	Medium
PT	PT	PT	Medium/low	Medium
PT	PT	F	Low	Medium
PT	F	T	Low	Medium
PT	F	PT	Low	Medium
PT	F	F	Low	Medium
F	T	T	Low	High
F	T	PT	Low	High
F	T	F	Low	High
F	PT	T	Low	High
F	PT	PT	Low	High
F	PT	F	Low	High
F	F	T	Low	High
F	F	PT	Low	High
F	F	F	Low	High

16. Unless accessibility can be recovered through decontamination and cleaning.

Collective process of assessment

AIRT activities usually reach consensus in individual ranking elicitation whether by integrating or agreeing the different votes.

Integration processes can follow quantitative or qualitative methods. Looking at the former, they are usually based on applying a weighted average to the number of votes ranking the item differently, and then linking different intervals, within which the numerical result is assigned, back with a specific ranking level. Such approach can be of importance for large PIRT panels whereas for the current case, due to the relatively low number of participants in the elicitation process, reaching common consensus through open discussion was taken as the optimal solution.

After collecting and summing up all the individual votes, the panel went through each item during a two-day meeting so that any concern and discrepancy in the interpretation and assessment was raised up and openly argued by the panel members. This consensus-based method leads ultimately to a single ranking level in most cases, and any divergent opinion and discussion therein are incorporated as side notes to the table.

Presentation of the results

The AIRT is organised upon LTM goals. For each goal, a table of actions will be presented. Within each table, each action or item will feature a twofold common-consensus ranking in terms of likelihood of success and technology and knowledge gaps, together with an additional last box highlighting the main challenges and open issues related to that item.

Alongside with each table, the two given rankings will be explained per item according to the arguments presented during the panel discussions. If non-reducible discrepancies were found, the reasons behind such lacking of consensus will be incorporated into the text and the main insights raised up during the meeting will be added in the third column.

It is worth noting that the supporting comments describing the underlying reasons and discrepancies for each action are as important as their final votes since they give the clue for having a good understanding of the results.

Comments and results of the action identification and ranking tables

General preliminary aspects

In order to assess and rank the actions, all panel members should share a common view of the initial and boundary conditions characterising the LTM scenarios, while recognising the wide spectrum of accidents falling under the long-term phase of a severe accident. The minimal conditions that all LTM scenarios should meet are the following:

A severe accident involving limited or extended core damage has occurred but the current status meet the LTM entry conditions as specified in the definition (Section 3.1):

- the decay heat is successfully being removed;
- no flammable situation exists on-site and there is no further hydrogen and carbon monoxide generation from steam-metal reaction or MCCI;
- the fuel is maintained subcritical at all locations;
- the containment pressure and RCS pressure are kept low;
- the release of fission products is exceedingly small;
- same conditions apply to the SFP.

Three options have been considered in the ranking exercise:

- a severe accident has occurred in the reactor;
- a severe accident has occurred in the SFP;
- a severe accident has occurred in both.

However, it appeared during the ranking exercise that some panel members mostly focused on in-reactor severe accidents where others did consider the three options.

Due to the wide variation LTM scenarios embrace, and since each action is assigned to a single ranking level, this level should ideally bound all type of LTM scenarios, and in case of highly depending on the magnitude of the accident, it should look at the worst situation to be on the conservative side. Appropriate side notes may then clarify the issue. Therefore, the AIRT tables should not be linked to any particular accident scenario.

Comments and results

4) Maintain coolable configuration and manage confinement

Inject water in-vessel

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Long-term performance of equipment uncertain. Back-up systems should be pre-planned.

Likelihood of success: Provided equipment is appropriately qualified and water sources adequately provisioned. The action has already been successfully implemented as a fundamental piece of the LTM entry criteria meeting conditions hence the ongoing part of the action is the only one here being assessed.

Technology and knowledge gaps: Robust designs and criteria exist for equipment and systems; procedures are validated in simulated environment. The necessary allocated resources in terms of structure, system and components, i.e. valves, pumps and heat exchangers, do not cope with highly ambiguous scenarios after having reached LTM entry conditions.

Raised concerns: The main arguments claimed by the panel members not agreeing with the high likelihood of success have to do with the technological issue on one side and the accessibility issue on the other. Regarding the former, the in-vessel injection, as well as the containment injection, can be carried out in permanent direct mode or by means of recirculation mode using either fixed and/or mobile equipment. The recirculation mode, even if a commendable practice because of avoiding further issues stemming from handling large masses of contaminated water, can contribute highly to risk as its good performance rely on a considerable number of components sensitive to fail. For instance, loss of long-term core/containment cooling can result from debris-induced direct clogging of the recirculation cooling loop. Efficiency of heat exchangers can also degrade because of fouling by deposits onto the surfaces coming out of contaminated water. When the recirculation system takes the contaminated water out of the containment, then it will be source of radiation and in the case of failure also source of leakage. Looking at the logic combinations as stated above in Table 3.3, doubts on technological performance will lead in average to medium likelihood of success.

Regarding the accessibility issues, depending on the level of radioactive releases and spread, reasonable doubts on necessary operability-related actions, e.g. maintenance and replacement, might be an issue difficult to cope with. A similar concern was raised if the action is performed through emergency core cooling system in recirculation mode since contaminated water will circulate outside containment (for certain designs), preventing field operators to get close enough to implement eventual necessary actions that must be taken locally. Looking at the logic combinations as stated above in Table 3.3, lack of accessibility will directly lead to a medium or even low likelihood of success.

Inject water in containment

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Long-term performance of equipment uncertain. Back-up systems should be pre-planned.

Likelihood of success and technology and knowledge gaps: the action of injecting water directly to the containment to remove the decay heat thereby depressurising and cooling down the containment shares almost every view and issue with the former action of injecting water into the vessel: the likelihood of success will then be ranked high and technology gaps low because of the same arguments presented above.

Dealing with raised concerns, the same issues as underlined above are here highlighted, i.e. debris-induced problems in recirculation-related devices; taking contaminated water outside containment/reactor building; accessibility issues in case local actions are needed; and portable equipment – thus related personnel in charge of operating the system – exposed to high dose environments, internal equipment failure along the long-term phase; similar clogging issues as mentioned in the previous action; etc.

RPV pressure control

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Plants should be equipped with reliable pressure control systems.

Likelihood of success: pressure control procedure and technology are robust and should not be an issue for LTM; provided equipment is appropriately qualified and strategy anticipated for station black-out (SBO) conditions.

Primary pressure will only contribute to meeting LTM goals under very limited scenarios: once achieved LTM entry conditions, this action will only be relevant provided the RPV has not failed. The primary system can be depressurised by correctly following SAMGs usual first action of depressurising the primary system through the pressuriser pilot operated relief valves or through the secondary side by means of steam generators cooling down for PWR designs, or through the primary system safety relief valves for BWR designs, or unintentionally in case of a loss of primary system integrity.

The need of reliable primary depressurisation for PWR was clearly recognised and existing plants are being upgraded by depressurisation systems reliable in severe accident conditions.

Technology and knowledge gaps: redundancies built into current designs and utilities have robust procedures; technology gap, if any, is largely a material performance issue.

Raised concerns: safety relief valves can play an additional role in PWR designs in case of secondary side recovery acting as a heat sink since the RCS will have to be first repressurised by closing the safety relief valves. Such heat transfer mechanism can substantially improve the LTM management by avoiding further issues related to handling large masses of contaminated water, and by significantly reducing the sources of risk threatening the goal of maintaining a coolable configuration by removing heat through a clean system fully separated from the fluid in contact with the radioactive source. Safety relief valve performance under highly contaminated, dirty water and harsh environment conditions should be therefore further investigated.

Maintain RPV integrity

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Medium	Knowledge of late-phase melt progression. Remaining strength of damaged RPV cannot be assessed. Materials performance issues; embrittlement, corrosion.

Likelihood of success: RPV failure may be avoided in design extended condition B-like scenarios – extended core-damage scenarios with partial or full core relocated to the lower plenum by injecting water into the vessel and/or into the reactor cavity. Significant uncertainties exist on the damaged status of the RPV (with possible fragilities induced locally by contact with hot materials) and whether the damaged RPV will bear the thermal and mechanical loads in the long term, even if it had resisted in the short term.

According to Table 3.3, a partly true statement in tools corresponding to *uncertainties stemming from coping well with any type of LTM scenario at the technology level*, directly yields medium ranks for the likelihood of success.

Technology and knowledge gaps: re-establishment of the in-vessel injection after successfully prevented RPV failure by ex-vessel cooling, long-term creep and embrittlement effects on the lower head materials, long-term resistance of damaged RPV and risk of failure, long-term effect of minor leakages, solidified corium retrieval operations from the RPV, etc.

Raised concerns: no additional long-term safety concerns coming from discrepant opinions other than those already pointed out above.

Containment pressure control

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Related to hydrogen control, see next item.

Likelihood of success: after entering into the LTM phase, the containment pressure should have been kept low enough to minimise further environmental release of radioactivity. This condition – together with the fact of meeting with the rest of the LTM entry criteria – makes that containment pressure control will hardly become an issue in the long term.

Several reliable means of containment pressure controls exist including containment sprays, hard venting and filtered containment venting system, decay heat removal by portable equipment or standard emergency core cooling system in injection (temporary) or recirculation mode, etc. Procedures for implementing these systems in any combination are well established.

Technology and knowledge gaps: robust designs and criteria exist for equipment and systems; procedures are validated in simulated environment.

Raised concerns: containment leak tightness partly relies on containment penetration materials exposed to typical harsh environment conditions. Experiments on penetration materials have been made by using nitrogen instead of hydrogen so that aggravated corrosion mechanisms have not been taken into account on their survivability. In addition, containment liner may undergo galvanic punctual corrosion thereby leading to containment leakage. With respect to containment cooling systems, such as fan coolers or containment sprays, aerosol deposition onto the heat exchanger surfaces or sprays nozzles can soundly degrade the heat transfer mechanism. Recirculation sumps blockage is also an issue if containment sprays is the removal heat mechanism. Portable equipment suffers from the same problems than the in-vessel injection action.

Containment hydrogen control

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Radiolysis needs to be considered.

Likelihood of success: once reached LTM entry conditions, hydrogen (and carbon monoxide) production must have lowered below safety thresholds. Provided control and monitoring of hydrogen is established and mitigation means appropriately designed, the likelihood of success should be high.

Technology and knowledge gaps: the only significant sources of hydrogen generation are water radiolysis and corrosion reactions, which will hardly stop even after entering into the LTM. Even though the generation rates are likely to be low enough to be easily detected before departing from a safe state, the main concern arises from flammable gases accumulation in places where no detection nor mitigation measures are provided. Such build-up situation is not limited to the reactor containment.

Raised concerns: N/A

Confinement of fission products

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Medium	Uncertainties regarding status of confinement, chemistry and remobilisation processes of fission products in the long-term affecting releases.

Likelihood of success: fission products confinement on the LT very much depends on the containment barrier status when a coolable configuration is reached and on the extent of releases either airborne (gas and aerosols) or liquid that occurred through containment breaches during the emergency phase of the accident.

If the containment leak tightness is not affected during the emergency phase, fission products confinement is achievable. Then with unaffected containment, the likelihood of success of fission products confinement is high.

Maintaining confinement of fission products is of course much more challenging if the containment leak tightness had been lost during the emergency phase. In such situations, the likelihood of success of fission products confinement is lower. Containment decontamination and debris removal will be more challenging.

With these considerations, the likelihood of success was classified medium.

Technology and knowledge gaps: potential remobilisation of gaseous iodine species in the weeks following the accident from pools, sumps, surface deposits (dose effects, evolution of water phase chemistry in scrubbers, in suppression pools, in sumps, radiolytic decomposition of deposited aerosols on surfaces and filters, effect of impurities and corrosion reactions). Leaching and transfer of Cs isotopes, Sr, Pu, Am in waters on the LT depending on water phase chemistry.

Raised concerns: N/A

Minimise environmental release (airborne and aqueous)

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Medium	Knowledge/technology gaps mainly concerning aqueous release.

Likelihood of success: airborne and aqueous release control requires previous analysis of the status of the containment and identification and control of the potential leakages, extending those activities to potential containment bypass situations, e.g. through the steam generators (for PWR designs) or primary interfacing systems, or recirculation interfacing system loss-of-coolant accident (interfacing loss-of-coolant accident during the recirculation switch connecting containment/primary systems with the outside environment). Depending on the severe accident evolution, pinpointing and control of releases might be a very complex issue to tackle (cf. confinement of fission products).

Technology and knowledge gaps: current ways for analysing and pinpointing the exact release locations are not satisfactory enough and they can become crucial to more suitable and faster face the problem and isolate the leakage; leak tightness of penetrations; of loops conveying contaminated liquid; survivability and trapping efficiency on LT of loaded filters in filtered containment venting systems; remobilisation from aerosol deposits and from pools and sumps (iodine); transfer to liquid phases (Cs, Sr, Pu, Am).

Raised concerns: N/A

Minimise likelihood of criticality

Likelihood of success	Technology gaps	Challenging/open issues
High	Medium	Risk of re-criticality sometimes difficult to assess.

Likelihood of success: likelihood of re-criticality small and effects of re-criticality events probably limited; re-criticality assessment is possible but challenging due to uncertainties on damaged fuel distribution and composition; monitoring and control is possible; non-borated water injection increases the risk. Criticality can be monitored by taking containment gaseous samples or by placing online monitors at containment ventilation outlet path, for instance, by measuring ^{135}Xe concentration and can be controlled by borated water injection.

Technology and knowledge gaps: risk of re-criticality is difficult to assess, Pu clusters formation upon cooling of corium in particular for mixed oxide fuel may increase the risk; clear water injection or boron dilution may increase the risk. Also, when starting to retrieve debris, risk of criticality is difficult to assess (e.g. when reflooding some years after the accident dried aged debris).

Raised concerns: potential delays between event detection and implementing appropriate countermeasure actions, e.g. boron acid injection.

Maintain SFP cooling and minimise risk

Likelihood of success	Technology gaps	Challenging/open issues
High (loss of cooling) Low (loss-of-coolant)	Medium	Validated tools to calculate progression and consequences of SFP accident.

Likelihood of success: the action is maintaining SFP cooling on the LT following a loss of cooling or loss-of-coolant accident in the SFP.

The likelihood of success is considered high for loss of cooling accident where limited fuel assemblies degradation and radioactive releases can be expected and due to the possible various means at effectively injecting water into the SFP.

The likelihood of success is considered low for unmitigated loss-of-coolant accident where large fuel assemblies degradation and radioactive releases can be expected. The success of the action may be challenged by SFP structure damages and reduced accessibility (in case of large releases on-site, particularly for SFP outside containment).

Loss-of-coolant accidents with fuel melting in SFP should be "practically eliminated" according to the Western European Nuclear Regulatory Association (WENRA), hence provisions for prevention of SFP accidents should be adequately verified.

Technology and knowledge gaps: tools to calculate progression of loss of cooling and loss-of-coolant accidents in SFP and assess their consequences, with consideration of plant-specific SFP configurations, need further development and validation.

Raised concerns: N/A

The resulting common ranking for actions aiming at maintaining the coolable configuration is shown in Table 3.5.

2) Water waste, solid waste, and effluent management

Minimise and collect contaminated water

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Low	If containment is leaking, it is a considerable challenge to handle contaminated water. Also, if cooling is not done on secondary side, large amounts of contaminated water are generated from cooling.

Likelihood of success: the aspects making collecting contaminated water a challenge are logistics in nature while technologically speaking this action does not constitute an open issue.

The complexity of successfully connecting the necessary devices to collect the water highly depends on whether the decay heat is removed through an open or a closed loop, and even much more relevant, on whether the containment has been kept isolated or it has breached. Even if the operators manage to make the contaminated water circulate through a closed loop provided with a pump and heat exchanger, in case there is a leakage in or out containment, part of the water will leave the containment so that it should have to be identified, stored and suitably treated. As a matter of fact, if one of the most critical issues at Fukushima Daiichi is currently collecting and treating large masses of contaminated water, it is just because the underground water coming into the reactor building is mixing with the water leaking from the containment. The underground water would not be a challenge if it could be pumped out or simply be deviated outside the building. Large efforts are currently implemented at Fukushima Daiichi NPP in that purpose.

Therefore, the action was ranked as medium because of partly true in terms of provisions.

Technology and knowledge gaps: robust designs and criteria exist for equipment and systems. The necessary allocated resources in terms of devices, i.e. valves, pumps and heat exchangers, do not cope with highly demanding scenarios.

Raised concerns: several panel members expressed different views on how likely this action is to be successfully implemented. Along with the specific challenge in implementing the action wherein the emphasis was placed by each member – basically whether on tools or provisions –, divergences were mainly rooted in the different scenarios foreseen since the success in implementing this action is highly scenario-dependent. According to the degree of core damage – arrested in-vessel or ex-vessel–, the status of the containment, and even more the confinement of radioactive products, to handle the resulting contaminated water might range from an easy action to deal with to a complex one.

Decontaminate collected waste water

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	No technology available to separate tritium in very large volumes and very low concentrations; capacity may be a problem; water treatment facilities may not be readily available as well as characterisation techniques (not easily detectable nuclides).

Likelihood of success: once arrangements for contaminated water collection are set up, equipment needed to decontaminate water is not technologically challenging (cf. TMI-2 and Fukushima Daiichi NPP feedback). Use of raw water or sea water may however complicate decontamination operations.

Logistics issues dealing with handling and storing large amounts of water, together with suitable further decontamination of derived by-products do not fall under this action. This action also shares with collecting contaminated water the lack of NPP preparedness in allocating the necessary devices and planning the necessary projects should it be needed. Nonetheless, such allocated efforts are ranked as not as demanding as those required for collecting contaminated water. This is why rather than a medium likelihood of success, this action is ranked as high.

Technology and knowledge gaps: equipment needed to processing contaminated water is appropriately qualified and extensively tested. The only remaining open issue deals with the current unavailability of separating tritium in very large volumes of water.

Raised concerns: tritium issue; logistics issues but this last one actually assessed within the former action so not applying here.

Recirculate decontaminated water for core cooling

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Damaged systems/piping may pose a problem.

Likelihood of success: when decontaminated water is available, its use for the core cooling is likely to be successful.

Decontaminated water for the core cooling is not necessarily completely clean water. It is water cleaned from any dirt, oil, dissolved salts etc. and decontaminated from the main radiation sources. The objective is that this water should not cause damage to pumps, valves and piping of the cooling system and the dose rate around the system should be kept in acceptable range.

Technology and knowledge gaps: no lack in the necessary equipment to perform this action is found.

Raised concerns: arrangement for a reliable recirculation alignment of necessary equipment including pumps, heat exchangers, valves and piping might be an issue if making use of devices or piping having undergone damage yet not easily identified because of harsh environment conditions, i.e. accessibility problems. From this point of view, the action should not be categorised as high but as medium in terms of the likelihood of success.

Collect and store waste water

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	The sheer volume of waste water may be problematic. Release criteria should be defined.

Likelihood of success: when focusing on the storing side of the more overarching action aimed at collecting and storing contaminated water, this action becomes as highly likely of being successfully implemented.

Technology and knowledge gaps: no lack in the necessary equipment to perform this action is found.

Raised concerns: the storage of large amount of contaminated (or in certain cases almost clean) water may prohibit other activity on-site due to lack of space. If this issue is taken into account, this action should be reassessed as medium in terms of the likelihood of success.

Monitor and control fission products including tritium in water

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Monitoring techniques may not be readily available for all radionuclides. Monitoring of some not easily detectable nuclides.

Likelihood of success: monitoring is well established for major isotopes (e.g. Cs, Sr) thus the likelihood of success was considered high. Also this action does not present strong challenging issues in terms of necessary equipment or knowledge, provisions (allocated resources could already be on-site or easily and rapidly delivered) and accessibility (no need to perform local actions where harsh environment conditions might prevent their correct implementation).

Technology and knowledge gaps: technology available and proven measurement methods well established for major isotopes. However, the monitoring of other isotopes (e.g. actinides, some activation products) is more challenging in highly active waters. "Routine" measurements techniques and methods for monitoring of all important nuclides in highly active waters for LTM could be further established and shared.

Raised concerns: additional development of devices for routine activity measurement and contamination evaluation required; international standards for measured values such as "gross beta" should be more openly advertised; procedures for results analyses should be promoted; different techniques and methods may be required for monitoring radionuclides for very different water contents.

Control and minimise effluent discharge

Likelihood of success	Technology gaps	Challenging/open issues
Low (if containment leak tightness has been lost)	Medium	Knowledge of containment status (leak paths). Technics/methods to recover the containment.

Likelihood of success: the arrest of existing leaking paths might be extremely difficult to cope with not because of the necessary tools falling short in meeting their function – since the action itself merely consists of plugging, welding, etc., the undesired radioactive leakages –, but because of highly demanding accessibility issues. These issues are mainly related to identifying the release location and to implementing the necessary local actions.

Technology and knowledge gaps: actions aimed at controlling and minimising radioactive liquid effluents do not feature important gaps in terms of knowledge/devices. Possibly, there might be some interest in developing technics and methods for recovering the containment leak tightness.

Raised concerns: no additional issues raised aside from those already included above.

Effluent characterisation

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Cf. challenges mentioned for fission product monitoring in waters.

Likelihood of success: this action requires applying well-known techniques from feedback of accidents, knowledge of fission products of interest and adequate measuring techniques. Significant challenges neither found in tools, provisions nor accessibility.

Technology and knowledge gaps: no fundamental gaps are found; long-term sampling is feasible; analyses of highly active waters may be challenging (multi-nuclide analyses).

Raised concerns: no additional issues raised aside from those already included above.

Collect, store solid wastes and monitor fission product contents

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Monitoring of fission product content in highly active and complex solid structures. Accessibility and handling of highly active solids (cf. defuelling).

Likelihood of success: here we consider solid wastes other than wastes related to damaged fuel and SFP fuel assemblies' retrieval (treated later in point 5). Collection and storage of solid wastes is considered with a high likelihood of success. Tools and technologies to collect and store them temporarily on-site are considered well known. However, monitoring of fission products in highly active and complex solid structures and accessibility to and handling of most active wastes may be challenging.

Technology and knowledge gaps: no fundamental gaps are found, possibly development of technics and methods for more "routine" fission product content measurements for highly active wastes could help solid waste management (storage and disposal).

Raised concerns: no additional issues raised.

Storage and disposal of filtered containment venting system water and filter

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Monitoring of fission product content in highly active solid and liquid wastes (see above). Accessibility and handling of highly active solids (see above).

Likelihood of success: see addressed arguments above dealing with collecting and storing solid and liquid wastes.

Technology and knowledge gaps: idem.

Raised concerns: no additional issues raised with the exception that filtered containment venting may still play a role in the long term for containment pressure and hydrogen control and to limit radioactive releases.

In case of liquid scrubber-type filter, provisions are set so that the contaminated liquid can be periodically removed if required. Issues associated to collection and treatment of highly active contaminated waters apply.

For solid filters (solid filter stages exist both in solid- and liquid-type filtered containment venting systems), removing of the filters might not be feasible for a long period of time if venting has to remain operable. This is favourable for power and dose decay in the filter with the elimination with time of short-lived isotopes (e.g. iodine isotopes decay).

In case filter clogging occurs during the accident, this may prevent possibilities for further venting. The action of removing the clogged filter will be difficult to implement in particular due to high doses on the filter.

Solid filters collecting aerosols (in particular pre-filters implemented upstream some venting lines) are expected to concentrate large activities even on the LT (Cs aerosols filtration). Then the action of retrieving the filter may be difficult to implement due to high doses on the filter.

The resulting common ranking for actions aiming at water waste, solid waste, and effluent management is shown in Table 3.6.

3) Site clean-up and decontamination

Plant site clean-up and rubbles/sludge removal (outside of the reactor)

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	If major releases have occurred, the likelihood of reaching green field status may be low. Rubbles/sludge characterisation for waste management.

Likelihood of success: technological gaps are not foreseen, as well as significant shortcomings in accessibility issues.

Technology and knowledge gaps: none foreseen. Site infrastructures may be developed to facilitate site clean-up if an accident occurs.

Raised concerns: while the suitable technology exists, removal of sludge and cleaning up of the site might be difficult to implement in those areas with low accessibility due to high dose rates. Also, large infrastructures damages (e.g. resulting from natural hazards or explosions or fires) will make the action implementation more challenging. If such conditions are considered, the likelihood of success should be ranged as medium.

Active rubble/sludge characterisation (fission product content) for waste management could be challenging (cf. solid waste management).

Surface decontamination

Likelihood of success	Technology gaps	Challenging/open issues
High (outside damaged plant) Medium (in highly contaminated zones of damaged plant)	Low	Accessibility may be limited due to contamination, physical constraints, water coverage. Contamination may have diffused into structures (e.g. in concrete). Implementation of strippable or washable coatings in zones were high contamination is expected (e.g. in containment).

Likelihood of success: likelihood of success of surface decontamination (including soil) on-site outside damaged buildings is expected to be high referring to recovery actions that were implemented for instance to clean the Fukushima Daiichi site.

However, the success of surface decontamination in highly contaminated zones of damaged plants has been ranked as medium because accessibility may be limited and contamination may not be easily removable as was evidenced at TMI-2 and Fukushima Daiichi. Despite many different decontamination techniques used for surface decontamination in TMI-2 containment, decontamination was very demanding and very abrasive technics had to be used. At Fukushima Daiichi, for example decontamination at the level of the refuelling floors to give access to the RPV, still appears very challenging. This is in great part due to contamination which diffused into concrete structures or accumulated in non-accessible zones or pipes.

Technology and knowledge gaps: many technics for surface decontamination exists but encrusted contamination is difficult to remove. Implementation of strippable or washable coatings in zones where high contamination is expected may be of help (e.g. in containment).

Raised concerns: similar comments dealing with accessibility as set forth in above action dealing with site clean-up apply.

Rubbles and sludge characterisation

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Robotics and remote measurement technology could be helpful. Composition, content of burnable substances, etc. are also of interest for transport and disposal. The sheer amount of debris may be a challenge.

Likelihood of success: this action has been analysed by most of panel members as if it only comprised the task of characterising debris itself. From this aspect, the action has been ranked as high since debris/sludge characterisation is not overly challenging except for fission product content characterisation for waste management as discussed above.

Technology and knowledge gaps: depending on the situation, debris characterisation can be performed by already existing devices. See also the related open issues.

Raised concerns: as many other actions involving local actions, the emphasis might not be put on characterising the debris itself but rather on the prior, necessary action to access the debris, which can be locally prevented by high dose rates.

Monitor cooling pond (and sea) contamination

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	None

Likelihood of success: as any other action dealing with monitoring activities, the current one is ranked as high – see for instance above the *monitor and control fission products including tritium in water*. This should be "routine" measurement.

Technology and knowledge gaps: none.

Raised concerns: none.

Monitor and control groundwater contamination

Likelihood of success	Technology gaps	Challenging/open issues
Low	Low	Specific technological solutions to limit groundwater contamination are highly situation/plant specific.

Likelihood of success: the fundamental reason why this action features a low ranking relies on the difficulty to controlling the groundwater contamination, mostly in the light of the Fukushima Daiichi NPP highly challenging scenario dealing with the groundwater issue. Further, an accident where the containment concrete floor would be breached by MCCI could also lead to significant groundwater contamination.

Such low likelihood of success derives from accessibility problems involved in the implementation of countermeasures to prevent groundwater to be contaminated, in particular because such countermeasures should aim at recovering containment leak tightness.

Technology and knowledge gaps: none.

Raised concerns: consensus in this action was achieved through long clarification discussions about different plant characteristics stemming from extremely different severe accident signatures, i.e. negligible importance in many scenarios (both looking at the severe accident evolution and the plant site, e.g. TMI-2) but highly relevant in some other ones, e.g. the worst case scenarios likely represented by the Fukushima Daiichi NPP accidents and the Fukushima Daiichi NPP site. In order to take the bounding case, the likelihood was finally ranked as low.

Radiological monitoring

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Technology is available; robotics and remote measurement technology would be helpful. Challenging in very high dose environment.

Likelihood of success: the same arguments for a high likelihood of success as stated above in monitoring-related actions apply.

Technology and knowledge gaps: none. Technology is available; robotics and remote measurement technology would be helpful.

Raised concerns: none.

Characterisation of fission products distribution in contaminated spaces

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Medium	Methods for source distribution analysis; improved techniques for hot spots detection and characterisation; difficulties to access highly contaminated areas; technology is available except for molten fuel location detection; codes cannot give this much detail.

Likelihood of success: reasonable doubts on tools and accessibility directly lead to a medium/medium ranking according to Table 3.3.

Depending on the extent of the contaminated areas, dose rate data analysis can be very complex and the retrieved data can be scarce from areas with low accessibility.

Technology and knowledge gaps: the main shortcomings as set forth in the likelihood of success paragraph would significantly decrease with advancement of clean-up and additional measurement campaigns. Efforts should be placed in developing innovative devices and instrumentation to have a better characterisation of fuel and radionuclides distribution in a damaged plant.

Raised concerns: aside from a highly scenario-dependent action in terms of its likelihood of success, no additional concerns were raised and a common agreement was achieved.

4) Defuelling of damaged reactors

Detect fuel debris locations and estimate debris masses (at start of defuelling)

Likelihood of success	Technology gaps	Challenging/open issues
Low	High	Feedback from accidents shows this action remains highly challenging. Particularly difficult for accident with failed RPV and/or larger fuel dispersion. With existing monitoring (imaging), no reliable information until access to damaged fuel is possible (accessibility, dose). Distant monitoring with sufficient resolution not yet developed. SA codes not sufficiently predictive. Required/wished progress in technics and analytical tools scientifically challenging. Damaged fuel and debris distribution poorly known up to start of defuelling, selection of appropriate defuelling options and techniques delicate at this stage.

Likelihood of success: the feedback from TMI-2, Chernobyl and Fukushima Daiichi NPP accidents shows determination of damaged fuel distribution after a severe accident remains a highly challenging issue. In the first five to ten years following such accidents, damaged fuel cannot be easily accessed due to high doses, even remotely, which prevents having information on its distribution in the damaged plant. Further, accidents with a failed RPV and with larger fuel relocation or dispersion outside RPV will make the action even more difficult. The traditionally used monitoring means such as optical imaging by camera generally conveyed by robots fail to provide reliable information in high radiation environment (limited lifetime, localised measurements, low resolution and often accessibility problems).

Alternative technics offering distant monitoring could be further developed. Muon tomography technique, recently applied at Fukushima Daiichi NPP, provided useful by limited information on damaged fuel location – essentially providing evidence of presence of fuel or not in the RPV after the accident in unit 1 and 2 (no quantification possible due to low resolution).

At TMI-2, knowledge on fuel distribution in RPV was gained progressively by visual inspections, first at RPV opening about six years after the accident and then upon progress of defuelling operations.

On the analytical side, as evidenced by recent international benchmark exercises (NDA, 2011; NEA, 2015a), severe accident calculation codes (e.g. Accident Source Term Evaluation Code (ASTEC), Modular Accident Analysis Programme [MAAP], MELCOR, SAMPSON) do not yet predict reliably enough RPV failure and damaged fuel mass distribution in-vessel and, eventually, ex-vessel.

Based on these arguments, panel members considered that, at start of defuelling, establishing damaged fuel distribution has a low likelihood of success. Thus, since damaged fuel and debris distribution is expected to be poorly known up to start of defuelling, anticipative selection of appropriate defuelling options and techniques is delicate.

Technology and knowledge gaps: non-intrusive instruments dedicated to measure during the accident any significant fuel movement in the RPV could be searched for (e.g. detection during the accident of massive fuel relocation in RPV, of fuel relocation outside RPV). Such instruments could be highly valuable both for severe accident management and LTM. Alternative technics allowing distant monitoring of damaged fuel, such as muon tomography, could also be further developed for LTM. Such technical developments appear highly challenging especially if debris masses estimates are searched for.

On the analytical side, efforts are underway to improve modelling of SA codes, particularly on late phases of core degradation in-vessel.

Raised concerns: progress both in technology and SA codes pose scientific challenges.

Identify state of degraded fuel/fuel debris including control rod and structural material

Likelihood of success	Technology gaps	Challenging/open issues
Low	High	Determining the composition, fission products content and mechanical behaviour of degraded fuel and debris is highly challenging without sampling and analysis of real material. Predictive analyses can be made using thermodynamic calculation tools but these tools are only partially valid for complex compositions. Simulant materials testing may help but experimentation representative of actual compositions is challenging. Knowledge from performed fuel degradation experiments to be considered but not necessarily easily transposable to specific accident transients. Knowledge of ageing phenomena limited.

Likelihood of success: as for getting information on debris distribution, identifying the state of damaged fuel/fuel debris by optical imaging has a low likelihood of success for the reasons earlier exposed. When accessible, low resolution imaging of damaged fuel/fuel debris may be obtained providing information on their morphology and appearance. Determining the composition, fission products content and mechanical behaviour of degraded fuel and debris is highly challenging without retrieval and analysis of real material.

Predictive information on composition in-vessel and ex-vessel and physical properties can be searched for using i) knowledge gained through performed fuel degradation experiments ii) experimentation performed using materials simulating accidental complex compositions iii) analytical calculations such as SA codes and/or thermodynamic calculation tools. This is the approach currently deployed at Fukushima Daiichi NPP. However, the transposition of such knowledge to specific accident transients is not straightforward.

Ageing processes (incl. long-term leaching) may also have to be considered for cases where fuel retrieval can only be engaged (due e.g. to dose and accessibility limitations) long times after the accident (cf. Chernobyl).

Technology and knowledge gaps: progress in analytical tools (SA codes, thermodynamic databases) searched for, but this is scientifically challenging. It could be valuable to generate a knowledge database related to damaged fuel/fuel debris recovered and analysed in accidents and in relevant testing (composition and physical properties and ageing). Limited knowledge on ageing processes (dose effects, leaching) on the LT.

Raised concerns: progressing in analytical tools and performing representative experimentation are considered scientifically challenging.

Collection of damaged fuel/debris samples and characterisation

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Low	It may be actually challenging to retrieve samples. Specific remote control technologies may have to be developed depending on damaged fuel/debris characteristics. As seen above, high dose, knowledge of debris characteristics. Characterisation techniques for damaged fuel/debris are well known from previous accidents and degradation experiments. Large experience in hot labs.

Likelihood of success: specific remote control technologies (cutting, lifting, collecting, handling) may have to be developed depending on damaged fuel/debris characteristics (e.g. mechanical behaviour), location (accessibility from top, side, etc.) and conditions (under water or in air). Since getting information on damaged fuel/fuel debris distribution and characteristics is challenging, organising and designing tools for samples collection is considered to have a medium likelihood of success. With progress in damaged fuel retrieval and knowledge of its characteristics, the likelihood of success of the action should increase. More challenges will arise in accidents where damaged fuel has relocated in many different places (failed RPV, fuel dispersion).

Since characterisation techniques for damaged fuel/debris are well known and large experience was built in hot labs, there was no raised concern related to damaged fuel/debris characterisation.

Technology and knowledge gaps: as discussed above, damaged fuel/debris characteristics.

Raised concerns: none.

Fuel debris examination and characterisation for storage

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Knowledge of damaged fuel characteristics should be obtained during fuel retrieval operation Specific tools may have to be developed for characterisation related to criteria (mass, dose, fission products content) set for waste storage

Likelihood of success: damaged fuel/fuel debris characterisation should be conducted during fuel retrieval operations. Specific device may have to be developed to get data relevant for waste storage management (cf. point 2 related to waste management). Panel members considered that once challenges associated to damaged fuel retrieval have been solved, the likelihood of success of the action is high.

Technology and knowledge gaps: see above.

Raised concerns: none.

Minimise environmental release during defuelling

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Medium	Depending on damaged fuel distribution, locations and characteristics may be challenging to minimise releases during fuel retrieval operations Development of cutting techniques limiting aerosol generation and dispersion considering damaged fuel characteristics

Likelihood of success: the likelihood of success of the action has been considered medium as depending on damaged fuel distribution (e.g. fuel relocated outside RPV in different areas), location (in air or under water, low accessible and high dose areas) and characteristics (hardness, brittleness, eventually considering ageing), it may be challenging to minimise releases by implementing countermeasures (confined area, filtering, etc.). Further, new cutting techniques limiting dust generation and dispersion may have to be developed considering damaged fuel characteristics (e.g. hardness, brittleness). Some developments are being pursued in the preparation of damaged fuel retrieval at Fukushima Daiichi NPP.

Technology and knowledge gaps: medium for the reasons exposed above.

Raised concerns: none.

5) Damaged fuel/fuel debris and radioactive waste disposal, SFP fuel removal

Temporary on-site storage of damaged fuel/fuel debris

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Some challenge may arise due to volume, dose and space restrictions on-site. More challenging for multi-units, SFP accidents with significant fuel damage, accidents with significant fuel dispersion. May necessitate large-scale constructions (temporary confinement before retrieval, installations for storage after retrieval). Criteria for site location and/or storage containers not always defined. Criteria for appropriate handling and storage approach to be defined considering materials characteristics (dose, mass, material composition, fission products content).

Likelihood of success: the likelihood of success of the action was considered high even if challenges may arise for accidents where fuel retrieval is particularly complex (e.g. dispersed fuel) and when large volumes of highly active material are generated (SFP or multi-units accident). Then the temporary on-site storage of damaged fuel or fuel debris may necessitate large-scale constructions (e.g. temporary confinement before retrieval such as at Chernobyl, installations for storage of retrieved material).

As for any nuclear waste management (cf. point 2), criteria (dose, mass, material composition, fission products content) for handling and storage approaches have to be defined.

Technology and knowledge gaps: as discussed earlier, damaged fuel/debris characteristics.

Raised concerns: none.

Temporary on-site storage of structural debris

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Space restrictions may be a serious problem depending on debris volumes. Criteria for appropriate handling and storage approach to be defined considering materials characteristics (dose, mass, material composition).

Likelihood of success: similar concerns as the one expressed above for damaged fuel/fuel debris were expressed in case of large volumes of structural debris.

Technology and knowledge gaps: structural debris characterisation.

Raised concerns: none.

Temporary on-site storage of filters used in water decontamination process

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Dose and space restrictions if large volumes of contaminated water have to be treated. Criteria for appropriate handling and storage approach to be defined considering filters loading (dose, mass, fission products content). Waste may need to be stabilised (limit remobilisation).

Likelihood of success: the likelihood of success of the action is considered high even if more challenging, due to space restrictions considerations, when large volumes of contaminated water have to be treated. Filters loading should be controlled during the water decontamination process. Contamination captured in the filter may need to be fixed to minimise any remobilisation.

As for any nuclear waste management (cf. point 2), criteria for handling and storage approaches have to be defined.

Technology and knowledge gaps: none.

Raised concerns: none.

Material control and accountability

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Necessary level of detail in accounting to be defined depending on considered material, not the same as under normal operation. Adequate devices and methods for "routine" material control and accountability to be designed especially when large volumes of wastes have to be treated.

Likelihood of success: the likelihood of success of the action is considered high even if a large variety of wastes has to be treated. Adequate devices and methods for "routine" material control and accountability have to be developed based on defined criteria.

Technology and knowledge gaps: none.

Raised concerns: none.

SFP defuelling

Likelihood of success	Technology gaps	Challenging/open issues
High (loss of cooling)	Low	Complexity depending on extent of fuel damage in SFP, SFP building damage (if external to confinement) and on-site radiological situation (accessibility to SFP). Identification, characterisation, collection and handling of mechanically weakened fuel assemblies, damaged fuel, fuel debris (similar to fuel in reactor, cf. point 4). Damaged racks.

Likelihood of success: the action of SFP defuelling is considered to have a high likelihood of success when considering a loss of cooling accident with moderate fuel degradation (no loss of fuel assembly integrity). The situation would be much more challenging for a loss-of-coolant accident which would result in large fuel damage. Then technical challenges would be similar in nature to those associated to fuel retrieval in a damaged reactor, possibly with higher difficulties arising from the volumes of damaged fuel to recover, handle and store.

If the SFP is external to confinement, a temporary confinement may have to be established over the damaged SFP to limit radioactive releases prior to and during damaged fuel retrieval.

Technology and knowledge gaps: see reactor defuelling and open issues associated to LT SFP cooling.

Raised concerns: see above.

6) Long-term management

Radiological and security monitoring on-site

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Define reliable and secure sensors network for radiological and security monitoring on-site (dose, cameras, etc.) considering the plant damaged state. Define adequate instruments to follow effective clean-up, decontamination, fuel retrieval operations.

Likelihood of success: high, no raised concerns.

Technology and knowledge gaps: none.

Raised concerns: none.

Occupational health protection

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Reliable monitoring of dose map in intervention zones, hot spots detection and treatment in complex environment. Assess radiological risk and exposure associated to complex operations (clean-up, decontamination, defuelling, waste treatment and storage). Plan interventions to minimise radiological exposure. Monitor radiological exposure of workers.

Likelihood of success: the action is considered to have a high likelihood of success if proper dose monitoring, radiological risk evaluation (see below) and planning of interventions are made. Also, monitoring of radiological exposure of workers is an important aspect.

Technology and knowledge gaps: possibly, radiological risk assessment for complex operations.

Raised concerns: none.

Emergency equipment maintenance

Likelihood of success	Technology gaps	Challenging/open issues
High	Low	Assess reliability of safety important equipment on the LT. Equipment behaviour on LT in harsh conditions (dose, chemistry).

Likelihood of success: the action is considered to have a high likelihood of success if proper monitoring, repairing or replacing of equipment is ensured.

Technology and knowledge gaps: equipment behaviour on LT in harsh conditions (dose, chemistry).

Raised concerns: none.

Risk assessment related to LT actions

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Medium	Little experience in the assessment of risks in complex situations on the LT, hence success probability is not known. Risk assessment and mapping will change substantially during LT. Non-trivial development of risk assessment tools may be required for LT.

Likelihood of success: risk mapping and implementing in response specific procedures, countermeasures and organisation is a challenge for complex situations. Risk mapping should also follow plant damaged state changes (e.g. upon recovery actions) and consider LT actions which may have different scope (maintain plant safety with damaged fuel in reactor or in SFP, clean-up, decontamination, defuelling, waste treatment and storage).

Technology and knowledge gaps: many risk ranking methods exist (Section 3.4) but little experience in application to LT management, with the exception of recent application to Fukushima Daiichi NPP. Methods applicable to LTM should be developed.

Raised concerns: none.

External hazards assessment and countermeasures

Likelihood of success	Technology gaps	Challenging/open issues
Medium	Medium	Failure risk assessment for damaged or weakened structures and equipment contributing to LT plant safety (reduced margins). Uncertainties concerning external hazards. Assess margins recovery by countermeasures.

Likelihood of success: risk assessment for damaged or weakened plant and implementing specific countermeasures (e.g. structure reinforcement, redundancy of emergency equipment) is not straightforward. Characterisation of the damaged state and remaining margins to failure is a difficult task. Further, recovery of margins set at the design stage may not be possible despite countermeasures.

Technology and knowledge gaps: uncertainties remain on external hazards assessment. Difficulty to assess margins reduction without a proper knowledge of structure or equipment damaged state (difficult to attain). No tool to appreciate resistance of weakened structures and systems.

Raised concerns: none.

Table 3.5. **Maintain coolable configuration AIRT**

Action	Likelihood of success	Technology and knowledge gaps	Challenges and open issues
Inject water in-vessel	High	Low	Long-term performance of equipment uncertain. Back-up systems should be pre-planned
Inject water in containment	High	Low	Long-term performance of equipment uncertain. Back-up systems should be pre-planned
RPV pressure control	High	Low	Plants should be equipped with reliable pressure control systems
Maintain RPV integrity (for in-vessel melt retention strategy)	Medium	Medium	Knowledge of late-phase melt progression. Remaining strength of RPV cannot be assessed. Materials performance issues; embrittlement, corrosion
Containment pressure control	High	Low	Related to hydrogen control, discussed below
Containment hydrogen control	High	Low	Radiolysis needs to be considered
Confinement of fission products	Medium	Medium	Uncertainties regarding status of confinement, chemistry and remobilisation processes of fission products in the long term affecting releases
Minimise environmental release	Medium	Medium	Knowledge/technology gaps mainly concerning aqueous release.
Minimise likelihood of criticality	High	Medium	Risk of re-criticality sometimes difficult to assess
Maintain SFP cooling and minimise risk	High (loos of cooling) Low (loss-of-coolant)	Medium	Validated tools to calculate progression and consequences of SFP accident

Table 3.6. **Waste water, solid waste and effluent management AIRT**

Action	Likelihood of success	Technology and knowledge gaps	Challenges and open issues
Minimise and collect contaminated water	Medium	Low	If containment is leaking, it is a considerable challenge to handle contaminated water. Also, if cooling is not done on secondary side, large amounts of contaminated water is generated from cooling
Decontaminate collected waste water	High	Low	No technology available to separate tritium in very large volumes and very low concentrations; capacity may be a problem; water treatment facilities may not be readily available as well as characterisation techniques (not easily detectable nuclides)
Recirculate decontaminated water for core cooling	High	Low	Damaged systems/piping may pose a problem
Collect and store waste water	High	Low	The sheer volume of waste water may be problematic. Release criteria should be defined
Monitor and control fission products including tritium in water	High	Low	Monitoring techniques may not be readily available for all radionuclides Monitoring of some not easily detectable nuclides
Control and minimise effluent discharge	Low	Medium	Knowledge of containment status (leak paths) Technics/methods to recover the containment
Effluent characterisation	High	Low	Cf. challenges mentioned for fission products monitoring in waters
Collect, store solid wastes and monitor fission products content	High	Low	Monitoring of fission products content in highly active and complex solid structures Accessibility and handling of highly active solids (cf. defuelling)
Storage and disposal of filtered containment venting system water and filter	High	Low	Monitoring of fission products content in highly active solid and liquid wastes (see above) Accessibility and handling of highly active solids (see above)

Table 3.7. **Site clean-up and decontamination AIRT**

Action	Likelihood of success	Technology and knowledge gaps	Challenges and open issues
Plant site clean-up and sludge removal (outside of the reactor)	High	Low	If major releases have occurred, the likelihood of reaching green field status may be low
Surface decontamination	High (on-site) Medium (in damaged plant)	Low	Accessibility may be limited due to contamination, physical constraints, water coverage. Contamination may have diffused into structures (e.g. in concrete). Implementation of strippable or washable coatings in zones were high contamination is expected (e.g. in containment)
Debris and sludge characterisation	High	low	Robotics and remote measurement technology would be helpful. Composition, content of burnable substances, etc. are also of interest for transport and disposal. The sheer amount of debris may be a challenge
Monitor cooling pond (and sea) contamination	High	Low	None
Monitor and control groundwater contamination	Low	low	Specific technological solutions to limit groundwater contamination are highly situation/plant specific
Radiological monitoring	High	Low	Technology is available; robotics and remote measurement technology would be helpful
Characterisation of fission products distribution in contaminated spaces	Medium	Medium	Methods for source distribution analysis; improved techniques for hot spots detection and characterisation; difficulties to access highly contaminated areas; technology is available except for molten fuel location detection; codes cannot give this much detail

Table 3.8. **Defuelling of damaged reactors AIRT**

Action	Likelihood of success	Technology and knowledge gaps	Challenges and open issues
Detect damaged fuel/fuel debris locations and estimate masses (at start of defuelling)	Low	High	High challenge particularly with failed RPV and/or large fuel dispersion. No reliable information by imaging until access to damaged fuel is possible. Distant monitoring with sufficient resolution not yet developed. SA codes not sufficiently predictive. Required progress in technics and analytical tools scientifically challenging. Selection of appropriate defuelling options and techniques delicate at start of defuelling.
Identify state of degraded fuel	Low	High	Determining the composition, fission products content and mechanical behaviour of degraded fuel and debris without sampling and analysis of real material. Predictive analyses using thermodynamic calculation tools. Representative simulant materials testing. Transposition of knowledge from performed fuel degradation experiments. Ageing of damaged fuel.
Collection of damaged fuel/fuel debris and characterisation	Medium	Low	Specific remote control technologies may have to be developed depending on damaged fuel/debris characteristics. High dose, knowledge of debris characteristics.
Damaged fuel examination and characterisation for storage	Medium	Medium	Knowledge of damaged fuel characteristics should be obtained during fuel retrieval operation. Specific tools may have to be developed for characterisation related to criteria set for waste storage.
Minimise environmental releases	High	Low	Depending on damaged fuel distribution, locations and characteristics may be challenging to minimise releases during fuel retrieval operations. Development of cutting techniques limiting aerosol generation and dispersion considering damaged fuel characteristics.

Table 3.9. **Damaged fuel/fuel debris and radioactive waste disposal, SFP fuel removal AIRT**

Action	Likelihood of success	Technology and knowledge gaps	Challenges and open issues
Temporary on-site storage of damaged fuel/fuel debris	High	Low	Volume, dose and space restrictions on-site. More challenging for multi-units, SFP accidents with significant fuel damage, accidents with significant fuel dispersion. May necessitate large-scale constructions (temporary confinement before retrieval, installations for storage after retrieval). Define criteria for site location and/or storage containers. Define criteria for appropriate handling and storage.
Temporary on-site storage of structural debris	High	Low	Space restrictions may be a serious problem depending on debris volumes. Define criteria for appropriate handling and storage.
Temporary on-site storage of filters used in water decontamination	High	Low	Dose and space restrictions if large volumes of contaminated water have to be treated. Define criteria for appropriate handling and storage. Waste may need to be stabilised (limit remobilisation).
Material control and accountability	High	Low	Define necessary level of detail for control and account. Design adequate devices and methods for "routine" material control and accountability particularly when large volumes have to be treated.
SFP defuelling	High (loss of cooling)	Low	Complexity depending on extent of fuel damage in SFP, SFP building damage (if external to confinement) and on-site radiological situation. Identification, characterisation, collection and handling of mechanically weakened fuel assemblies, damaged fuel, fuel debris. Damaged racks.

Table 3.10. **Long-term management AIRT**

Action	Likelihood of success	Technology and knowledge gaps	Challenges and open issues
Radiological and security monitoring on-site	High	Low	Define reliable and secure sensors network for radiological and security monitoring on-site (dose, cameras, etc.) considering the plant damaged state. Define adequate instruments to follow effective clean-up, decontamination, fuel retrieval operations.
Occupational health protection	High	Low	Reliable monitoring of dose map in intervention zones. Assess radiological risk and exposure associated to complex operations (clean-up, decontamination, defuelling, waste treatment and storage). Plan interventions to minimise radiological exposure. Monitor radiological exposure of workers.
Emergency equipment maintenance	High	Low	Assess reliability of safety important equipment on the LT. Equipment behaviour on LT in harsh conditions.
Risk assessment associated to LT actions	Medium	Medium	Assessment of risks in complex situations on the LT. Risk assessment and mapping will change substantially during LT. Non-trivial development of risk assessment tools may be required for LT.
External hazards assessment and countermeasures	Medium	Medium	Failure risk assessment for damaged or weakened structures and equipment contributing to LT plant safety. Uncertainties concerning external hazards. Assess margins recovery by countermeasures.

Conclusions derived from the AIRT exercise results

The AIRT should serve as a filtering process looking at identifying the most critical challenges and open issues related to the main actions carried out during the long term of the severe accident. The priority taken by such challenges and issues should be in line with the importance of the actions as hierarchically ranked in the exercise.

In order to limit the recommendations (Chapter 5) naturally aimed at closing the identified issues and gaps by orienting future efforts, a qualitative threshold should be established. As stated above in Section 3.5, such barrier has been placed at the "medium" rank level, i.e. challenges and open issues will be identified only coming from those actions assessed as "medium" or "low" in terms of the likelihood of success, and/or as "high" or "medium" in terms of gaps. And within these actions, challenges and open issues linked to those featuring L/H combinations should deserve a distinct, more careful treatment.

From the 6 LTM-goal-oriented tables, there are only 3 out of 42 actions (7%) featuring an L/H combination, all of them under the defuelling goal:

- detect fuel debris location and estimate debris masses;
- identify the state of the degraded fuel;
- design and implement defuelling alternatives.

However, most of the actions present at least one M value whether in the likelihood of success or gap side, in particular, 21 actions from the 6 tables, meaning 45% of the entire list:

- maintain RPV integrity (medium in technology/knowledge gaps);
- confinement of fission products (medium in technology/knowledge gaps);
- minimise environmental release (medium in both ranks);
- minimise likelihood of re-criticality (medium in technology/knowledge gaps);
- maintain SFP cooling and minimise risk (medium in technology/knowledge gaps);

- minimise and collect contaminated water (medium in likelihood of success; high in technology gaps);
- decontaminate collected waste water (medium in both ranks);
- control and minimise effluent discharge (low in likelihood of success; medium in technology/knowledge gaps);
- collect, store solid wastes and monitor fission products contents (medium in likelihood of success);
- plant site clean-up and sludge removal (medium in technology/knowledge gaps);
- surface decontamination (medium in technology/knowledge gaps);
- debris and sludge characterisation (medium in both ranks);
- monitor and control groundwater contamination (low in likelihood of success; medium in technology/knowledge gaps);
- characterisation of fission products distribution in contaminated spaces (medium in both ranks);
- collection of damaged fuel/fuel debris characterisation (medium in both ranks);
- damaged fuel examination and characterisation for storage (medium in technology/knowledge gaps);
- minimise environmental releases during the defuelling process (medium in both ranks);
- risk assessment associated to LT actions (medium in both ranks);
- external hazards assessment and countermeasures (medium in both ranks).

It is worth noting that the conducting efforts should be oriented according to the type of identified issue/gap. For instance, medium or low likelihood ranks can just derive from the lack of dedicated arrangements on-site. If this were the case, efforts on how addressing the issue should not be oriented towards further research and development but on providing recommendations to the plants to accommodate the necessary measures allowing the action to be suitably performed in place. Therefore, it will be fundamental to link the challenges/ open issues with appropriate recommendations in conducting future activities aimed at closing that issue/gap. Among others, Chapter 5 will address these recommendations taking as one of the sources of information the AIRT results coming from the 24 actions ranking medium/low in likelihood of success and/or high/medium in technology gaps as listed above and upon tracing back the underlying challenge/open issue.

Chapter 4. Summary of challenges and main issues for long-term management

Significant challenges, open issues and knowledge and technological gaps for the long-term management (LTM) of severe accidents synthetised in this chapter were identified through the feedback of the Three Mile Island unit 2 (TMI-2), Chernobyl and Fukushima Daiichi accidents (Section 2.1), through the review of the status of post-accident long-term management and actions (LTMA) in NEA member countries (Section 2.2) and through the various approaches to LTMA (Chapter 3).

Though the TMI-2, Chernobyl and Fukushima Daiichi accidents posed tremendous challenges for LTM, each accident was different in nature and had different consequences. Thus challenges and issues may be different for each of them. TMI-2 LTM can be considered as completed as the fuel has been transferred from the site to long-term storage facilities whereas Chernobyl and Fukushima Daiichi LTM are ongoing with fuel retrieval to come. Thus for these two accidents the learnings for LTM will surely evolve in the years to come.

In comparison to the TMI-2 accident where the reactor vessel, the primary circuit, and the confinement integrity were maintained after the accident, other challenges arose for the LTM of the Chernobyl and Fukushima Daiichi accidents. This was due to highly damaged infrastructures caused by explosion and fires at Chernobyl, and by earthquake, tsunami, explosion, and fires at Fukushima Daiichi, loss of confinement due to explosion at Chernobyl and mostly due to containment over-pressurisation at Fukushima Daiichi and consequent large release of contamination. Among these challenges, extensive site cleaning and construction works had to be performed to reduce on-site radiation levels, facilitate damaged plant accessibility and protect the environment (hydrosphere, atmosphere) from further radioactive releases (e.g. sarcophagus at Chernobyl, sea and land walls, groundwater bypass systems at Fukushima Daiichi). These extensive works were largely specific to the plant damaged state and site, particularly works to protect the hydrosphere.

Though the confinement had been maintained at TMI-2, the accident posed significant LTM challenges with unprecedented actions, organisations, and technical means (infrastructures, systems, equipment and instrumentation) to recover and treat highly contaminated waters, decontaminate highly contaminated buildings (in particular the reactor building where most of the radioactivity was confined), defuel a severely damaged reactor core and manage transport and storage of unique wastes.

In the sections below, the main challenges and issues that were highlighted in the TMI-2, Chernobyl and Fukushima Daiichi accident feedbacks, in the information provided by utilities and safety organisations through a questionnaire as described in Section 2.2, and through the plant damaged state (PDS) classification and action identification ranking table (AIRT) exercises are summarised.

Regulatory and organisational aspects

The feedback from past accidents showed that unique regulatory and licensing requirements had to be developed for LTM at TMI-2 and Fukushima Daiichi[1] with necessary "flexibility" compared to existing regulation. New regulatory practices had to be applied as changes in the

1. Little feedback was provided on these aspects for Chernobyl LTM.

facility's post-accident mode of operations required unique regulatory and licensing actions. In order to properly reflect evolving plant status, the regulators issued orders, modified those orders, approved licence amendment requests, and granted relief from certain regulatory requirements. Exemptions were necessary because of the plant's damaged configuration and changing status during clean-up. This had to be done for very different LTMA with sometimes limited knowledge of plant status and risks for their implementation, with sometimes use of unique systems and equipment and involving objects difficult to fit in regulatory classification (e.g. degraded fuel, radioactive wastes).

Not only the regulatory requirements, but also the organisational structure and functions of the utilities and their support organisations changed during the long-term management of the accidents. Various organisations formed working groups to provide guidance on addressing specific issues, problems, and research activities. Independent oversight groups reviewed, monitored, and advised on the overall direction of recovery and clean-up plans and activities. Also, both at TMI-2 and Fukushima Daiichi, international collaborative activities were established in support of the long-term management and decommissioning of the plants.

Concerning plant staff, long-term management programmes are intense and staff working conditions should be considered in long-term recovery activities and staff turnover should be planned where appropriate.

Relation to the public during long-term management

Increased public scrutiny after an accident was reported as an important issue both for the TMI-2 and the Fukushima Daiichi accidents where organisations dedicated to relations to the public were established and public relations plans were made as described in Section 2.1. Even with extensive efforts, gaining public trust was proven to be a significant challenge. In many cases it was more difficult to obtain public trust and acceptance than to find a technical solution.

Risk evaluation for long-term management

For all accidents, it has been reported that risk assessment for LTM is a challenge due to lack of knowledge on plant damaged state when implementing some LTMA, e.g. damaged fuel distribution and characteristics when starting fuel retrieval at TMI-2 and Fukushima Daiichi, and its evolution during LTM. The need of continued risk assessment during LTM implementing new knowledge on plant damaged state for risk-informed LTM is highlighted, particularly for Fukushima Daiichi LTM. Risk assessment is further complicated by lack of experience on risk assessment in the long-term management of severe accidents, and the lack of validated methods to apply to this task. Some approaches to the risk assessment have been discussed in Chapter 3 and, in connection with the Fukushima Daiichi experience, in Section 2.1. It was recognised that further work is needed to develop methods capable to effectively guide long-term management.

More specific aspects have been highlighted from LTM feedback as well as analyses presented in Chapter 3:

- LT reliability of infrastructures and of some important safety systems, e.g. failure of weakened infrastructures related to ageing or external events may result in a new unstable state. Examples of this are the non-stabilised first sarcophagus built in Chernobyl to protect the damaged reactor, and the reactor buildings of units 1, 3 and 4 at Fukushima Daiichi before consolidation works.
- Situations posing increasing challenges with time should be particularly assessed, e.g. ageing of the first sarcophagus at Chernobyl, waste accumulation at Fukushima Daiichi, leaching and ageing of damaged fuel material and structures at Chernobyl and Fukushima Daiichi.
- Assessment of risk of criticality.

- The possibility of generation of combustible gases.
- The risk of release of radioactive compounds from damaged fuel due to leaching and ageing, and release of fission products from surfaces and water volumes due to ageing as well as de-commissioning activities.

Both at Chernobyl and at Fukushima Daiichi, the present state of the fuel-containing materials is not sufficiently known which creates nuclear and radiation risks for defuelling works.

Radioprotection in long-term management

One of the primary objectives for LTM is to improve working conditions by reducing dose rates on-site by simultaneously avoiding further release of activity or increase of the dose rates. The challenges come from the unknown status of the plant and the unknown distribution of the radionuclides. The difficulty to acquire sufficiently accurate dose rate mapping that could guide radioprotection implementation for LTMA and the search for improved data acquisition and analyses methods were highlighted for the past accidents and identified as a knowledge and technological gap in the AIRT exercise.

The work is challenging due to high dose rates in the contaminated areas, and the potentially severely damaged structures. The dose rate reduction is done, as far as feasible, in the damaged plant(s) by cleaning, fixing or removing contamination and by implementing protections wherever necessary. Different approaches have been implemented effectively easing progressively interventions in more and more areas in the damaged plants. In many cases, dose reduction by shielding has proven to be more effective than decontamination. The use of robotics to perform monitoring and decontamination is invaluable in reducing radiation exposure to workers and in allowing operations in areas not accessible to humans, as evidenced by the feedback from past accidents as well as highlighted in the AIRT exercise.

To manage radioprotection, specific radioprotection organisations were established at TMI-2 and Fukushima Daiichi. As low as reasonably achievable (ALARA) approach was set as an objective. Many programmes and activities were established to help improve radiation protection practices. New approaches were needed in a number of basic worker protection and dose reduction areas, including protective clothing, respiratory protection, dosimetry, radiation field and contamination characterisation, exposure-tracking systems, dose reduction planning, procedures, and training.

It should be emphasised that at Chernobyl and Fukushima Daiichi, damaged plants are still facing considerable radioprotection challenges related to future defuelling and decommissioning actions. A significant challenge will be to assess and mitigate the risks of contamination dispersion related to these actions. Also, difficult configurations in terms of accessibility and doses may pose radioprotection challenges.

Coolable configuration and confinement of radioactivity

The three past accidents in TMI-2, Chernobyl and Fukushima Daiichi had very different accident progression and end state of the reactors. Therefore, we give a short summary of each.

In TMI-2, the reactor pressure vessel was intact and this enabled coolant water injection into the reactor even though the pressure control did not function and non-condensable gases needed to be released not to block the heat removal. The state of the core was not known until first visual observations were made and it was seen that the core was severely damaged. As the state of the core was not known, a high boron concentration was maintained in the coolant to ensure that the core would not become critical.

The main challenges to the confinement of radioactivity in the long term were the large quantities of contaminated water in the reactor building, and the release of fission gases from the damaged fuel. The fission gases caused airborne contamination inside the auxiliary building and measures were taken to remove them. A significant risk of release of radioactivity to the environment in the long term was related to the large quantities of contaminated water in the reactor building. Water management was one of the most challenging activities in the long-term management of TMI-2.

In Chernobyl, the explosion and fire destroyed the structures designed to cool the reactor core and to confine the radioactivity. A sarcophagus was erected within six months of the accident to limit the release of radioactivity mainly to the atmosphere. However, the sarcophagus was not entirely leak tight. Cooling of the corium inside the sarcophagus was carried out by installing sprays which also reduced the amount of airborne radioactive aerosols formed due to ageing of the corium materials. Contamination release through water pathways was controlled by building a plate under the reactor, and by several measures to limit the flow of water to the nearby rivers.

In Fukushima Daiichi, the reactor pressure vessels and containments in all three damaged units have been observed to be compromised thereby providing a path for release of radioactive compounds. A closed-loop cooling has been established to control the releases with water. All three containments are filled with water even if to a different level, and a large fraction of the contamination in the containments is expected to be in the water thereby controlling the air releases. Continuous coolant injection keeps the temperatures in the reactor pressure vessels and containments at low levels.

Even though very different in progression and end state, all accidents described here had similar challenges related to cooling the reactor core and confining radioactivity. The AIRT exercise especially highlighted the challenges related to the unknown state of the reactor pressure vessel (RPV) in the long run and material performance issues, as well as to confinement of aqueous releases and uncertainties regarding remobilisation of radioactivity as elaborated below.

First and foremost, the state of the core was not known when long-term management actions had to be designed and decided. This indicates that both analytical and measurement methods are incapable of determining the accident progression and its end state accurately enough as also emphasised by the AIRT process. Only after visual observations and samplings of the active materials, it was possible to determine the state of the core in TMI-2. In Chernobyl and Fukushima Daiichi, samplings have been carried out and they have given indications of the extent of the damage, but with the lack of instrumentation capable of working under the conditions prevailing in the reactors, there are still uncertainties concerning the state of the core and distribution of the fission products.

Secondly, there is not much knowledge about the processes that take place in the long term under humid, high irradiation, possibly high temperature atmosphere. Structures which have kept the coolant flowing and confined the radioactivity may deteriorate with time. This means that long-term accident management measures need to be implemented taking into account that the status of the plant is changing, but it is difficult to predict these changes.

And third, it was not possible to determine which systems in the plants were operable and could be used in the long-term management of the accidents. Therefore, long-term management actions had to be planned using systems without always knowing if those systems would work and without instrumentation to support these actions with measurements.

It should be emphasised that these issues and challenges are not accident progression specific and therefore would probably exist for most severe accidents as highlighted by the AIRT process in Chapter 3.

Development and use of unique systems, equipment and instrumentation

Unique systems, equipment and instrumentation had to be developed for LTM of the three accidents. Important cited aspects are not always common to the three accidents:

- robotic or remote control means for decontamination, for investigations in harsh environment (imaging, monitoring, sampling) and for defuelling;

- sampling systems for contaminated waters and active debris and related analytical means;
- instrumentation for detailed and accurate radiation monitoring;
- methods for determining the distribution of degraded fuel, e.g. muon tomography at Fukushima Daiichi;
- systems and instrumentation for monitoring sub-criticality, hydrogen and radioactive dust;
- liquid waste treatment facilities and related analytical means;
- transport canisters for solid wastes;
- temporary radioactive waste storage facilities for low, medium and highly active liquid and solid wastes.

LTM of the three accidents highlights the importance of containment media and analyses, of monitoring critical data (to avoid reaching a new unstable state) and instrumentation providing reliable information for a risk-informed LTM. The monitoring of sub-criticality remains a challenge, even though the detection of re-criticality events is possible. The importance of developing further remote or robotic techniques of investigations to guide LTMA should be also highlighted. These technological gaps were also pointed at in the AIRT exercise which indicated that available monitoring and robotic technologies are not always compatible with conditions in the vessel and/or containment due to high radiation and high temperature, and that there are improvement needs for imaging and distant monitoring in high radiation zone.

Contaminated water treatment

Common to the three past accidents was the management of large amounts of highly contaminated liquids. This required the design and construction of treatment facilities to eliminate, through filtering systems, Cs, Sr, and other radionuclides. At the same time, the water treatment facilities generated highly active solid wastes in the form of filters and sludge. Tritium cannot be removed from the liquid using standard liquid waste treatment. In TMI-2, water contaminated by tritium was evaporated. In this way any contamination of nearby river by tritium was avoided and release of residual activity of other radionuclides remaining in the water was avoided. Due to the use of sea water during the emergency phase of the accident at Fukushima Daiichi for cooling of damaged reactors, specific separation processes had to be implemented to eliminate chlorine from contaminated waters. Information about water treatment systems for TMI-2 and Fukushima Daiichi are provided earlier in this report. Management of large amounts of highly contaminated liquids would probably be a challenge for most severe accidents as highlighted by the PDS classification (Section 3.3) and AIRT (Section 3.5) exercises.

Leaching of the fuel and fission products would also pose for most severe accidents a specific challenge to the water treatment systems. At Fukushima Daiichi, the reactors are continuously cooled with water, and some fission products have been found to be transferred to the coolant water. Due to groundwater intake in the damaged reactors, contaminated waters are posing specific problems at Fukushima Daiichi with accumulation of very large volumes that have to be stored on-site. A lot of efforts have been implemented to reduce the produced volumes: implementation of close loop cooling, of sea and land walls, and of groundwater bypass.

At Chernobyl, water in the sarcophagus presents increasing Pu concentrations which may be a sign of leaching and ageing effects. The water originates from condensation inside the sarcophagus, sprays used to limit dust dispersion, and leakage of rain water into the sarcophagus. Extensive efforts were made over the years to successfully limit the release of radioactivity to the groundwater and the surrounding rivers.

Site clean-up, de-contamination and waste management

In past accidents, different strategies have been implemented to reduce as much as feasible the radioactive inventories and risks of further radioactive release.

For instance, at TMI-2 the approach was progressive with first recovery and treatment of the highly contaminated waters to give access to the plant buildings, followed by decontamination of the auxiliary and the reactor building to give access to RPV and damaged fuel for defuelling.

At Fukushima Daiichi, many aspects had to be treated in parallel such as recovery and treatment of large volumes of contaminated waters, cleaning of rubbles from buildings damaged by the explosions, decontamination of the reactor building above the refuelling floor and of the refuelling floor, fuel retrieval from spent fuel pools (SFPs), etc. Access to primary containments remains a challenge in the end of 2017.

At Chernobyl, the prime objectives were to re-establish a confinement by building the first sarcophagus within six months, and to ensure its safety. There too, access into rooms containing damaged fuel and corium remains a challenge in the end of 2017. Even though samples have been taken of the fuel-containing materials, de-fuelling has not started yet.

As shown by the Fukushima Daiichi feedback and through approaches to LTM described in Chapter 3, risks identification and ranking could efficiently guide LTM strategies, but related tools and methods require development and validation.

At TMI-2 and Fukushima Daiichi, decontamination of highly contaminated surfaces was done using many different techniques (chemical or mechanical treatment) and sometimes by robotic means. Cleaning and decontamination operations were reported to be challenging especially where contamination had penetrated deeply into the contaminated material, in particular in concrete and in some areas with low accessibility. Then fixing contamination, e.g. by spraying anti-scattering agents, or implementing protections and shielding where necessary are alternatives that have been used to decrease radiation level and provide safe working conditions. This should be a challenge for most severe accidents as highlighted in Chapter 3.

In common to the three accidents, radioactive wastes had unique characteristics not fitting regular waste classifications and technologies for reprocessing, transport and disposal. Therefore, specific equipment, containers and installations for waste treatment and temporary waste storage had to be constructed within limited amount of time to manage the large quantities of radioactive waste. Even though AIRT process estimated a high likelihood of success to these actions, it identified challenges due to space restrictions, time management, and selection of criteria for a suitable treatment method due to the unknown characteristics of the waste.

For TMI-2, special design was developed for fuel canisters, which were then suitable for the transfer and storage using filling material to stabilise the fuel-containing solid waste.

Solutions for temporary and final waste storage have yet to be established for Chernobyl and Fukushima Daiichi.

Fuel retrieval

Regarding the fuel retrieval from the damaged reactors, all three past accidents have very distinctive characteristics. In TMI-2, the reactor pressure vessel remained undamaged, and fuel retrieval could be carried out from the top of the reactor vessel. In Fukushima Daiichi, the reactor pressure vessels as well as primary containments in all three damaged units are compromised, and some fraction of the corium is expected to be on the containment floor with possible molten corium-concrete interaction (MCCI). Therefore, fuel retrieval will be more complicated. In Chernobyl, the explosion destroyed the confinement, and the fuel is spread to several rooms both vertically and laterally.

At TMI-2, the fuel retrieval strategy and equipment had to be revised when information was gained on damaged fuel status after RPV opening and investigations in the RPV. The fuel retrieval was completed with specific equipment (defuelling platform) for which exercises were performed beforehand to limit workers exposure.

At Chernobyl, the fuel retrieval strategy is yet to be designed and planned. The degraded fuel material distribution and characteristics were studied in detail but it is recognised that ageing, mostly through leaching in water and interactions with atmospheric gases, may affect the integrity of the damaged fuel with possible fine dust formation on the long term. The long-term behaviour was investigated but the knowledge is not sufficient to assess the risk of fine dust formation. The new sarcophagus was designed to avoid release of radioactivity to the environment during fuel retrieval actions.

At Fukushima Daiichi, the fuel retrieval is planned to be started around 2021. It is emphasised that Fukushima Daiichi presents an added complexity with respect to TMI-2 since it is a three-unit accident with different accident progression in each unit. Consequently, the optimal fuel retrieval strategy may be different for each unit. It is noted that presently, some uncertainty still exists regarding the amounts of material in the reactor pressure vessel and in the containment, as well as regarding the extent of MCCI in the three units despite investigations conducted so far (analytical investigations, muon tomography, robotic investigations in the different units). Important efforts are planned from 2018 to increase knowledge on degraded fuel for preparation of plans for fuel retrieval operations.

As highlighted in Chapter 3, fuel retrieval from damaged reactors or SFPs after a severe accident should remain a tremendous challenge as long as there are remaining knowledge and technology gaps which prevent obtaining information that could effectively guide defuelling, before direct observations and characterisation of damaged fuel are possible. In addition, further development of remote control technologies was emphasised also for these actions.

Chapter 5. Recommendations for enhancing long-term post-accident management and for future research

This chapter presents recommendations for improving the long-term management (LTM), arranging the recommendations upon whether they address general, cross-cutting issues covering all aspects of LTM, or specific LTM actions.

Recommendations are related either to necessary knowledge development and consolidation to enhance LTM and are then addressed to research organisations, safety organisations[1] and industry (operators and designers), or to development of accident management provisions and they are then addressed more specifically to plant designers and operators.

Organisations such as the Nuclear Energy Agency (NEA), the European Commission (EC) and the International Atomic Energy Agency (IAEA) could foster international initiatives aiming at knowledge development and capitalisation.

It should be emphasised that proposed recommendations are based on LTM feedback for past major accidents (Section 2.1 and Chapter 4) and on risks, challenges, issues and knowledge and technological gaps identified through discussed approaches to LTM (Chapters 3 and 4).

5.1. Recommendations related to knowledge consolidation for cross-cutting issues

Calculation tools and methods for severe accident analysis in reactor and spent fuel pool

Consolidation of reactor and spent fuel pool (SFP) severe accidents knowledge base and calculation tools (e.g. severe accident (SA) codes, thermochemical databases) should be pursued in the future to enhance capabilities to predict i) the effect of mitigation measures used to reach a stabilised state and ii) the stabilised plant damaged state following an accident, e.g. distribution and properties of radioactive material in- and ex-vessel in a damaged reactor.

Methods to validate reactor and SFP accident mitigation strategies to reach a plant stabilised state, including strategies using non-conventional approaches and equipment should be consolidated.

These research efforts should be targeted at providing information, data and methods to enhance validation of reactor and SFP accident mitigation strategies, addressing in particular scaling issues and uncertainties for main reactor designs.

LTM of past SAs, including the ongoing LTM for the Fukushima Daiichi accident, has evidenced that the existing knowledge base and calculation tools for reactor and SFP SA are not sufficiently developed and validated to provide some critical information on the stabilised plant damaged state after an accident that would support LTM implementation (e.g. containment status, distribution and properties of radioactive material in the damaged plant). Though additional information and data should be obtained and verified after reaching a stabilised state, LTM would greatly benefit from more predictive tools.

1. Safety authorities and their technical support organisations.

Further, positive and negative effects of some accident mitigation measures, e.g. on components, equipment, systems and structures, remain difficult to assess with existing knowledge. Failure or success of these mitigation measures will in great part determine the plant damaged state after stabilisation and risks and challenges for LTM.

This is particularly true for example for strategies and actions involving water injections in-vessel, ex-vessel or in SFP in view of recovering the cooling of the degraded fuel and strategies involving pressure control of the containment (e.g. venting) as these actions, if improperly implemented, generate risks to lose the confinement of radioactive material and of increased radioactive releases in the LT. For instance, water injections can either result in positive (e.g. effective degraded fuel cooling in or ex-vessel, in SFP) or negative effects when applied inadequately.

The assessment of confinement failure risks and associated releases is part of SA analysis. However, an accident with failure of the confinement will obviously lead, due to contamination by radioactive releases on-site, to a challenging LTM with complex recovery actions. Even after the plant has been stabilised, it may be a continuing challenge to control and limit further releases particularly through contaminated water transfers to the environment and avoid groundwater contamination. As learnt from the Fukushima Daiichi accident, this may lead to the need to implement highly technically complex and costly measures to limit radioactive liquids releases and manage large amount of liquid wastes in the long term.

After the Fukushima Daiichi accident, significant efforts have already been internationally undertaken to enhance some SA mitigation strategies. However, validation of these strategies should still be examined with:

- full consideration of past accidents knowledge, including knowledge that will be obtained through post-Fukushima Daiichi accident projects (e.g. NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station, PreADES, TCOFF and ARC-F and future post-Fukushima Daiichi LT projects);
- full consideration of knowledge obtained through major international research programmes aiming at severe accident management guideline (SAMG) validation (e.g. the EC H2020 In-vessel Melt Retention [IVMR] project);
- increased knowledge and feedback on implementation of non-conventional approaches and equipment;
- further consideration of site and plant specificities and expected resources in challenging situations.

Additional research efforts that are being conducted within NEA and EC initiatives for the consolidation of SA mitigation strategies (e.g. in-vessel or ex-vessel melt retention, and radioactive release mitigation) and of SFP accident mitigation should be supported. Such consolidation, implementing in particular knowledge gained through LTM of the Fukushima Daiichi accident, will necessarily enhance severe accident management including for LT phases.

Status of equipment, systems and structures in the long term of a severe accident

Knowledge of status of equipment, systems, including passive ones, components and structures in the long term of an SA should be consolidated with emphasis on those that are expected to contribute maintaining a stabilised state on the LT.

Development of harmonised methods should be fostered at the international level to assess response and reliability of systems, including passive ones, equipment components and structures during an SA considering LTM.

With recognition that progressing in the field may be a complex task due to the variability of reactor designs and of their systems and equipment, shared common knowledge base and approaches would be highly valuable for safety assessment processes of nuclear power plants (NPPs). Such an action could also result in proposing further industrial developments for systems and equipment better qualified to prototypic SA conditions.

In particular, the capacity to assess reliability of cooling systems, including non-conventional ones, and status of confinement (leakages paths and rates and their evolution) on the LT is of critical importance for LTM.

Uncertainties around the containment status are a major issue, particularly when containment pressure and temperature loads have resulted in containment failure and significant radioactive releases to the environment. With respect to knowledge of the confinement leak tightness, further testing and characterisation for prototypic SA conditions of flanges and seals used to isolate containment openings, of concrete walls, of liners and coatings would certainly be valuable to determine most probable releases paths to the environment and where one can expect main radioactivity deposition in the damaged plant. The ageing of structures and equipment could as well be considered in such investigations. Materials, conditions and configurations to be tested could however be very containment design-specific and defining generic approaches to gain knowledge and develop methods to assess containment leakages and their evolution with time should be undertaken as recommended.

Severe accident long-term phenomena

International sharing and capitalisation of knowledge on phenomena that could affect LTM should be fostered to define appropriate future research on such phenomena.

Appropriate research should be conducted to acquire sufficient knowledge for LTM and related risk evaluation for re-criticality, hydrogen production, phenomena that could induce mechanical failure of safety important components (with consideration of cliff-edge effects) such as corrosion-erosion reactions, sediment formation and clogging of recirculation loops, solidified corium and debris leaching, ageing and fuel dusting and dispersal (in particular during debris retrieval operations).

Such actions should benefit from knowledge sharing between various communities (SA, ageing and waste management experts).

Further development of knowledge on some of these phenomena through investigations on real samples from passed accident should be supported, e.g. debris leaching and ageing from Chernobyl accident (some testing may be performed in the NEA TCOFF project). These investigations should also provide important knowledge for Fukushima Daiichi LTM.

LTM of an SA after damaged plant stabilisation could certainly be improved with better knowledge of long-term processes affecting systems, equipment and structures (e.g. damaged or weakened reactor pressure vessel [RPV], containment, SFP) but also radioactive products deposits and solidified corium and fuel debris. This is of special importance for risk evaluation related to LTM actions (cf. next section), including recovery and decommissioning actions. It has already been identified that knowledge of effects of corrosion and radiolysis reactions and related hydrogen production, of long-term radiation-induced damages, of fuel debris leaching and exposition to humidity and of radiobiological processes is scarce. From Chernobyl observations on lava ageing, it can be expected for instance that LT radiation-induced damages, leaching and/or exposition to humidity and radiobiological processes may affect solidified corium and fuel debris mechanical properties and their retrieval.

Knowledge sharing and capitalisation on long-term phenomena could be partly tackled in short-term projects which will be launched to prepare the decommissioning of the Fukushima Daiichi damaged reactors (e.g. NEA PreADES, TCOFF and ARC-F projects) with the objective to identify additional research that should be undertaken in the field and in which frame. From now on however, some support to further investigations on Chernobyl samples could be organised.

Methods or systems for risk assessment for long-term management

> Opportunities to launch at the international level a joint effort to progress in methods and/or expert systems for risk evaluations for LTM and LTM optimisation should be fostered.

Such an action would certainly benefit from a common work between risk and safety analysts from safety organisations and operators, taking full advantage of past accidents LTM feedback.

Expected results are methods and criteria for LTM optimisation in terms of safety to reach site remediation.

Conducting a risk-informed LTM of a damaged plant is necessary but challenging at this point since, as earlier pointed, i) knowledge of the stabilised plant damaged state and remaining safety margins for critical components is limited as well as their evolution with time and ii) knowledge and existing analytical tools are not developed to assess effects of long-term phenomena. In addition, methods and tools for risk mapping and ranking for LTM, tracking evolution of the stabilised plant damaged state and considering substantially different objectives of recovery actions (e.g. maintaining a stabilised state, managing wastes, cleaning and decontaminating accessible areas, retrieving fuel from SFP or from damaged reactors, etc.) and nature of risks are not well established. Possible approaches have been discussed in this report but they should be reviewed and developed further for concretely supporting long-term management and actions (LTMA) implementation.

In addition to the development of methods for risk assessment, there could be some interest to develop methods and criteria for LTMA optimisation in reaching site remediation.

5.2. Recommendations related to provisions development for cross-cutting issues

Enhance monitoring of plant damaged state and of long-term management actions

> Further joint investigations to better define critical data and instruments (existing or contemplated) that would effectively guide management of all phases of an SA including LTM and to promote the development of innovative approaches and technics should be fostered.
>
> This should include, more specifically for LTM, robotic means and remote technologies. Enhanced technologies for direct or distant imaging and more accurate radioactivity and dose mapping in harsh environment would strongly support LTMA implementation.
>
> Provisions to enhance monitoring of both short-term SA progression and related releases and LTM should be developed.
>
> These should in particular include instrumentation dedicated to provide information on degraded fuel/debris distribution in-vessel and ex-vessel (at least detection of vessel rupture during the accident), to provide information on containment leakage paths, to monitor re-criticality risks for various degraded fuel/debris configurations, to monitor combustible gas contents in premises where combustion could affect plant safety, to monitor atmospheric and liquid radioactive releases and radioactive products deposits in the damaged plant.

Large efforts have been undertaken in the past on instrumentation for normal operation and accidents in the emergency operating procedures domain but less on SA instrumentation. In the SA field, particularly after the Fukushima Daiichi accident, the focus of international attention was on consolidation of existing techniques and systems that support implementation of mitigation strategies on the short term, looking particularly at improved methods for better assessing their capability for SA conditions.

While this is of importance, less effort seems to have been undertaken internationally i) to identify if major possible monitoring improvements that would efficiently support comprehensive management strategies for all phases of an SA in a reactor or in an SFP – including emergency planning and LTM of damaged plants – are possible, ii) to identify available (including simple, robust and remote technologies), under development or promising new technologies that could provide such major improvements, and iii) to conduct research that would help better assessing critical instruments (existing or contemplated) performance in SA and promote development of innovative approaches and technics.

It should be emphasised that the research in this field should benefit from technologies both from the nuclear and the non-nuclear fields and should consider the development of fixed and mobile systems and their protection, including remote technologies and of robotically operated systems.

The following actions would help defining a well-targeted research programme to better assess critical instruments performance in SA and promote development of innovative approaches and technics:

1. Review of critical information for severe accident management, emergency planning, LTM (up to decommissioning after an accident) with required criteria (range, accuracy, response and mission time) – based on approaches developed or under development for NPPs worldwide – that would efficiently support comprehensive management strategies (both for reactor and SFP accidents). With consideration of 2), identification of major potential improvements (additional supportive critical information and relevant measuring technics) for different rector designs.
2. Review of already implemented instruments (including design basis accident instruments), their expected performance in SA (level of qualification, reading validity, margin to failure) and their possible use in severe accident management, emergency planning and LTM approaches for NPPs. And also identification of additional needed research, if any, to better assess their performance.
3. Identification of available technologies and/or technologies under development and/or innovative technologies that would provide these major improvements, considering possible implementations, protections and expected environmental conditions and covering all accident phases.

LTM could largely be improved with a reliable instrumentation providing pertinent information to characterise and monitor the stabilised plant damaged state and its evolution with recovery actions, to monitor the performance of systems and equipment – including back-fitted and non-conventional ones – involved in maintaining a stabilised state on the LT and to support the optimisation of LTM actions and their risk monitoring, including for clean-up, decontamination, fuel retrieval and decommissioning operations. Identified critical instrumentation is earlier listed with the recommendations.

Upgrades of components, equipment, systems and structures

Provisions to enhance reliability and availability of systems, equipment, components and structures should be implemented, including for LTM. Replacement, repairing and maintenance feasibility wherever contemplated should be assessed.

Provisions for post-accident samplings (contaminated liquids and atmosphere) in the damaged plant, retrieval, transfer and analyses should be prepared.

Numerous equipment and systems upgrades have already been proposed or implemented to various extents by operators in NPPs worldwide after the Fukushima Daiichi accident as a result of international stress tests to reinforce severe accident management. Some of these upgrades should be beneficial for LTM, in particular those implemented for strengthening mitigating systems as e.g. mobile equipment in common protected storage in different countries.

Some general guidance for upgrades – including for LTM – has already been provided in international documents (see for instance ENSREG [2012] and NEA [2014a]):

- Components, systems and equipment should remain operable to maintain a stabilised state for an extended period (several months or more) or they should be made easily replaceable. They should also be resistant or protected against on-site and off-site hazards. This concerns both fixed and mobile systems and equipment including passive systems.
- Provisions should be made to maintain mitigation measures for long-term phases of an accident. Due to the contamination, it may be difficult to repair mitigation means. Therefore, easily implementable substitution means should be provided.
- Sampling systems and analytical means for plant surveillance and following recovery actions should be provided.
- Treatment, storage and management of radioactive wastes (solid, liquid, gas) should be considered.

This complements guidance for strengthening mitigating systems for accidents such as the Fukushima Daiichi accident:

- Management and operation of mobile or non-conventional equipment should be well defined (storage, maintenance, testing, transfer and training) and should consider possible large contamination. Strategies for deployment in the event of anticipated hazard should be developed (e.g. for situations with degraded infrastructures and access due to extreme external events and/or explosions, accidents with prolonged loss of electrical power and heat sinks, SFP accident, multi-units accidents, etc.).
- Implementation of several connections for mobile equipment (e.g. water make-up pumps and diesel generators) should be considered for relevant safety functions.
- Systems and equipment for cooling the reactor core and spent fuel pools should be independent of normal/emergency power supplies and heat sink (e.g. diesel or turbine-driven pumps with alternate water sources). Sufficient battery capacity or portable power supplies need to support systems, equipment and instrumentation in an extended period of station blackout.
- Reinforced or alternative emergency management facilities and additional or alternative equipment dedicated to providing emergency electrical power supplies and water sources should be considered.

With respect to systems and equipment, including post-Fukushima Daiichi upgrades, the task group estimated that efforts remain to be done to assess their response and reliability under SA conditions including for LTM phases.

Develop harmonised practices and technical means to limit workers occupational exposure during long-term management

> Capitalisation and sharing of good practices and development of guidance to minimise workers occupational exposure in LTMA implementation should be organised.
>
> Development of additional technical means to enhance management of workers occupational exposure should be fostered.
>
> The second recommendation relates for instance to enhancement of existing radiation monitoring systems and related analyses methods for more accurate dose mapping in harsh environment.

Feedback from past accidents has shown that it is of prime importance to provision in an NPP sufficient and adequate workers protection and monitoring means to manage interventions during an SA emergency phase. Improvements both in terms of protection and dose mapping and monitoring could further limit worker occupational exposure.

Dose monitoring and mapping in harsh environments that could guide radioprotection implementation and provide information on radioactive deposits in a damaged plant remain challenging with currently available instrumentation. Further technological and methodological developments in the field should be supported.

Management and operations of mobile and non-conventional equipment for LTMA should be well defined and drills carried out to limit as much as possible workers exposure.

Chernobyl and Fukushima Daiichi damaged plants are still facing considerable radioprotection challenges related to future defuelling and decommissioning actions. Thus, important feedback for radioprotection issues for major LTMA is still expected in the future from these two accidents.

5.3. Recommendations related to knowledge consolidation for specific long-term management

Maintaining long-term cooling

Knowledge should be developed on:

- long-term RPV resilience for situations where RPV failure has been prevented;
- long-term reliability of systems and equipment involved in maintaining a coolable configuration to assess failure risks due to induced mechanical weakness (particularly for fixed systems that have withstood challenging SA conditions), due to clogging with debris and due to corrosion-erosion reactions;
- degradation under SA conditions (high dose and temperature) of materials that may form significant amount of debris, posing challenges for the LT cooling.

These recommendations come in addition to recommendations related to implement provisions for replacement of critical systems and equipment for LT cooling (as far as feasible depending on design).

One of the challenges of LTM is to ensure long-term cooling of damaged fuel/fuel debris, possibly for years depending on the accident scenario, up to their retrieval from the damaged plant.

If an IVMR strategy has been successful (RPV failure prevented when entering LTM), knowledge of long-term RPV resilience considering wall thinning by contact with corium, induced fragilities in some designs (e.g. penetrations welds attacks by corium) and cooling configurations (cooling from inside and outside RPV) is important for LTM as RPV failure on the LT could result in a new unstable state. A large international research initiative is ongoing in the Horizon 2020 Framework Programme (H2020) (IVMR project [Fichot and Carénini, 2016]) to consolidate methods to demonstrate the safety of IVMR strategies for different reactor designs. However, the project will hardly address the recognised issue of RPV resilience on the LT considering in particular the effect of erosion-corrosion reactions on a RPV weakened by an SA transient.

Though management of IVMR on the long term should pose less challenge with the gradual decrease of residual power, thinking about optimised strategies for efficient cooling on the long term and limiting challenges (e.g. erosion-corrosion effects) to the RPV and reactor coolant system (RCS) could also be conducted.

If IVMR has failed (failed RPV when entering LTM), maintaining a coolable configuration may even be more challenging as degraded fuel and debris may be distributed in and ex-vessel, possibly in low accessibility areas. The corium dispersion in the containment might be worse when steam explosion occurs after the RPV failure.

In all cases, fixed or mobile emergency core/containment cooling systems recirculating highly contaminated waters will be involved for maintaining in the long term a coolable configuration. Whether these systems are placed inside or partly outside the containment depending on reactor design, their failure would be critical and should be avoided since possibly resulting in a loss of cooling and in a new unstable state or possibly releasing contaminated liquid in the environment (if a failure occurs on a system outside the containment).

For investigation of clogging issues, a prerequisite is to review existing knowledge on debris sources in an SA (e.g. insulation material, paint chips, latent debris, debris from degraded core materials, etc.) and their expected evolution under SA conditions (chemical effect with formation of precipitates). A significant amount of knowledge exists for loss-of-coolant accident conditions (NEA, 2013) and approaches to assessment of clogging risks of emergency core cooling system for such conditions have been reviewed, but much less knowledge exists for SA conditions.

For investigation of corrosion reactions which may be catalysed in the presence of radiation, main corrosion mechanisms for SA conditions would have to be identified and characterised to assess failure risks of RPV and of circuits recirculating contaminated liquids.

Maintaining sub-criticality

Develop, both for short-term management and LTM of an SA, methods and tools to assess criticality safety margins for damaged fuel considering uncertainties on damaged fuel characteristics (shape, size, distribution, porosity, composition, etc.) and assess the potential consequences of re-criticality events in terms of damage to safety equipment and structures and radioactive material remobilisation.

This recommendation is addressed to criticality experts but requiring support of SA material and mechanical experts. Establishing harmonised criticality risk analyses would also be beneficial.

This recommendation comes in addition to recommendations related to investigate the development of instrumentation to monitor re-criticality risks.

There are presently persistent questions about the possibility of re-criticality events both during short-term and long-term phases of an SA. This is mostly due to uncertainties of the damaged core configuration in the short term and on damaged fuel characteristics and their evolution upon ageing in the long term. It is also linked to the fact that no sub-criticality monitoring exists for damaged configurations. The risk could be more significant in the short term for in-vessel configurations following neutron absorber rods or blade degradation with relocation of neutron absorber material and in the long term in solidified corium and debris due to radio-induced ageing processes. In both situations, water with no or little amount of neutron absorber material can come in contact with degraded fuel material.

There are also some questions about the possibility of re-criticality events in some scenarios of SFP accidents warranting further investigation.

Further assessment of the consequences related to possible re-criticality events in LTM is also important in view of establishing if they could result in new unstable states.

Limiting radioactive releases

Knowledge of the phenomena of remobilisation of radioactive products in the long term, both in contaminated waters and in the atmosphere should be developed. These phenomena include leaching of debris, dust formation from degraded fuel and remobilisation from aerosol deposits.

These recommendations come in addition or complete recommendations to consolidate knowledge on containment behaviour and on long-term phenomena (cf. Section 5.1).

When the containment has failed or its leak tightness deteriorated during the accident, limiting further liquid or atmospheric releases of radioactivity to the environment faces in the following two challenges: i) lack of knowledge of leakage paths and leakage rates and their evolution the LT, and ii) lack of knowledge of transfer and remobilisation processes of radioactive products on the LT in contaminated waters and in the atmosphere.

Having the capacity to assess containment leakages and their evolution in the LT is of critical importance as earlier discussed (cf. Section 5.1).

In order to limit contaminated liquid transfer to the environment, strategies for flooding and cooling the corium should avoid conveying contaminated waters outside the confinement as far as feasible. Systems outside the containment should be robust enough to prevent leakages.

Relative to remobilisation of radioactive products in the LT, knowledge of ageing processes both in contaminated water and in air, particularly to assess the importance and kinetics of fuel fragmentation and dust formation from degraded fuel/debris in the long term (radio-induced damage and oxidation) is today very limited. Also, in the LT some fission products (e.g. Sr) and actinides (e.g. Pu) may contribute more significantly to the radiological hazards by transfer to contaminated waters. For LTM, such knowledge is important for many aspects: management of liquid releases from the damaged plant, of liquid wastes and of fuel retrieval operations.

Atmospheric releases from the damaged plant in the long term up to fuel retrieval can be minimised as long as cooling of degraded fuel/fuel debris can be maintained and energetic events (e.g. hydrogen combustion, re-criticality events) can be avoided. However, knowledge is required to assess remobilisation risks particularly during fuel retrieval operations.

Whatever the state of the containment, if cooling is maintained in the long term, containment pressure control should not be an issue for LTM. Provisions to manage hydrogen combustion risk should be implemented considering its production in the long term by radiolysis and radio-catalysed corrosion reactions.

Site clean-up, decontamination and wastes management

Knowledge should be capitalised (with feedback from past accidents), shared and guidance provided on:

- best approaches and techniques (cleaning, fixing or implementing protection and confinement measures) for the treatment of various contamination types;
- multi-nuclide analysis methods in highly active SA solid and liquid wastes (detection, qualitative and quantitative analyses of major isotopes);
- purification technologies and related processes for highly contaminated SA liquid wastes;
- approaches and techniques for handling, routine monitoring and storage of highly active SA wastes and for defining criteria for selection of mode of management of SA wastes.

An important aspect in implementing these recommendations is the sharing of knowledge between the SA and waste management communities.

Two main aspects have been discussed by the task group in relation to site clean-up and decontamination: i) technics and methods to better characterise contamination – including in not easily accessible areas – would be most helpful to support the management of site clean-up and decontamination operations and could also provide valuable information on contaminants distribution in premises of the damaged plant, and ii) feedback from past accidents should be further capitalised to provide guidance on best technics and approaches for the treatment of various contamination types expected in an SA.

Research to have at hand mapping technics and methods providing data on dose – or major radioactive isotopes distribution – in highly contaminated spaces following an SA, supporting LTMA implementation, should be encouraged.

Sharing of experience from past accidents on existing approaches and technics to treat SA contamination should be organised to identify possible progress in the field and provide guidance on most effective approaches and technics for different types of contamination and for minimising workers exposure.

In relation to waste management, challenges are associated to limiting of volumes, treating and handling highly active wastes and waste characterisation for its handling and storage.

Waste volumes will of course be reduced if mitigation measures have been effective in maintaining containment and reducing the extent of radioactive releases outside the damaged plant and if an optimised management of cooling waters for all LTM phase has been followed (cf. Section 5.4).

On-site solid waste volumes depend on the extent of contamination in the damaged plant and of operations to clean and decontaminate the plant and treat wastes. These operations should be managed so as to limit as much as possible production of contaminated materials in addition to existing ones in particular by avoiding as much as possible dispersion of contamination. Limitation of liquid waste volumes will also reduce solid wastes resulting from their treatment (the filters used for purification).

Relative to treatment and characterisation of highly active wastes, methods, technics and large facilities were developed for past accidents (e.g. for Three Mile Island unit 2 (TMI-2), Chernobyl and Fukushima Daiichi). It seems however that the related experience, notably on multi-nuclide analyses in highly active waste samples and purification processes for contaminated liquids produced during an SA has not fully been capitalised and shared. This would be of interest to ease implementation of waste treatment facilities and improve waste characterisation and monitoring, should another major accident occur. Also, quantitative analysis of major nuclides in waste would provide useful information about radioactive releases from the damaged plants.

The task group also noted that the management of tritium in water wastes may be an issue particularly for large volumes of water and that no technology is presently available to eliminate it when present at low concentrations. No specific recommendation was formulated but it could be valuable to internationally reassess and harmonise criteria for the management of tritium-containing waters.

Fuel retrieval and disposal

Knowledge should be capitalised and developed on:

- damaged fuel distribution and characteristics and their evolution over time, including ageing and leaching effects;
- cutting and recovery methods and technics for damaged fuel minimising contamination dispersion;
- risks related to damaged fuel retrieval (re-criticality, fuel dust formation and dispersion).

These recommendations come in addition to those related to consolidation of SA calculation tools and to enhancement of accident monitoring that should provide more reliable information on radioactive material distribution and properties in a damaged plant with fuel degradation.

Both recommendations are of special importance for the preparation of fuel retrieval and disposal at Fukushima Daiichi and relevant actions have been engaged in the NEA PreADES and TCOFF projects.

As seen at TMI-2 and Fukushima Daiichi, one major issue to manage these operations is the limited knowledge of degraded fuel/debris distribution and characteristics, until access is possible (which could be years after the accident as at TMI-2 and Fukushima Daiichi). One issue is the mechanical resistance of damaged and aged fuel as debris, as solidified corium, as remnants of fuel rods or as visibly intact fuel assemblies. Retrieval may necessitate cutting for most resistant materials but some material may collapse or form dust particles with ageing-thereby possibly affecting significantly material resistance.

For an accident in a SFP, similar challenges may result (with in some cases, the additional issue that no barrier against radiation exists). One possible additional challenge for fuel handling and evacuation could be to identify assemblies that are mechanically weakened if the accident extent was limited to part of the pool. Apparently intact fuel assemblies could collapse during handling.

Consolidation of calculation SA tools relative to material aspects with the support of dedicated experimental work (some performed with simulant materials and some with prototypic materials, particularly as preparation for the decommissioning of the Fukushima Daiichi plant) and knowledge capitalisation from detailed analyses made on degraded fuel samples from past accidents (e.g. TMI-2, Chernobyl) and relevant experiments (e.g. Phebus FP, VULCANO, etc.) should be performed as recommended in Section 5.1. These actions are engaged in the NEA Pre-ADES and TCOFF projects.

The task group noted that more predictive tools could be supportive for fuel retrieval operations. However, enhanced monitoring of the accident progression in and ex-vessel should also be developed as it would directly provide information on the damaged fuel distribution.

This could involve both SA instrumentation providing information during the transient to guide severe accident management (e.g. demonstrating the success of in-vessel or ex-vessel melt retention strategies) and also remote techniques after the accident (distant monitoring of fuel material with sufficient resolution, sampling).

In relation to the limitation of contamination dispersion during the fuel retrieval operations, it has already been suggested to design and conduct research on degraded fuel ageing in water and in air. In addition, it is thought that research for optimised cutting technics should continue. The objective is the selection of appropriate techniques in view of minimising fuel dispersion and workers' occupational exposure during fuel retrieval operations.

5.4. Recommendations related to provisions developments for specific long-term management and actions

Provisions should be developed for optimisation of management of cooling waters, e.g.:

- During the emergency phase, closed-loop cooling should be implemented as early as possible to limit transfer of contaminated waters outside the confinement.
- Strategies for flooding and cooling the corium should as far as feasible avoid transfer of contaminated waters outside the confinement or use qualified and tight systems to prevent contamination spread on-site.
- Methods/criteria to optimise the mode of cooling (such as active cooling with water, natural cooling with water, cooling in air) in the long term in order to minimise contaminated water volumes should be developed.
- Provisions for ensuring the injection at short and long term of water with "controlled" chemistry should be further studied with the following potential objectives: limit re-criticality risks up to the LT, limit possibilities of fission products remobilisation (particularly iodine at short term), limit long-term corrosion reactions, limit possibilities of precipitation and clogging in cooling loops, and facilitate liquid waste management in the long term.

As the Fukushima Daiichi accident has shown LTM, limiting contaminated water volumes and their transfer to the environment could reduce significantly LTM challenges. Analyses of required cooling modes for foreseen plant damaged states and corresponding decay heat distributions could provide guidance for optimised cooling in LTM.

Injections of untreated raw or sea water could potentially increase criticality risks for some degraded configurations (in the absence of neutron absorber additives). The effect in the short term of some impurities on degraded fuel coolability and fission products chemistry (e.g. chlorine affecting iodine chemistry) is not adequately known. The pH control in the days following the accident may be of interest to limit gaseous iodine releases. Kinetics of radiolytic and corrosion reactions as well as precipitation processes may be accelerated due to the presence of specific impurities. This may increase the risk of failure or clogging of circuits and equipment in contact with highly contaminated liquids. The presence of some impurities in significant concentrations (e.g. chlorine) may make the separation processes in waste management more complex.

Appendix A. Questionnaire on long-term management and actions for a severe accident in a nuclear power plant

Background information

The kick-off meeting of the NEA Task Group on Long-Term Management and Actions for a Severe Accident in a Nuclear Power Plant was held on 23-24 February 2015 at the NEA, Issy Les Moulineaux. The work scope of the task group is:

- to review existing regulatory requirements, guidance, practices, issues under consideration and existing technical bases in NEA member countries with respect to long-term management and actions for a severe accident in a nuclear power plant (NPP);
- to identify, describe and review in an exhaustive way the issues to be tackled;
- to potentially propose recommendations and areas of investigations (studies and research) to improve the management and actions for a severe accident in an NPP in the long term.

Following the kick-off meeting, the following definition was proposed for the long-term management of a severe accident:

Long-term management[1] of severe accidents refers to *accident management actions* implemented after a plant reaches a safe stable state following a degraded core accident and up to defuelling of the reactor and removal of spent fuel off-site to permanent or intermediate storage. The timing of a plant reaching a stable state depends on, among other things, the initiating event, plant operating conditions and any and all prevention and mitigation measures to terminate the accident.

A plant is considered in a *safe stable state* when all components of the degraded core are in a *coolable* configuration, either still in place and/or relocated in-vessel and/or ex-vessel, and any stored spent fuel is also in a coolable configuration. The degraded core, if retained in-vessel, is considered to have reached a coolable configuration when there is no further hydrogen production from water-metal (clad and structural materials) interaction, the release rate of fission products is exceedingly low and there is no risk of re-criticality or of corium rupturing the vessel. Similarly, the degraded core, if ex-vessel, is considered to have reached a coolable configuration when there is no further incondensable gas generation from molten core-concrete interaction, release of fission products from core-concrete interactions is exceedingly small, there is no risk of re-criticality and the ex-vessel core debris is retained in the containment without breaching the containment integrity. The spent fuel inventory is considered in a coolable configuration if all the spent fuel rods, degraded or not, are confined in the pool without the risk of a runaway oxidation reaction, there is no significant production of hydrogen and no risk of criticality.

Accident management actions for long-term management of severe accidents comprise the actions taken to: i) ensure the plant remains in a coolable state as defined above until complete defuelling; ii) mitigate unintended off-site consequences; iii) ensure that there are no unintended consequences during the safe stable state of the plant; and iv) prepare for the

1. The definition of "long-term management" in the context of this project is not to be interpreted as the so-called long-term management action traditionally considered by the industry as part of the severe-accident management. The latter definition still relates to bringing a plant to a safe stable state following a degraded core accident, and involves implementation of mitigation measures employing mobile equipment, among other things, brought from off-site locations.

defuelling to ensure that there are no unintended consequences during defuelling of the plant and transfer of spent fuel to a permanent or intermediate storage location.

For more information about the project, the kick-off meeting, participating organisations and the report outline, please visit: www.oecd-nea.org/download/wgama/ltmnpp.

To complete Action 1.7 recorded in the kick-off meeting action list, a set of survey questions are provided below in order to solicit inputs from the task group participants. The input will be used for the preparation of a draft report to achieve the project objectives.

Survey completed by state country, organisation(s).

Survey	Intention of the survey	Response (or provide write-up in separate pages)
a. Are there any existing and planned regulatory requirements and/or guidance specifically for long-term management and actions for a severe accident in your country? If yes, what are they? b. What could be the most important safety objectives associated to the long-term management and actions following a severe accident in your country?	Input to be used for the Status Report: • To summarise the current regulatory requirements and guidance in the field • To list important safety objectives for the long-term management of a severe accident (emphasis may be on other objectives than for short-term management of the accident up to the emergency phase), e.g. linking severe accident management and radiation protection and waste management aspects	
Are there any commendable practices/strategies pertinent to long-term management and actions from your experience? If yes, describe them.	Input to be used for the Status Report: • To summarise possible practices pertinent to long-term management and actions (e.g. use of available on-site and off-site resources not primarily dedicated to severe accident management, etc.) • To provide recommendations regarding practices for long-term management and actions	
What kind of measures and actions to maintain the plant in a long-term safe stable state are planned, envisaged or existing for long-term management of a severe accident?	Input to be used for the Status Report: • To summarise existing, planned or envisaged measures and actions in the field (may be generic or plant/site specific) • Strategies • Procedures and guidelines • Equipment, infrastructure and instrumentation • Human and organisational resources	
Which factors/criteria are/were/should be considered for the design/implementation of such measures/actions?	Input to be used for the Status Report: Potential factors: • Site/plant specificities • Multi-units accidents • Accidents involving spent fuel pool (SFP) • Degradation of equipment, infrastructure and instrumentation • Positive/negative impact of short-term prevention/mitigation actions (more or less controlled during the emergency phase) • Extent of radioactive material releases • Existing on-site and off-site organisation and resources • Etc.	

Survey	Intention of the survey	Response (or provide write-up in separate pages)
Which factors/criteria are/were/should be considered for the design/implementation of such measures/actions?	Potential criteria: • Fulfilment of safety objectives (cooling core and SFP, mitigate further radioactive releases, others targeted to recovery phases, etc.) • Prevention/mitigation of risks (gas combustion, further radioactive releases – gas and liquid, other industrial risks) • Easing recovery, limitation of workers exposure and further environment contamination • Etc.	
Which issues/challenges should be considered? What is the associated knowledge base? Are there any technical issues that deserve further investigations in view of designing efficient long-term management measures and actions?	Input to be used for the Status Report: • Identify, describe and review issues to be tackled • Review existing technical bases and identified gaps • Provide recommendations for further research and studies • Potential technical issues: – assessment of plant conditions after a severe accident (infrastructure, components, systems, instrumentation, radiological conditions, etc.) – risks assessment after an SA – assessment of positive/negative impact of emergency actions (feedback on existing severe accident management strategy?) – corium and debris location and behaviour after an SA – liquid effluent chemistry and treatment – recovery and mitigation techniques and systems – etc.	
What approaches (method or combination of methods, tools, data and computational aids, evaluation criteria, etc.) are, or could be, used for designing long-term management and actions? Based on your experience, what kind of new methods or tools would be convenient or necessary to better address the issue?	Input to be used for the Status Report: • To provide a general description of existing or under development methods, tools, data, criteria (laboratory capacities, monitoring equipment, advanced analysis and evaluation tools, SA codes, prognosis/diagnosis tools, probabilistic safety assessment, etc.) that could be used to assess the degraded core state and the radiological situation, to plan and optimise recovery actions and design well focused long-term management and actions • To discuss the merits and shortcomings (limitations) of these methods and tools for the purpose of assuring long-term management and actions effectiveness • To provide recommendations for future developments devoted to increasing confidence in methods to design long-term management and actions (e.g. integrated accident management approaches, etc.)	

Appendix B. Simplified application of SA-LT categorisation from level 2 probabilistic risk assessment release category figures of merit

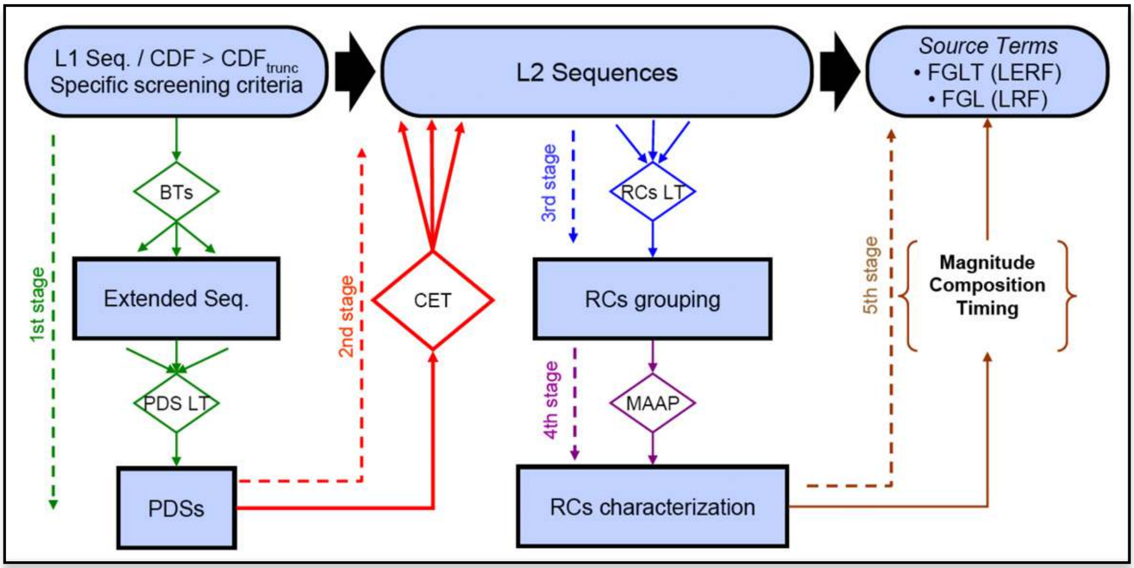
Level 2 probabilistic risk assessment release categories as a starting point

Conduct of long-term operations, decision making and preparedness before loss of safe stable state strongly depends on plant configuration-specific issues. Even though large number of variables contributes to plant damaged state (PDS) characterisation, only a few of them turns to be significant from a back-end approach hence making the long-term PDS identification process subject to logic-binning-based classification.

The overarching objective of such classification targets towards improving the entire set of means for managing the severe accident long term by helping highlighting main risks specific of the plant configuration. Therefore, plant state characterisation can be seen as the preliminary task for addressing long-term safety issues.

Level 2 probabilistic risk assessment (PRA) release categories constitute groups of core-damage accident sequences featuring similar source term release characterisation in terms of magnitude, composition and timing as a consequence of similar containment evolution and failure location and type. Figure B.1 depicts a generic level 2 flowchart where main tasks are placed along the top row and connected below to user-dependent subtasks.

Figure B.1. **Level 2 probabilistic risk assessment flowchart**



Source: OECD/NEA.

- First stage: level 2 starts from coupling sequences leading to core damage with containment systems performance through so-called bridge trees, whose outputs are classified into so-called level 2 plant PDSs through the use of the PDS logic tree (PDS LT) according to similar evolutions in containment.
- Second stage: single PDS evolves following different sequences as a result of high level of uncertainties related to severe accident phenomena through containment event tree/accident progression event trees application for uncertainty propagation.
- Third stage: release category logic tree (RCs LT) application to group containment event tree/accident progression event trees outputs according to similar release term characterisation: magnitude, composition and timing[1].

Transformation process from level 2 PRA RC into SA-LT PDS

Generic features dealing with release categories are briefly set forth to help carrying out transformation into SA-LT PDS:

- release category sequences lead either into containment/bypass failure, i.e. source term release to environment, or into safe state;
- release category logic tree classification criteria basically attend to the following issues:
  - containment isolated or bypassed;
  - containment type of failure;
  - related source term characterisation (mostly timing and event);
  - probability issues since release category sequences are also risk oriented, i.e. frequency values assigned to each accident sequence must be tracked throughout the entire sequence hence treated distinctly.
- Most of level 2 PRA applications limit mitigating systems availability to design basis accident-related, hence alternative safety equipment of any kind, such portable or non-conventional, are not credited.

According to the former considerations, steps carried out to convert release categories into SA-LT PDS are the followings:

1. Combine release categories by removing above criteria classification on source term characterisation related phenomena and probabilistic issues[2]: From the perspective of SA-LT, plant configuration after attaining safe stable state will not be influenced by whether severe accident sequence yields *x* or *2x* release values, and whether containment failure occurs *y* or *2y* hours after initiating event.

   Nonetheless, several particular accident management actions belonging to the "accident analysis" category and "accident management" will be affected by related source term characterisation criteria, e.g. outside radiation levels linked with accident management action 1 on ensuring the plant remains in a safe stable state through "working conditions" issue, action 2 on mitigating releases. Such differences will be taken up in further step 4 by considering different scenarios each of which corresponds to particular boundary conditions as specified in removed top events. Last, the majority

1. Fourth stage dealing with RC characterisation through system code calculations such as Accident Source Term Evaluation Code (ASTEC), MELCOR or Modular Accident Analysis Programme (MAAP) might be of interest when performing accident analysis and boundary conditions evolution. Fifth stage for regulatory purposes to identify RCs whose source term exceeds certain magnitude threshold at a particular time, i.e. large early release frequency identification, goes beyond our goals.
2. Different risk contributions of each sequence will turn out to be very useful if thermal-hydraulic evolutions need to be simulated, for instance to characterise containment environmental evolution or to identify cliff-edge effects. Therefore, it should be possible to trace back each SA-LT PDS to level 2 PRA release categories, and from here to level 1 interfacing level 2 PRA PDS within which each sequence leading to core damage is associated with a specific frequency.

of level 2 PRA applications includes top event questioning on reactor pressure vessel (RPV) integrity. Since this aspect of severe accident evolution is fundamental for SA-LT PDS characterisation, this top event should be kept.

SA-LT PDS characterisation task may be finalised at this point. However, since many crucial aspects dealing with improving severe accident management depend on full plant configuration, i.e. state of the plant linked with mitigating system alignment, each of the resulting scenarios will hence be completed through coupling with mitigating systems fulfilling safety functions as stated in Section 3.2.

2. Safety functions addressed in each sequence will be identified through a check list wherein only those minimal configurations to meet with minimal safety functions will be pointed out and subsequently addressed.
3. Each release category is branched into and coupled with mitigating systems alignment to fully meet with SA-LT safety functions, where each alignment corresponds to a specific plant configuration (including backup systems). With this aim, a parallel task of identifying all means of carrying out each safety functions will be performed.
4. For each obtained PDS configuration according to previous steps 1-3, distinctions previously neglected on source term releases (magnitude, time, leakage receiver building/area) will be here taken into account within each PDS category as long as leading to a different thermal-hydraulic evolution or radioactive spreading map hence affecting mitigating release related actions.
5. The entire list of SA-LT PDSs will then be the result of i) RC sequences coupling with ii) mitigating systems alignment within which iii) different thermal-hydraulic and radioactive spread boundaries affecting accident management issues. It is worth noticing that particular i) mitigating releases or ii) defuelling actions may apply to more than one PDS as classified.

Application

The method set forth above will be applied to generic Western large-dry containment pressurised water reactor (PWR) as plant reference and level 2 PRA release category logic tree classification criteria as depicted in Table B.1. Top row collects the binning criteria and bottom row the possible answers to each of the criteria. Aside from design basis accident standard safety systems, back-fitting system includes active ex-vessel cooling by reactor cavity flooding once reached 649 Celsius degrees at Core Exit Thermocouples. Back-fitting passive safety systems include passive autocatalytic recombiners installed inside containment. In order to simplify the application, and given its high reliability degree, passive autocatalytic recombiners will not be part of the containment event tree/release category logic tree, i.e. they will not be put into question.

Table B.1. **Release category logic tree**

CONTAINMENT INTEGRITY	IN-VESSEL STEAM EXPLOSIONS	CONTAINMENT ISOLATION STATE	IN-VESSEL COOLING	EX-VESSEL COOLING (IVMR)	EARLY CONTAINMENT FAILURE (DCH)	MCCI	SG ISOLATION STATE	WATER-COVERED BREACH	CONTAINMENT COOLING STATE	CONTAINMENT SPRAYS	CONTAINMENT FAILURE TYPE
Not bypassed SGTR ISLOCA-V ISLOCA-VR	Yes No	Isolated Large failure Small failure	Yes No	Yes No	Yes No	No MCCI Flooded Not flooded	Yes No	Yes No	Yes No	Yes No	ALPHA DCH Success H2 Exp. Basemat erosion Overpressure

Source: OECD/NEA.

Removal of source term release related classification criteria

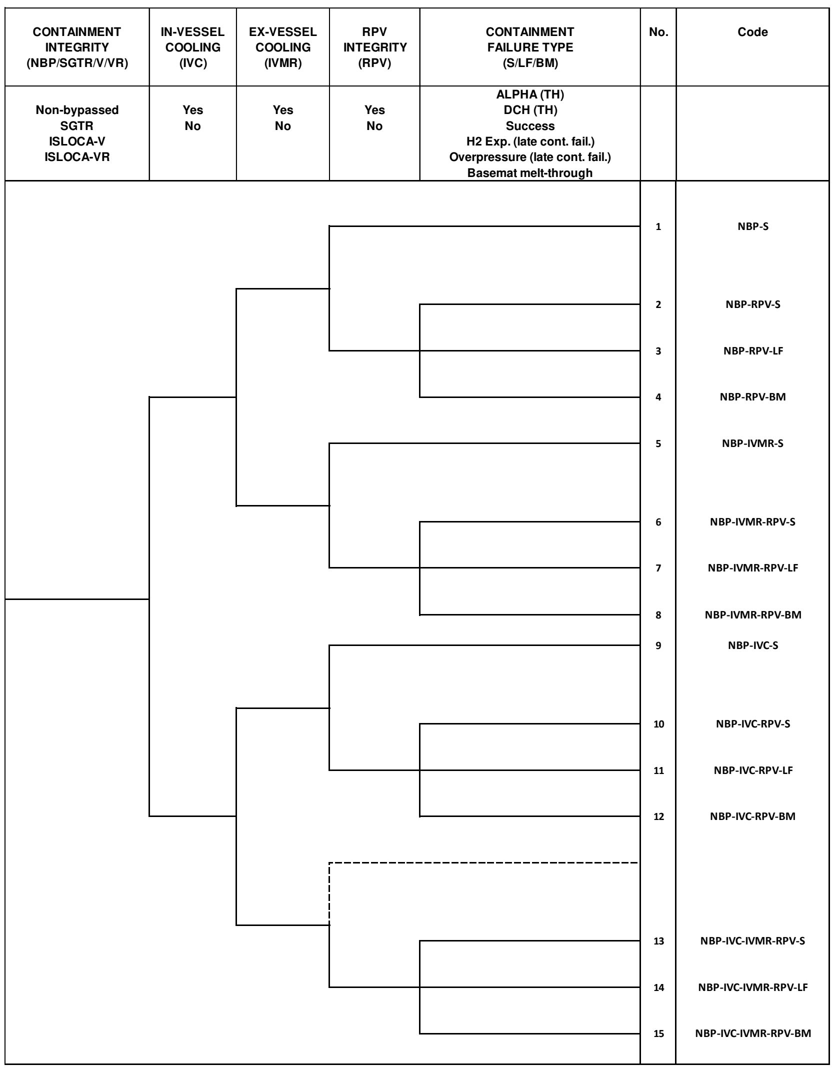
In order to identify the reasons underlying potential screening out of release category logic tree classification criteria, let us proceed analysing one by one:

- Containment integrity: bypass alternatives comprise steam generator tube rupture as initiating or induced event, interfacing system loss-of-coolant accident as initiating event (ISLOCA-V) or at recirculation switch (ISLOCA-VR), e.g. through refuelling water storage tank. Plant configuration will largely be affected depending on containment failure location.
- In-vessel steam explosions: this particular phenomenon stands for alpha containment failure mode where RPV lid thrust by in-vessel steam explosion at low pressure causes containment failure. Contribution from this category to the spectrum of plant configurations only deals with source term release time hence should only be subsequently taken into account in step 4 within early containment failure thermal-hydraulic sequence characterisation.
- Containment isolation state: this classification criterion points out at source term magnitude of the release. Alongside "isolated", alternatives are "large", meaning failure size between 5 and 20 inch equivalent diameter, and "small", meaning isolation failure size between 3 and 5 inch equivalent diameter. In this regard, it will only be taken into account within thermal-hydraulic sequence characterisation. From the perspective of plant configuration, e.g. water flooding from one building to another, it may be subsumed within containment failure condition.
- In-vessel cooling (IVC): plant configuration will obviously be affected in case of preventing RPV failure by flooding the core. Core reflooding without success might instead be considered as a different PDS or included within failure event sequences wherein it will be separately treated for thermal-hydraulic issues. For our application, failure to succeed in IVC will be taken as a different PDS.
- Ex-vessel cooling (in-vessel melt retention, IVMR): IVMR will be treated as IVC.
- Early containment failure (direct containment heating): Early failures will be further taken into account within thermal-hydraulic characterisation.
- Molten corium-concrete interaction (MCCI): MCCI is classified according to quenching degree in "no MCCI" for very rapid corium quenching by high efficient corium to water heat transfer; "flooded" for situations where water is available but high MCCI likely leading to containment failure whether because of basemat melt-through, overpressure or hydrogen explosion; in case no water is present to flood the reactor cavity, "not flooded" alternative should be selected. On one hand, plant configuration will not be affected by the extent of MCCI as long as not jeopardising the containment barrier; on the other hand, these categories allow distinguishing between whether RPV is intact ("no MCCI") or breached (other two alternatives). In order to keep this crucial criterion for PDS characterisation, this top event will thus be transformed in question on RPV status.
- Steam generator isolation state: this criterion addresses specifically steam generator tube rupture scenarios and distinguishes between steam generator relief valves stuck open or cycling. This criterion only looks at source term magnitude hence it will be neglected in terms of PDS configuration.
- Water-covered breach: this criterion only looks at the source term magnitude (and composition) of the release so it will not be taken into account for SA-LT plant configuration spectrum characterisation.
- Containment cooling state and containment sprays: the rationale underlying both criteria rests on different containment failure frequencies since depending on gas concentration in the containment atmosphere, probability linked to hydrogen explosions or over-pressurisation will be likewise different. This criterion also affects to source term magnitude of the release through containment spray scrubbing. Therefore, it will be neglected since dealing with probabilistic and magnitude issues. Moreover, mitigating system alignment will further be taken into account in the next step.

Containment failure type: "alpha" and "direct containment heating" early kind of failures already treated. "Success" was initially limited to core reflooding or IVMR and it will be accounted for. Basemat erosion will also be taken into account. Regarding "hydrogen explosion" – which actually refers to late containment failures –, and over-pressurisation, both categories, even if featuring different phenomenological signatures, do not correspond to different SA-LT plant configurations so that they will be grouped as "containment late failure"[3].

Figure B.2 depicts the filtered classification criteria acting as top events of an event tree where each sequence represents SA-LT PDS characterisation, which constitutes the first step if coupling with mitigating system alignment is to be performed. Depicted sequences only account for containment integrity "not-bypassed" option.

Figure B.2. **Not-bypassed release category event tree**



Notes: NBP – non-bypassed; RPV – reactor pressure vessel; IVMR – in-vessel melt retention; IVC – In-vessel cooling; S – success; LF – late failure; BM – basemat melt-through failure.

Source: OECD/NEA.

3. Instead, containment failure location becomes highly relevant for identifying the spectrum of SA-LT plant configurations and subsequent actions related to accident management.

SA-LT PDS minimal safety functions

According to Section 3.2, safety functions comprise reactivity control (A), generated heat removal (B) either in-vessel (B1) or ex-vessel (B2), and containment protection within which combustible gas control, basemat melt-through prevention and containment pressure control (C1, C2, C3, respectively).

Each non-bypassed SA-LT PDS sequence is coupled with requested safety functions through check list as shown in Table B.2 to highlight which safety functions need to be addressed.

Table B.2. **Minimal safety function configuration for non-bypassed SA-LT PDSs**

SA-LT PDS		Safety functions					
No.	Code	A	B1	B2	C1	C2	C3
1	NBP-S	X	X	X[4]	X	N/A	X
2	NBP-RPV-S	X	N/A	X	X	X	X
3	NBP-RPV-LF	X	N/A	X	X	X	N/A
4	NBP-RPV-BM	N/A[5]	N/A	X	X	N/A	X
5	NBP-IVMR-S	X	X	N/A	X	N/A	N/A
6	NBP-IVMR-RPV-S	X	N/A	X	X	X	X
7	NBP-IVMR-RPV-LF	X	N/A	X	X	X	N/A
8	NBP-IVMR-RPV-BM	N/A	N/A	X	X	N/A	X
9	NBP-IVC-S	X	N/A	X	X	N/A	X
10	NBP-IVC-RPV-S	X	N/A	X	X	X	X
11	NBP-IVC-RPV-LF	X	N/A	X	X	X	N/A
12	NBP-IVC-RPV-BM	N/A	N/A	X	X	N/A	X
13	NBP-IVC-IVMR-RPV-S	X	N/A	X	X	X	X
14	NBP-IVC-IVMR-RPV-LF	X	N/A	X	X	X	N/A
15	NBP-IVC-IVMR-RPV-BM	N/A	N/A	X	X	N/A	X

Notes: NBP – non-bypassed; RPV – reactor pressure vessel; IVMR – in-vessel melt retention; IVC – In-vessel cooling; S – success LF – late failure; BM – basemat melt-through failure.

Source: OECD/NEA (2018).

SA-LT PDS coupled with minimal safety functions configurations

Within each safety function, the entire list of combinations of hydraulic pumps, water/boric acid/fuel reservoirs, heat exchangers, heat sinks and associated required instrumentation and control including support systems will be developed. Whenever several safety functions might be accomplished by one single mitigating system combination, minimal safety functions linked to SA-LT PDS will hence be simplified. This might be for instance the case of functions B2 and C2, or in some cases of A and B1.

Each combination might be tagged according to the safety function representative letter followed by a number so that each SA-LT PDS will then be linked and branched, for instance, as 1/B1-1/B2-1/C3-1; 3/B2-1; etc. In these two examples, there is only one option, i.e. one combination of safety systems, support systems and instrumentation and control, to address combustible gas control (C1) so that it could be removed from the sequence codification.

4. Success in saving the RPV through IVC together with IVMR arise problems for the technical support centre to know whether RPV integrity depends whether on RCS injection or reactor cavity flooding or on both. Therefore, this configuration compels the technical support centre to keep operating both systems.

5. Corium re-criticality may not be an issue in such a case as corium would have dissolved a significant amount of concrete.

Thermal-hydraulic sequences linked to each SA-LT PDS minimal safety functions configuration

According to the current application as depicted in Figure B.2, thermal-hydraulic initial and boundary conditions will be characterised as the combination of the following aspects:

- most risk-significant sequence traced back upon sequences grouped under level 2 PRA Release Categories included in each SA-LT PDS;
- Minimal safety functions alignment coupled with the selected sequence;
- combination of consistent thermal-hydraulic and radioactive different conditions as identified in Table B.1, e.g. isolated/small failure/large failure dealing with containment isolation system; direct containment heating (DCH) (application limited to high reactor coolant system (RCS) pressure at RPV failure scenarios), etc.

Resultant sequences as codified are for instance 1/B1-1/B2-1/C3-1/Isolated/non-DCH and 1/B1-1/B2-1/C3-1/Small_Isolation_Failure/DCH.

Appendix C. Fault tree analysis and event tree analysis

Fault tree analysis has proven fundamental to identifying safety systems weak points. Design extension conditions concept envisages the enhancement of plant capabilities to withstand or mitigate severe accidents. After the events unfolded at Fukushima Daiichi, a huge number of actions, both in back-fitting systems and guidelines, have been applied to such an extent that baseline capabilities in the field of beyond design basis accidents have been fully modified. Available mitigating systems increasing complexity, including back-fitted or non-conventional, fixed or portable, prompts parallel extension of the probabilistic risk assessment (PRA) scope to better address this new field of safety and help predict potential weak points hence improving its long-term working performance with typical insights coming from PRA.

Aside from fault tree analysis implementation, event tree analysis tool application, through an *accident* set of event trees and a *baseline* set of event trees (one per each plant damaged state (PDS) and each mitigating system failure/challenging or loss of critical safety function), could shed light on predicting accident sequences in the long term.

A*ccident* event tree analysis will follow traditional level 1 approach, i.e. a top-down analysis of all kind of initiating events leading to loss of so-called safety pillar. *Accident* event tree initiating events should be identified through a twofold approach based on the initiating event together with plant configuration[1], i.e. considering the challenging and/or loss of a critical safety pillar for every type of severe accident long-term PDS.

In the *baseline* event trees, failures come from challenging or loss of critical safety functions stemming only from intrinsic mitigating system failures given that working systems in the long term will be practically limited to systems performing safety functions, i.e. mitigating system themselves.

Furthermore, additional benefits will derive from event tree application since many systems will make use of same equipment and/or human actions and backup systems and infrastructure, hence dependencies among them might be better highlighted by means of Boolean treatment of the entire sequence rather than on a single-system basis.

Within fault tree analysis, each option available to meet one safety function will be made up of a house event through which appropriately select the mitigating system configuration meeting that safety function.

Accounted for systems should be all those having been included in severe accident management guidelines (SAMGs) – whether conventional and non-conventional, fixed and portable, on-site and off-site – together with dedicated back-fitting mitigating systems, as long as their deployment to cope with the severe accident is foreseen.

Even if SA sequence event tree prognosis in the long term might be a difficult issue to realistically cope with and foreseen sequences might depart from reality, benefits will draw more from qualitative prioritisation and identification results through relative ranking of highlighted potential relevant events and mitigating systems structure, system and components than from quantitative results of selected figures of merit (i.e. the *absolute* result – the number – not as important as the *relative* result – how the component or minimum cut set, i.e. long-term accident sequence evolution, is ranked).

1. Plant configuration will be sensitive not only to plant damaged state itself (accident sequence signature affecting the long-term phase management and actions) but also to the mitigating system alignment.

It is worth noting the following aspects concerning PRA applications:

- Benefits drawn from PRA applications are not limited to quantifying the sources of risk but to carrying out a comprehensive analysis of the plant performance as a whole. In this regard, PRA is in the position of highlighting weak points thus allowing utilities addressing them.
- In terms of allocated efforts, level 1 PRA application mostly relies on an accurate quantification of the likelihood side of risk, whereas the consequence side, namely core damage, is initially fixed according to a particular figure of merit, e.g. temperature excursion, core exit thermocouples at 1 200 F, collapsed water level at TAF (top of active fuel), etc., and then simply calculated with the help of a thermal-hydraulic system code. The underlying reason allowing not paying too much attention to the consequence side is the simplification made when considering that all sequences coming into a hazard situation share exactly a similar consequence, i.e. core damage, no matter whether such damage is more or less limited, i.e. more extended, more rapid, etc.
- On the other hand, allocated efforts to develop a level 2 PRA – with the exception of those interfacing event trees on containment safety systems – both rely on the probabilistic and consequence side, the former being addressed by carefully computing the so-called split fractions, the latter by carefully performing the necessary severe accident thermal-hydraulic calculations.
- As of the long-term phase of the accident, the consequence side of risk can be highly difficult to compute since to have a clear picture of the radioactive releases resulting from a loss of a safe stable state, or mishandling activities during decontamination or fuel retrieval, presents so high level of uncertainties to prevent an accurate quantification.
- This is why PRA application should apply to one and only field of activities, i.e. one and only long-term management (LTM) goal, so that, just as in level 1 PRA, for all sources of risk only one consequence applies, i.e. all sources of risk end up with a similar consequence. This is the way to simplify the PRA application by limiting the attention to the likelihood side of risk.
- Among the LTM top goals, PRA is well suited to cope with maintaining a coolable configuration since the comprised actions follow a very similar nature than that of level 1 PRA: they all rely on interconnected safety systems interfacing with a source of heat and radiation.
- Notwithstanding the above, the PRA tool might be applied to other LTM top goals different than the one related to maintaining a coolable configuration, provided the goal ultimately relies on systems constituted by mechanical, electrical and instrumentation and control components, subjected to human actions and environmental constraints.

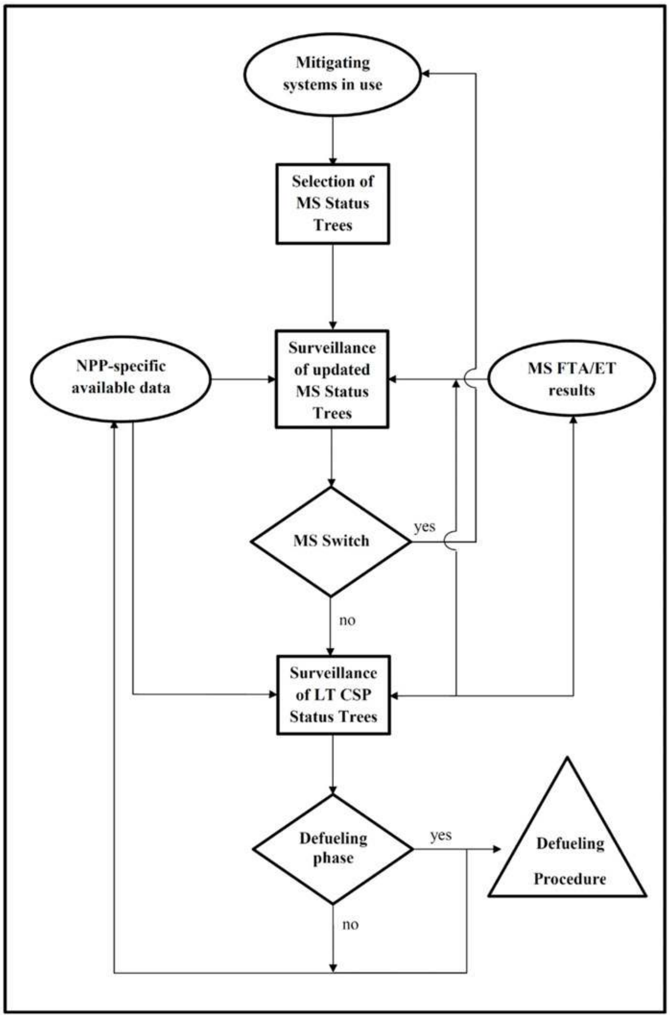
Appendix D. SA-LT management schematic guiding procedure

Assuming that i) a coolable situation has been reached ii) by the use of preconceived mitigating systems, the goal is to provide managing staff with an integral comprehensive guideline linked with specific mitigating system-oriented and safety function-oriented collection of guidelines. This is a response for tackling with severe accident management guideline (SAMGs) extension to cover the damaged plant long-term operation.

The SA-LT management integral process tool layout is depicted in Figure D.1, wherein a round-shape box means "information source" and a rectangular-shape box means "action".

The meaning of the most significant items is addressed below.

Figure D.1. **Long-term management flowchart**



Source: OECD/NEA.

- Mitigating systems in use: according to hypothesis (ii), coolability conditions have been reached through the use of pre-existing systems, either located on-site or off-site, conventional or non-conventional, fixed or portable. Mitigating system in use should be updated upon surveillance criteria or system failures.
- Selection of *mitigating systems status trees*: for each of the working mitigating systems accomplishing a specific critical safety function, there is a dedicated flowchart to indicate the associated surveillance programme, alternative equipment and corrective actions (please refer to the following section below).
- Surveillance of updated mitigating system status trees: mitigating system status trees will be updated (if possible) whenever available nuclear power plant (NPP) data in terms of the affected physical quantities such as inlet and outlet flowrates, liquid levels for keeping with system performance such as water inventory and fuel refilling, room ventilation, etc., is available. Updated values will be taken from critical safety pillars trending values information. Alternatives for *blind* scenarios, i.e. without instrumentation and control, might be considered. Surveillance criteria, component and human action risk ranking, etc., will be taken offline from the corresponding fault tree analysis and (if needed) event trees.
- Mitigating system switch: should mitigating systems in use be replaced for operating or maintenance reasons, the process will be restarted.
- Surveillance of *long-term critical safety pillars status trees*: long-term critical safety pillars are safety issues dealing with either keeping the coolable state (*inner* pillars) or with mitigating radioactive releases and effluents (*outer* pillars). Please refer to the section below.

Mitigating system status trees

Mitigating system status trees (MSST) should be developed for each mitigating system. The MSST content shall be updated with available relevant in-plant data and fault tree analysis results. MSSTs shall include the following aspects:

- Surveillance programme according to risk-informed (offline, pre-calculated) outputs from probabilistic safety assessment fault tree analysis (in line for instance with US Nuclear Regulatory Commission (NRC) maintenance rule) and potential recovery actions (on that particular mitigating system) derived from accident event trees.
- Table of mandatory maintenance, preventive actions and related keep-going available times (for instance, upon an inlet flowrate there is the need of refuelling water storage tank/CST replenishment after X hours). The fill-in form should be built up from probabilistic safety assessment outputs dealing with actions to maintain the performance of the system.
- Frequency of parameters reading should also be indicated.
- *Blind scenario* performing values (conservative values).

Long-term critical safety pillars status trees

As said before, long-term critical safety pillars are those safety functions dealing with either keeping the coolable state (*inner* pillars) or with mitigating radioactive releases and effluents (*outer* pillars).

Each pillar shall contain precursors to safety challenges and associated actions, including relevant plant input data such as pressures, temperatures, liquid levels or flammable gases concentration for tracking purposes. Alternatives for *blind* scenarios should be considered. Significant parameter trends and potential corrective measures (embracing all kind of pre-existing alternatives to meet that safety function aside from the current working mitigating system) coming from accident event trees should be also taken into account (in the line of *what if* responses challenging the safety pillar). Mitigating and generic surveillance actions, alternative solutions, etc., specific for every plant (or accident sequence type) configuration (and, if applicable, to every particular mitigating alignment), should be part of the long-term critical safety pillars guideline (for instance, checking the stuck-open position of pressurised water reactor (PWR) steam generator pilot operated relief valves in a steam generator tube rupture (induced or not) accident sequence type).

Each safety function-oriented pillar guideline will depend or will have subsections depending on plant configuration (accident sequence type and coolability (mitigating) alignment) where accidents would be classified according to a set of critical factors affecting success on pillar safety goals (hence leading to different sequence evolutions): steam generator tube rupture, containment isolation failure, in-vessel melt retention (IVMR), containment mechanical failure, interfacing system loss-of-coolant accident (ISLOCA), etc. For instance, should an ISLOCA occurred, an in connexion with radioactive releases mitigating pillar, specific instructions should be provided to manage the auxiliary/reactor building flooding level to identify the appropriate containment/CPV flooding flowrate in order to minimise the water being flowing out to the attached buildings to containment, simultaneously classifying the containment/CPV pressure pillar as a non-important pillar; if the containment has not mechanically failed, provisions to check the state of potential leakages through the weakest seals such as personal or emergency hatches should be addressed; etc.

First pillar, called *diagnosis pillar*, different in nature with respect to the safety function-oriented pillars, shall be specifically devoted to accident sequence diagnosis and SA-LT Plant Damaged States since knowledge of these issues will affect surveillance implementation for subsequent pillars:

- NPP general safety issues status (infrastructures, communication, operating staff) to feed subsequent safety pillars and mitigating system fault tree analysis/event trees;
- SA-LT plant damaged state to feed subsequent safety pillars and mitigating system fault tree analysis/event trees;
- status of defence in depth barriers;
- available expertise staff (management issues);
- available safety equipment (conventional/non-conventional, on-site/off-site, fixed/FLEX) for each critical safety pillar;
- tracking of additional sources of risk (flooding, fire, habitability);
- SA-LT PDS-specific (including mitigating system alignment) event tree sequence results to subsequently feed long-term critical safety pillars in terms of due surveillance actions, mitigating preparedness (alternative equipment), recovery actions, predictive risks and available time for associated corrective actions (outside the working mitigating system boundaries), etc.

Each other SA-LT PDS-specific (including mitigating system alignment) long-term critical safety pillar shall contain:

- recovery actions derived from the diagnosis pillar dealing with mitigating releases (e.g. preparation for managing large continuous masses of radioactive liquid effluents; closing of stuck-open valves), repairing unavailable equipment, etc.;
- precursors (quantitative settings) to anticipate safety challenges, each of which linked with and leading to modifications of the mitigating system working values;
- baseline event tree results for available time before departing from a safe stable state as defined in terms of critical safety functions/pillars, e.g. coolability, combustible gas concentration, containment pressure (inside the active mitigating system boundaries), etc.;
- alternative systems able to accomplish with identical safety functions according to mitigating system (offline, pre-calculated) fault tree analysis results (linked to currently existing appropriate guidelines to perform preparedness actions on alternative systems);
- significant plant input data such as pressures, temperatures, liquid levels or flammable gases concentration values and trends for tracking purposes and feedback adjustment of mitigating system performing values;

- corrective, recovery actions in case of unanticipated loss of the safety pillar both from the accident event tree (if originated by an external event or internal different than mitigating system failure) and baseline event tree (if originated by an active mitigating system failure);
- surveillance related actions (risk oriented).
- specific managing staff in charge of addressing and monitoring each long-term management of critical safety pillars.

The preliminary list includes the following pillars[1]:

- criticality (inner action);
- decay heat removal (inner action);
- flammable gases and oxygen (inner action);
- containment over and low-atmospheric pressure (inner action);
- gaseous radioactive releases (outer action);
- liquid or solid (effluents) radioactive releases (outer action);
- status of defuelling (outer action).

Mitigating system fault tree analysis/event tree results

Fault tree analysis/event tree database should be fed with the following information (in line with applications such as the monitor risk):

- plant configuration (accident sequence type);
- available/unavailable equipment;
- available/unavailable infrastructures;
- available/unavailable technical (operator fields) and managing staff;
- time since the initiating event or scram (only for related severe accident system code simulations).

Fault tree analysis database should feed mitigating system status trees with the following information:

- surveillance programme hence risk-ranked components to be monitored (offline);
- available times for addressing keep-going actions (online), entirely in case of blind scenarios; partly in case of non-blind such as battery depletion time according to the SA-LT PDS (including mitigating system alignment).

Event tree database should feed long-term critical safety pillars with the following information:

- alternative systems available to meet the critical safety function (once the entire list of SA-LT PDSs have been updated according to the supplied information as listed above);
- available times in case of a sudden loss of the critical safety pillar (inside (departure time from coolability) and outside (available time for the corrective chain of actions) the operating mitigating system boundaries), from associated deterministic simulations; offline but updated according to the elapsed time from the initiating event or scram;
- corrective actions in case of a sudden loss of the critical safety pillar both in case of external or internal event (common-cause failure for the entire mitigating systems meeting with that safety function as identified in SA-LT PDS plus mitigating system alignment characterisation).

1. Pillars shall focus on monitoring quantities rather than on accomplishable safety functions, i.e. containment/CPV flooding is not a pillar, but containment/CPV water level or flammable gases grown-up (whether in-vessel or ex-vessel) is a pillar (the latter being a representative figure of merit of the coolability state of the corium).

References

Abagyan, A.A. et al. (1986), “Information Prepared for the IAEA on the Chernobyl NPS Accident and its Consequences”, *Atomnaya Energiya*, Vol. 61, N5, pp. 301-320 (in Russian).

Anderson, E.B, B.E. Burakov and E.M. Pazukhin (1993a), “Chernobylite, a Reaction Product of Nuclear Fuel and Construction Materials of the Destroyed Fourth Block of the Chernobyl NPP”, *Soviet Radiochemistry*, Vol. 34(5), p. 624.

Anderson E.B., B.E. Burakov and E.M. Pazukhin (1993b), “High-Uranium Zircon from Chernobyl Lavas”, *Radiochim. Acta*, Vol. 60, pp. 149-151.

Anderson, E.B., B.E. Burakov and E.M. Pazukhin (1992), “Secondary Alterations of Fuel-Containing Masses of 4th Unit of Chernobyl NPP”, *Radiochemistry*, Vol. 34(5), pp.135-138.

ANS (1989), “Proceedings of the ANS Topical Meeting on TMI-2, Washington DC, 30 October to 4 November 1988”, *Nuclear Technology*, Vol. 87 (138 documented papers), ANS.

Arutyunyan, R.V. et al. (2010), “Nuclear Fuel in the ‘Shelter’ Encasement of the Chernobyl NPP”, M. : Hauka, p. 240 (in Russian), www.ibrae.ac.ru/docs/109/velihoviallismallisize1.pdf.

Belyaev, S.T. et al. (1990), “Technical Justification of Nuclear Safety of ‘Shelter’ Object”, research report of RRC Kurchatov Institute, Chernobyl, p. 160 – in Russian (Беляев С.Т., Боровой А.А., Волков В.Г. и др. Техническое обоснование ядерной безопасности объекта «Укрытие». Отчет о НИР КЭ при ИАЭ им. И.В. Курчатова, Чернобыль, 1990, 160 с).

Belyaev, S.T. et al. (1989), “Some Aspects of Post-Accident Work in the Control Zone of the Chernobyl Nuclear Power Plant”, *Proceedings of a Symposium: Recovery Operations in an Event of Nuclear Accident or Radiological Emergency*, IAEA, Vienna, 6-10 November, p. 185-202.

Belyaev, S.T., A.A. Borovoj and A.Yu. Gagarinskij (1989), “Peculiarities of Nuclear Fuel Inside the Confinement System of the Damaged Reactor at Chernobyl”, *Proceedings of the Symposium on Recovery Operations in an Event of Nuclear Accident or Radiological Emergency*, 6-10 November, p. 185-202, IAEA, Vienna.

Borovoy, A.A. and E.P. Velikhov (2013), “Experience of Chernobyl (Works in “Shelter” object)”, Part 3, RRC Kurchatov Institute, Moscow, p. 156 – in Russian (А.А. Боровой, Е.П. Велихов. Опыт Чернобыля (работы на объекте «Укрытие»). Часть 3. М.: НИЦ «Курчатовский институт», 2013, 156 с).

Borovoy, A.A. and E.P. Velikhov (2012), “Experience of Chernobyl (Works in “Shelter” object)”, Part 1, Moscow, RRC Kurchatov Institute, p. 168 – in Russian (А.А. Боровой, Е.П. Велихов. Опыт Чернобыля (работы на объекте «Укрытие»). Часть 1. М.: НИЦ «Курчатовский институт», 2012, 168 с).

Borovoy, A.A. et al. (1996a), “Analysis of Current Safety of ‘Shelter’ Object and Projections of Situation Development”, ISTC Report No. 3601, Chernobyl, p. 272 – in Russian (Анализ текущей безопасности объекта «Укрытие» и прогнозные оценки развития ситуации. Отв. Исполнитель Боровой А.А. Отчет МНТЦ «Укрытие», № 3601, Чернобыль, 1996 г., 272 с).

Borovoy, A.A. et al. (1996b), “Dynamics of Radiation Situation in the “Shelter” Object, “Shelter” Object in 10 years”, main research results, National Academy of Sciences of Ukraine, Chernobyl, pp. 100-111, – in Russian (Боровой А.А., Евстратенко А.С., Криницын А.П. и др. Динамика радиационной обстановки на объекте «Укрытие». В сборнике «Объект «Укрытие» – 10 лет», основные результаты научных исследований», Национальная Академия Наук Украины, Чернобыль, 1996, с. 100-111).

Borovoy, A.A. et al. (1990), "Artificial Products of Interaction of Fuel with Construction Materials of 4th Unit of Chernobyl NPP", *Sov. Radiochem*., Vol. 32, pp. 103-113.

Borovoy, A.A. (1989), "Fission Product and Transuranic Release during Chernobyl Accident", International Conference "The fission of nuclei – 50 years", Leningrad.

De Geer L.-E., C. Persson and H. Rodhe (2017), "A Nuclear Jet at Chernobyl Around 21:23:45 UTC on April 25, 1986", *Nuclear Technology*, Issue 1, Vol. 201, pp. 1-12.

Devell, L., et al. (1986), "Initial Observations of Fallout from the Reactor Accident in Chernobyl", *Nature*, Vol. 321, pp. 192-193.

DOE (1993), "Historical Summary of the TMI-2 Core Debris Transportation Campaign", DOE-ID-10400.

DOE (1988), "Historical Summary of Fuel and Waste Handling and Disposition at TMI-2, 1980-1988", EGG-2529.

DOE (1990), "TMI-2 Lessons Learned by the DOE, 1979-1990", DOE-ID-10276.

ENSREG (2012), "Peer Review Report – Stress Tests Performed on European Nuclear Power Plants", www.government.nl/documents/reports/2012/06/20/peer-review-report-on-stress-tests-performed-on-european-nuclear-power-plants.

EPRI (1990a), "Decontamination Experience During the Cleanup of Three Mile Island Unit 2", EPRI Report NP-7157.

EPRI (1990b), "The Cleanup of Three Mile Island Unit 2, A Technical History: 1979 to 1990", EPRI Report NP-6931, http://svcf.jp/pdf/Three_Mile_Iland2NP-6931.PDF.

EPRI (1992a), "Final TMI-2 Technology Transfer Progress Report", EPRI Report TR-100643.

EPRI (1992b), "TMI-2 Post-Accident Data Acquisition and Analysis Experience", EPRI Report NP-7156.

EPRI (1992c), "TMI-2 Waste Management Experience", EPRI Report TR-100640.

Fichot, F. and L. Carénini (2016), "Improving Reliability of In-vessel Melt Retention Strategy for Severe Accident Mitigation", *Proceedings of the ICONE 24 Conference*, Charlotte, NC, US, 26-30 June 2016,

Frolov, V.V. (1996), "Abnormal Incident in 'Shelter' Object of ChNPP on 27-30 June 1990", *Atomnaya Energia*, Vol 60, Issue 3, pp. 216-219 – in Russian (Фролов В.В. Аномальный инцидент 27–30 июня 1990 г. в объекте «Укрытие» Чернобыльской АЭС. Атом. Энергия. 1996, т. 60, вып. 3, с. 216-219).

Government of Slovenia (2010), "National Emergency Response Plan for Nuclear and Radiological Accidents" (2010, revised end of 2015), www.sos112.si/eng/tdocs/plan_nuclear _radiological.pdf.

IAEA (2016), *Safety of Nuclear Power Plants: Design, Specific Safety Requirements*, No. SSR-2/1 (Rev. 1), IAEA, Vienna.

llyin, L.A. and O.A. Pavlovskij (1988), "Radiological Consequences of the Chernobyl NPS Accident and Measures Taken to Mitigate their Impact", *Atomnaya Energiya*, Vol. 65, pp. 119-129 (in Russian).

IAEA (2014), *Lessons Learned Worldwide in the Cleanup and Decommissioning of Accidents*, IAEA-NW-T-2.7, IAEA, Vienna.

IAEA (2006), *Environmental Consequences of the Chernobyl Accident and their Remediation: Twenty Years of Experience*, Report of the Chernobyl Forum Expert Group "Environment", IAEA, Vienna.

IAEA (2001), *Risk Management: A Tool for Improving Nuclear Power Plant Performance*, TECDOC-1209, IAEA, Vienna.

IAEA (1992), *Cleanup and Decommissioning of a Nuclear Reactor after a Severe Accident*, IAEA-TRS-346, IAEA, Vienna.

IAEA (1991a), *Catalogue of Methods, Tools and Techniques for Recovery from Fuel Damaged Events*, IAEA-TECDOC-627, IAEA, Vienna.

IAEA (1991b), *Management of Severely Damaged Nuclear Fuel and Related Waste*, IAEA-TRS-321, IAEA, Vienna. NEA (2016), *Management of Radioactive Waste after a Nuclear Power Plant Accident*, NEA No. 7305, OECD, Paris.

IAEA (1986), *Summary Report of the Post-Accident Review Meeting on the Chernobyl Accident*, Safety Series No. 75-INSAG-l, IAEA, Vienna.

llyin, L.A. and O.A. Pavlovskij (1988a), "Radiological Consequences of the Chernobyl Accident in the Soviet Union and Measures Taken to Mitigate their Impact", *Nuclear Power Performance in Safety* (Proc. Conf. Vienna, 1987), Vol. 3, pp.149-166, IAEA, Vienna.

llyin, L.A. and O.A. Pavlovskij (1988b), "Radiological Consequences of the Chernobyl NPS Accident and Measures Taken to Mitigate their Impact", *Atomnaya Energiya*, Vol. 65, pp. 119-129 (in Russian).

ISTC (2000), "Study of Phenomena and Dynamics of Fuel Contained Material Disintegration in 'Shelter' Object under the Influence of External Factors", Final ISTC Report No. 3806, Chernobyl, p. 43 – in Russian (Изучение механизмов и динамики разрушения топливосодержащих материалов объекта «Укрытие» под действием внешних факторов. Отчет о НИР (заключительный) МНТЦ «Укрытие», № 3806, Чернобыль, 2000. – 43 с.).

ISP NPP (2007), "Investigation of Physicochemical Properties of Fissile Nuclear Hazardous Materials which Influence Radiation and Environmental Safety of the ' Shelter' Object", final report of Institute of NPP Safety Problems of National Academy of Sciences of Ukraine, No 3970, Chernobyl, p. 248 – in Russian (Изучение физико-химических свойств ядерно опасных делящихся материалов объекта «Укрытие», влияющих на степень ядерной, радиационной и радиоэкологической безопасности. Отчет (заключительный) ИПБ АЭС НАНУ № 3970, Чернобыль, 2007 г., 248 с.).

Ivanov, E.A. et al. (1994), "Environmental Radioactive Contamination by Am-241 from the Chernobyl NPP Accident", *Atomnaya Energia*, Vol. 77, Issue 2, pp. 140-145 – In Russian (Иванов Е. А., Носовский А. В., Хамьянов Л. П. и др. Радиоактивное загрязнение окружающей среды Am-241 вследствие аварии на Чернобыльской АЭС// Атом, энергия,- 1994.-Т. 77.-Вып. 2.-С. 140-145).

Izrael, Y. (Ed.) (1998), *Atlas of Radioactive Contamination of European Russia, Belarus and Ukraine*, Federal Service for Geodesy and Cartography of Russia, Moscow.

Izrael, Y.A. et al. (1990), "Chernobyl: Radioactive Contamination of Natural Media", Gidrometeoizdat, Moscow, p. 296 – In Russian (Израэль Ю.А., Вакуловский С.М., Ветров В.А. и др. Чернобыль: радиоактивное загрязнение природных сред / Под ред. Ю. А. Израэля. – М.:Гидрометеоиздат, 1990. – 296 с).

Japanese Government, (2011) ""Roadmap towards Settlement of the Accident at Fukushima Daiichi Nuclear Power Station, TEPCO" Step 2 Completion Report", available on www.tepco.co.jp/en/press/corp-com/release/betu11_e/images/111216e4.pdf (page 71 onwards).

Kachalovsky, E.V. (1989), "Experience of the Government of the Ukrainian SSR in Cleaning up the Accident at the Chernobyl Nuclear Power Plant", *Proceedings of the Symposium on Recovery Operations in an Event of Nuclear Accident or Radiological Emergency*, 6-10 November, p. 235-248, IAEA, Vienna.

Kashparov, V.A., (1994), "Inhalation of Radionuclides During Agricultural Work in Areas Contaminated as a Result of the Chernobyl Reactor Accident", *J. Aerosol Science*, Vol. 25, No. 5, p. 761-767.

Kluchnikov, A.A, et al. (2011), "Shelter Object: 1986/2011, On the way to transformation", National Academy of Sciences of Ukraine, Institute of NPP Safety Problems, Chernobyl, p. 288 – in Russian (Объект «Укрытие»: 1986-2011. На пути к преобразованию / А. А. Ключников, В. А. Краснов, В. М. Рудько, В. Н. Щербин; - НАН Украины, Ин-т проблем безопасности АЭС. - Чернобыль (Киев, обл.) : Ин-т проблем безопасности АЭС, 2011. - 288 с.).

Klyuchnikov, A.A., Krasnov V.O. (2015) “Development of means improving radiation and ecological safety of new safe confinement and shelter object”, final report of Institute of NPP Safety Problems of National Academy of Sciences of Ukraine, No 4003, Vol.2, Chernobyl.

Kochetkov, O.A. et al. (1989), “Accumulation of the Exposure Dose During Operations to Deal with the Consequences of a Serious Radiation Accident”, *Proceedings of the Symposium on Recovery Operations in an Event of Nuclear Accident or Radiological Emergency*, 6-10 November, p. 151-166, IAEA, Vienna.

Kononovich, A.L. et al. (1994), “Estimation of Radiation Conditions of Ground Water in the Area of Chernobyl NPP”, *Atomnaya Energia*, Vol. 77, Issue 5, pp. 386-391, – in Russian (Кононович А. Л., Носовский А. В., Осколков Б. Я. и др. Оценка радиоактивного состояния подземных вод в районе Чернобыльской АЭС // Атом, энергия. - 1994. - Т. 77. - Вып. 5. - С. 386-391).

Krasnov, V.O. and V. Khan (2015), “Monitoring of Radioactive Dust and LFCM State in Destroyed Unit 4 ChNPP”, International Seminar on Decommissioning of Fukushima Daiichi NPP, Tokai – Mura, Japan, 9-13 November 2015.

Krasnov V.O. et al. (2016) ”Object Shelter: 30 years after the accident”. Ukrainian National Academy of Sciences, Institute of NPP Safety Problems, Chernobyl, Kiev region, 512 p., 2016 (In Ukrainian: Об’єкт «Укриття»: 30 років після аварії: монографія / В. О. Краснов, А. В. Носовський, В. М. Рудько, В. М. Щербін ; НАН України, Ін-т проблем безпеки АЕС. – Чорнобиль (Київ. обл.) : Ін-т проблем безпеки АЕС, 2016. – 512 с.).

Krupny, V.I., (1996), “Shelter Object: Yesterday, Today and Tomorrow, Shelter Object in 10 Years”, main research results, National Academy of Sciences of Ukraine, Chernobyl, pp. 57-77 – In Russian (Купный В.И. Объект «Укрытие»: вчера, сегодня, завтра // Объект «Укрытие» - 10 лет. Основные результаты научных исследований. - Чернобыль, 1996. - С. 57-77).

Lebedev, G.V. and V.F. Shikalov (1995), “The Sub-criticality Measurement of the Fuel Containing Masses of the Shelter”, Moscow, p. 24 (Preprint of the Kurchatov IAE, No.5876/5).

Legasov, V.A. et al. (1988), “One Year after the Chernobyl NPS Accident”, *Atomnaya Energiya*, Vol. 64, Issue 1 (1988), pp. 13-19 (in Russian).

Lewis, H.W. (1986), “The Accident at the Chernobyl NPP and its Consequences: USSR State Committee on the Utilization of Atomic Energy”, *Environment: Science and Policy for Sustainable Development*, Vol. 28, Issue 9.

NDA (2011), “NDA Prioritisation – Calculation of Safety and Environmental Detriment Scores”, Doc No EGPR02 Rev 6, April 2011.

NDF (2016), “Technical Strategy Plan 2016 for Decommissioning of Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.”, www.dd.ndf.go.jp/en/strategic-plan/book/20170322_SP2016eFT.pdf.

NEA (2015a), “Ability of Current Advanced Codes to Predict In-Vessel Core Melt Progression and Degraded Core Coolability: Benchmark Exercise on Three Mile Island-2 Plant – Final Report”, NEA/CSNI/R(2015)3, OECD Publishing, Paris.

NEA (2015b), “Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF Project) – Phase I Summary Report”, NEA/CSNI/R(2015)18, OECD Publishing, Paris.

NEA (2014a), “Accident Management Insights after the Fukushima Daiichi NPP Accident”, NEA/CNRA/R(2014)2, OECD Publishing, Paris.

NEA (2014b), “Occupational Radiation Protection in Severe Accident Management”, NEA/CRPPH/R(2014)5, www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/CRPPH/R(2014)5&docLanguage=En, OECD Publishing, Paris.

NEA (2013), “Update Knowledge Base for Long Term Core Cooling Reliability”, NEA/CSNI/R(2013)12, OECD Publishing, Paris.

NEA (2010), *Nuclear Fuel Behaviour under RIA Conditions*, NEA No. 6847, OECD Publishing, Paris.

NEA (2000), "Impact of Short-term Severe Accident Management Actions in a Long-term Perspective", OECD/NEA/CSNI/R(2000)8.

Nosovsky, A.V. et al. (2006), "Chernobyl NPP accident: Experience of Overcoming Lessons learned", Tekhshka, Kiev, p. 264 – in Russian (Авария на Чернобыльской АЭС: Опыт преодоления. Извлеченные уроки / А. В. Носовский, В. Н. Васильченко, А. А. Ключников, Б. С. Пристер; Под ред. А. В. Носовского. - К.: Техшка, 2006. - 264 с. + цв. вкл. на 32 с. - (Безопасность атомных станций)).

NRC (2016), "Three Mile Island Knowledge Management Digest – Recovery and Cleanup", NUREG/KM-0001, Supplement 1, https://tmi2kml.inl.gov/HTML/Page1.html.

OECD (2002), "Chernobyl: Assessment of Radiological and Health Impacts, Updates of Chernobyl Ten Years On", OECD Publishing, Paris.

Pasukhin, E.M. (1997), "About Efficiency of Filling the Reactor Pit of the 4th Unit of Chernobyl NPP During the Accident of April 26, 1986", *Radiochemistry*, Vol. 39, Issue 4, pp. 375-378, – In Russian (Пазухин Э.М. Об эффективности засыпки шахты реактора 4-го энергоблока ЧАЭС при аварии 26 апреля 1986 г. // Радиохимия. - 1997. - Т. 39, вып. 4. - С. 375-378).

Pazukhin, E.M. (2008), "Interaction of Lava-like Fuel-containing Masses with Structural Concrete in the Active Phase of the Chernobyl Accident", *Radiochemistry*, Vol. 50(3), pp. 324-331.

Pazukhin, E.M. et al. (2006), "Fuel at Upper Levels of the Destroyed Fourth Block of Chernobyl NPP, Refining the Formation Scenario of the Polychromatic Ceramics", *Radiochemistry*, Vol. 48(5), pp. 522-534.

Pazukhin, E.M., A.A. Borovoi and K.G. Rudya (2002), "Effect of α-Ray Self-Irradiation on Stability of Lava-Like Fuel-Containing Masses from the Fourth Block of the Chernobyl NPP", *Radiochemistry*, Vol. 44(6), pp. 558-563.

Pazukhin, E.M. et al. (2002), "Investigation of LFCM Materials Sampled from Different Depths in Lava Flows, Problems of Nuclear Power Plants Safety and of Chernobyl", 9, pp. 66-75.

Pazukhin, E.M. (1994), "Lava-like Fuel-containing Masses of 4th Unit of Chernobyl NPP: Topography, Physical-chemical Properties, Scenario of Formation", *Radiochemistry*, Vol. 36(2), pp. 97-142.

Petelin, G.I. et al. (2003), "Hot Nuclear Fuel Particles of the Source Term in Retrospective Assessment of Accidental Phenomena in the 4th Unit of ChNPP", *Radiochemistry*, Vol. 45, Issue 3, pp. 278-281 – in Russian (Петелин Г.И., Зимин Ю.И., Тепикин В.Е и др. Горячие частицы ядерного топлива чернобыльского выброса в ретроспективной оценке аварийных процессов на 4-м блоке ЧАЭС // Радиохимия. - 2003. - Т. 45, № 3. - С. 278 -281).

Rasmussen, N. et al. (1975), *Reactor Safety Study*, WASH-1400, NRC, Washington DC.

Rimsky-Korsakov, A.A. et al. (2009), "Study of Accidental Source Term of Unit 4 of Chernobyl NPP", *Proc. of Radium Institute Named after V.G. Khlopin*. Vol 14, pp. 10-22 – in Russian (Римский-Корсаков А.А., Федотов П.И., Дубасов Ю.В. и др. Исследование выброса аварийного реактора 4-го блока Чернобыльской АЭС. Труды Радиевого института им. В.Г. Хлопина, т. XIV, 2009, с. 10-22).

Shestopalov, V.M. et al. (2000), "Radioactive Contamination of Ground Waters in the Area of 'Shelter' Object, Water Exchange in Hydrogeological Structures and Chernobyl Disaster", Kiev, Part 1, pp. 87-125 – in Russian (Шестопалов В.М., Кухаренко Д.Е., Панасюк Н.И. и др. Радиоактивное загрязнение подземных вод в районе объекта «Укрытие»: Науч. статья «Водообмен в гидрогеологических структурах и Чернобыльская катастрофа»/Отв. ред. акад. НАН Украины В.М. Шестопалов. – Киев, 2000, часть 1, с. 87-125.).

Shiryaev, A.A. et al. (2016), "Physico-chemical Properties of Chernobyl Lava and their Destruction Products", *Progress in Nuclear Energy*, Vol. 92, pp. 104-118.

SNSA (2015), "The Ionising Radiation Protection and Nuclear Safety Act", ZVISJV, revised in October 2015, Official Gazette of the Republic of Slovenia No. 67/2002.

SNSA (2011), "Rules on operational safety of radiation or nuclear facilities", JV9 (2011, in process of revision), Official Gazette of the Republic of Slovenia, No. 85/2009 and 87/2011.

SNSA (2010), “Rules on the use of potassium iodide”, SV9 Official Gazette of the Republic of Slovenia 142/2004.

SNSA (2009a), “Rules on radiation and nuclear safety factors”, JV5 (2009, in process of revision), Official Gazette of the Republic of Slovenia, No. 92/2009.

SNSA (2009b), “Rules on radioactivity monitoring”, JV10 (2009, in process of revision), Official Gazette of the Republic of Slovenia, No. 20/2007 and 97/2009.

SÚJB (2003), “Decree of the SÚJB No. 185/2003 Coll., on Decommissioning of Nuclear Installation or Category III. or IV. Workplace”, www.sujb.cz/en/legal-framework/old-nuclear-law.

SÚJB (2002), “Decree of the SÚJB No. 318/2002 Coll., on Details of Emergency Preparedness of Nuclear Facilities and Workplaces with Ionising Radiation Sources and on Requirements on the Content of On-Site Emergency Plan and Emergency Rule”, amended in Decree SÚJB No. 2/2004 Coll, www.sujb.cz/en/legal-framework/old-nuclear-law.

SÚJB (1997), “Decree of the SÚJB No. 215/1997 Coll., on Criteria for Siting Nuclear Facilities and Very Significant Ionising Radiation Sources”, www.sujb.cz/en/legal-framework/old-nuclear-law.

WENRA RL (2014), “WENRA Safety Reference Levels for Existing Reactors, Update in Relation to Lessons Learnt from TEPCO Fukushima Dai-ichi Accident, September 2014”, www.wenra.org/media/filer_public/2016/07/19/wenra_safety_reference_level_for_existing_reactors_september_2014.pdf.

WENRA RL (2008), “WENRA Reactor Safety Reference Levels, January 2008”, www.wenra.org/media/filer_public/2012/11/05/list_of_reference_levels_january_2008.pdf.

NEA PUBLICATIONS AND INFORMATION

The full catalogue of publications is available online at www.oecd-nea.org/pub.

In addition to basic information on the Agency and its work programme, the NEA website offers free downloads of hundreds of technical and policy-oriented reports. The professional journal of the Agency, NEA News – featuring articles on the latest nuclear energy issues – is available online at www.oecd-nea.org/nea-news.

An NEA monthly electronic bulletin is distributed free of charge to subscribers, providing updates of new results, events and publications. Sign up at www.oecd-nea.org/bulletin.

Visit us on Facebook at www.facebook.com/OECDNEA or follow us on Twitter @OECD_NEA.

OECD/NEA PUBLISHING, 2 rue André-Pascal, 75775 PARIS CEDEX 16

Long-Term Management and Actions for a Severe Accident in a Nuclear Power Plant

As the Fukushima Daiichi nuclear power plant (NPP) accident illustrates, many challenges have to be faced in maintaining safety over the long term in a damaged NPP following a severe accident. These comprise maintaining and monitoring a stabilised and controlled state of the damaged plant; implementing provisions against further failures; evaluating the plant damaged state from a physical and radiological standpoint and ranking related risks; preparing and achieving fuel retrieval (either fuel assemblies stored in spent fuel pools or fuel debris from damaged reactors); and managing safely plant recovery and accident waste. All these actions are to be conducted protecting plant personnel from radiation exposure.

This status report reviews knowledge and experience gained through long-term management (LTM) of the Three Mile Island, Chernobyl and Fukushima Daiichi accidents, by identifying and ranking main issues and knowledge gaps. It also reviews the existing regulations and guidance, practices, technical bases and issues considered in member countries of the Nuclear Energy Agency regarding LTM of a severely damaged nuclear site. Finally, it proposes recommendations and areas for future investigation to enhance LTM of an NPP as regards necessary knowledge and provisions development, particularly for the optimisation of management of contaminated cooling waters.

Nuclear Energy Agency (NEA)
46, quai Alphonse Le Gallo
92100 Boulogne-Billancourt, France
Tel.: +33 (0)1 73 21 28 19
nea@oecd-nea.org **www.oecd-nea.org**

NEA No. 7506