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Safety Research Opportunities Post-Fukushima

Initial Report of the Senior Expert Group

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Table of contents

List of figures

List of tables

Executive summary

One of the imperatives following the accident at the Fukushima Daiichi nuclear power station is for the nuclear science and industry communities to ensure that knowledge gaps in nuclear safety are identified and that research programmes to address these gaps are being instituted.

In recognition of broad international interest in additional information that could be gained from post-accident examinations related to Fukushima Daiichi, Japan recommended to the Committee on the Safety of Nuclear Installations (CSNI) in June 2013 that a process be developed to identify and follow up on opportunities to address safety research gaps. Consequently, a Senior Expert Group (SEG) on Safety Research Opportunities post-Fukushima (SAREF) was formed. The members of the group are senior technical experts from technical support organisations, nuclear regulatory authorities and Japanese organisations responsible for planning and execution of Fukushima Daiichi decommissioning. The domain of interest for the group is activities that address safety research knowledge gaps and also the needs of Fukushima Daiichi decommissioning. SEG on SAREF identified areas where these two interests intersect or overlap, and activities that could be undertaken to generate information of common benefit. Briefly, these were categorised into the following broad areas:

- severe accident progression:
  - in-vessel phenomena;
  - primary system and reactor pressure vessel (RPV) failure;
  - ex-vessel phenomena;
  - containment failure and venting;
  - hydrogen distribution and combustion;
  - fission product (FP) behaviour and source term;
  - pool scrubbing.
- system, structure and component (SSC) performance and condition:
  - salt water and concrete debris effects;
  - mission time and system survivability (further broken down into cables and sealing, instrumentation, reactor core isolation cooling (RCIC) system, relief valves and piping, and environmental degradation of metallic components).
- recovery phase:
  - long-term accident management (AM) and recovery;
  - debris and waste management.

Several other areas were also considered to be of interest, but no post-accident examinations were identified which would address them. For completeness, these areas are:

- external effects, multi-unit risk and loss of ultimate heat sink;

- robustness of electrical systems;
- spent fuel pools;
- human performance;
- seismic response (of building and components).

For each of these areas, SEG on SAREF assessed the status of the existing knowledge base and identified what information could be gained during Fukushima Daiichi decommissioning to address any knowledge gaps. Ongoing R&D activities in the CSNI member countries that could address the gaps and complement information from Fukushima Daiichi were also identified. SEG on SAREF assessed safety research interest on a consensual basis, as high, medium or low based on the extent of the knowledge gap and on the potential for information from Fukushima Daiichi to address the gap.[1]

The group also assessed what information could be gained to support decommissioning planning and execution, and, again on a consensual basis, whether the decommissioning interest was high, medium or low, based on the importance of the information to making decisions regarding the safe execution of decommissioning activities. Decommissioning needs that were given priority were:

- evaluation of the distribution of fuel debris and fission products remaining in primary containment vessel (the PCV); for the purpose of risk assessment;
- safety and risk assessments for planning and implementing fuel debris retrieval;
- design and development of technologies and systems for fuel debris, retrieval and storage and other decommissioning processes.

In terms of determining which examinations should be recommended, those that are of high safety and high decommissioning interest are clearly higher priority than those with lesser interest. At the same time, SEG on SAREF recognised that it may be possible to examine reactor systems and components or obtain samples that, while of low interest for decommissioning, are of high safety research interest and can be gained without an adverse impact on decommissioning. These examinations cannot easily be identified in advance and must be undertaken as the opportunity arises (i.e. they can be characterised as opportunistic examinations). Examples are visual examinations that contribute to an understanding of RCIC system performance, or to understanding the performance of primary system safety relief valves (SRVs).

SEG on SAREF decided to focus on areas that are of high decommissioning interest, and either high or medium safety research interest, and on mission time and system survivability where opportunistic examinations could provide information. These areas are (with the safety research interest and decommissioning interest shown in brackets):

- severe accident progression:
  - in-vessel phenomena (H-H);
  - ex-vessel phenomena (H-H);
  - containment failure and venting (H-H);
  - FP behaviour and source term (H-H).

1. Note that the assessment of interest does not necessarily correlate with the general priority of the area from the perspective of importance to severe accidents, nor to understanding the accident at Fukushima. There are areas that are of high priority from either of these perspectives for which not much information can be obtained during Fukushima Daiichi decommissioning, and therefore the overall safety research interest is assessed as low or medium.

- stress corrosion cracking performance and conditions:
  - mission time and system survivability (H-L).
- recovery phase:
  - long-term accident management and recovery (M-H).

Giving some consideration to challenges associated with examinations at Fukushima Daiichi and the current status of the preliminary plans for decommissioning, SEG on SAREF is recommending undertaking activities to evaluate four long-term considerations:

- in-vessel phenomena and RPV failure;
- ex-vessel phenomena;
- FP database compilation and measurement;
- mission time and system survivability.

For the first long-term consideration, RPV failure has been added to in-vessel phenomena recognising that while its decommissioning interest is medium, there may be opportunities to investigate the source of any RPV failure without much adverse impact on the decommissioning schedule. Moreover, mechanisms for RPV failure may be closely tied to in-vessel phenomena such as melt formation in the lower head, and to ex-vessel phenomena such as melt relocation.

The first long-term considerations will be addressed by fuel debris retrieval and characterisation from inside and outside reactor pressure vessels that will directly characterise the areas of in-vessel and ex-vessel phenomena. The ex-vessel examination will also address, in part, containment failure and venting. An additional activity will be to compile a database on fission product measurements to address gaps in understanding of fission product behaviour and source term. The last long-term consideration (mission time and system survivability) is based on opportunities for examinations of reactor components that may arise as decommissioning proceeds.

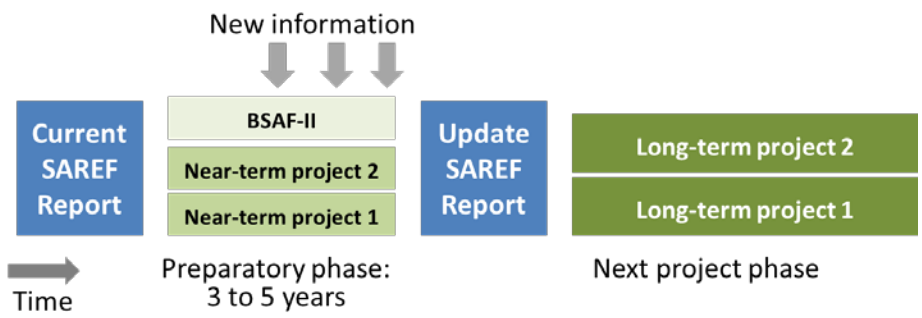
The SEG on SAREF also determined that it would be beneficial to undertake some near-term activities to provide additional information for planning long-term activities and is recommending two near-term proposals:

- preparatory studies for fuel debris analysis;
- examinations inside reactor buildings (RBs) and PCVs and water sampling.

The proposal for preparatory studies on debris analysis is aimed at ensuring processes are in place for the long-term debris retrieval and analysis activities. The second proposal covers examinations inside the reactor buildings and water sampling, and addresses, in part, the safety research area on long-term AM and recovery. The near-term proposals will also ensure ongoing interaction between international safety research experts and experts from Japanese organisations relevant to decommissioning of Fukushima Daiichi nuclear power station (NPS) and co-ordination with other activities, notably the Benchmark Study of the Accident at Fukushima Daiichi Nuclear Power Station (BSAF) project.

Over the next few years, additional information will be gained from decommissioning, examinations at Fukushima Daiichi, development of sampling methodology and BSAF modelling studies. This information will have an influence on the ongoing near-term activities, and it will be important to have a mechanism for sharing the information between relevant organisations in Japan and international experts in reactor safety. New information will require updated assessments. As a result, SAREF activities are foreseen to be ongoing, with the assessment driving near-term activities, feedback from new information driving revised assessments and new activities being recommended during each assessment phase. One mechanism that allows for ongoing interaction between international experts and relevant organisations within Japan are joint annual meetings and/or workshop be held by inviting international experts including those in the CSNI working groups where updates from the near-term activities and from the decommissioning activities at Fukushima Daiichi will be presented. The SEG on SAREF also recommends that the CSNI monitor the ongoing process of SAREF. (See the following figure.)

Overview over the near- and long-term activities following the SAREF report



List of abbreviations and acronyms

AC	Alternating current
ACE	Advanced containment experiment
ADS	Automatic depressurisation system
AECL	Atomic Energy Canada Limited
AM	Accident management
ASTEC	Accident Source Term Evaluation Code
ATTIHLA	Measurements of corium thermodynamic properties, experimental facility at CEA (France)
BDBE	Beyond design basis event
BSAF	Benchmark Study of the Accident at Fukushima Daiichi Nuclear Power Station
BWR	Boiling water reactor
CEA	French Alternative Energies and Atomic Energy Commission
CFD	Computational fluid dynamics
CNRA	Committee on Nuclear Regulatory Activities (NEA)
CORA	Experimental investigations on severe fuel damage, core degradation and quench at KIT (Germany)
CORCON	Code for the analysis of core concrete interaction
CRD	Control rod drive
CRGT	Control rod guide tubes
CRIEPI	Central Research Institute of Electric Power Industry (Japan)
CSNI	NEA Committee on the Safety of Nuclear Installations
DBA	Design basis accidents
DEFOR-A	Debris Bed Formation and Agglomeration experiments at KTH (Sweden)
DENOPI	Experimental project to acquire data on the physical phenomena associated with a spent fuel pool loss-of-cooling and loss-of-coolant accidents
DOE	United States Department of Energy

D/W	Dry well
ELAP	Extended loss of alternating current power
ENACCEF	Hydrogen experimental facility at CNRS/ICARE (France)
FOREVER	Failure Of REactor VEssel Retention, experimental facility at KTH (Sweden)
EPDM	Ethylene propylene diene monomer
FCI	Fuel coolant interaction
FCVS	Filtered containment venting systems
FEM	Finite Element Model
FLEX	Diverse and flexible mitigation capability
FP	Fission product
EPRI	Electric Power Research Institute
HYCOM	Hydrogen combustion test project at IBRAE (Russia)
HYMIX	Hydrogen mixing test facility
IAEA	International Atomic Energy Agency
IRID	International Research Institute for Nuclear Decommissioning (Japan)
IRSN	Institute for Radiological Protection and Nuclear Safety (France)
ISTP	International Source Term Program
ITU	Institute for Transuranium Elements (Germany)
JAEA	Japan Atomic Energy Agency
KIT	Karlsruhe Institute of Technology (Germany)
KROTOS	Fuel-coolant interaction experiments
LHF	Lower head failure
LOCA	Loss of coolant accident
LWR	Light-water reactor
MAAP	Modular Accident Analysis Program
MASCA	Experiment program on corium compositions prototypical of nuclear power reactors
MCCI	Molten core concrete interaction
MELCOR	Fully integrated, engineering-level severe accident computer code developed by Sandia National Laboratories (United States)
METCOR	Primary system experimental facility

METI	Ministry of Economy, Trade and Industry (Japan)
MIRE	Mitigation of Releases to the Environment in the event of a nuclear accident project
MISTRA	Large CFD experimental facility belonging to the CEA (France)
MOCKA	Large-scale MCCI experimental facility at KIT (Germany)
MSL	Main steam line
NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
NEA	Nuclear Energy Agency
NITI	Alexandrov Research Institute of Technology
NPP	Nuclear power plants
NPS	Nuclear power station
NRA	Nuclear Regulation Authority (Japan)
NRC	Nuclear Regulatory Commission (United States)
NUGENIA	International non-profit-making association, Next Generation II & III Association
OECD	Organisation for Economic Co-operation and Development
OLHF	NEA Sandia Lower Head Failure Project
PANDA	Large-scale thermal-hydraulics test facility at PSI (Switzerland) for containment system behaviour investigations and large-scale separate effect tests
PAR	Passive autocatalytic recombiner
PEARL	"Programme Expérimental Analytique sur le Renoyage de Lits de Débris", debris reflooding
PHEBUS	International research programme to improve the understanding of the phenomena occurring during a core meltdown accident in a light water reactor conducted in France between 1988 and 2010
PHITS	Particle and Heavy Ion Transport code System, general purpose Monte Carlo particle transport simulation code
PRELUDE	Small-scale prototype test facility for PEARL
PROGRES	Research programme on degraded core coolability at IRSN (France)
PULiMS	"Pouring and underwater liquid melt spreading" experimental facility at KTH (Sweden)
PCV	Primary containment vessel
PSA	Probabilistic safety assessment

PWR	Pressurised water reactor
PWROG	PWR Owners Group
QUENCH	Experimental investigations on core heat-up and quench
RASPLAV	Primary experimental facility on prototypic materials
RB	Reactor building
RCIC	Reactor core isolation cooling
RCS	Reactor coolant system
REMCOD	REmelting of Multi-COmponent Debris project at KTH (Sweden)
RPV	Reactor pressure vessel
SA	Severe accident
SAFEST	Severe Accident Facilities for European Safety Targets
SAM	Severe accident management
SAMG	Severe Accident Management Guideline
SARNET	European Severe Accident Network of Excellence
SAREF	Safety Research Opportunities post-Fukushima
S/C	Suppression chamber
SCC	Systems, Components and Structures
SEG	Senior Expert Group
SES	Steam Explosion Stratified experimental facility at KTH (Sweden)
SERENA	Steam Explosion Resolution for Nuclear Applications (NEA)
SFP	Spent fuel pool
SICOPS	Large scale MCCI experimental facility at Areva GmbH (Germany)
SGTS	Standby Gas Treatment System
SNL	Sandia National Laboratory (United States)
SOAR	State-of-the-Art Reports
SOARCA	State of the Art Consequence Assessment
SOCRAT	System of Codes for Realistic Analysis of Severe Accidents developed by IBRAE (Russia)
SPARC	Suppression pool aerosol removal code
SRV	Safety relief valve

SSC	Systems, structures, components
TB	Turbine building
TDAFW	Turbine driven auxiliary feed water
TEPCO	Tokyo Electric Power Company Holdings, Inc. (Japan)
THAI	Thermal-hydraulics, Hydrogen, Aerosols, Iodine
TIP	Traversing In-Core Probe (in-core instrumentation tube)
TROI	Ex-vessels steam explosion experimental facility at KAERI (Korea)
VERCORS	Analytical experimental programme on the release of FP and actinides from an irradiated fuel rod during a severe accident in a PWR at CEA (France)
VERDON	Experimental facility on FP release at CEA (France)
VTT	VTT Technical Research Centre of Finland Ltd as research and technology company
VULCANO	MCCI experimental facility at CEA (France)
WECHSL	Code for the analysis of core concrete interaction
WGAMA	Working Group on Analysis and Management of Accidents (NEA)
WGIAE	Working Group on Integrity and Ageing of Components and Structures (NEA)

1. Introduction

1.1. Background

One of the imperatives for the nuclear science and industry communities following the accident at the Fukushima Daiichi nuclear power station (NPS) following the Great East Japan earthquake on 11 March 2011 is to ensure that knowledge gaps in nuclear safety are identified and consequently, corresponding research programmes to address these gaps are being instituted.

Following a proposal from Japan, the Committee on the Safety of Nuclear Installations (CSNI) initiated the "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant" (BSAF Project), phase 1 in 2012. The study had the following objectives [1]:

- to analyse accident progression of Fukushima Daiichi NPS utilising a common information database;
- to improve the understanding of severe accident (SA) phenomena which took place during the accident through comparison of participants' analysis results and measured plant data;
- to validate the SA analysis codes by using data gathered during the decommissioning process in order to improve the SA codes' models and to reduce uncertainties in the analyses;
- to contribute analysis results on accident progression, status in the reactor pressure vessels (RPVs) and primary containment vessels (PCVs), and status of debris distribution in order to assist the planning of debris removal at Fukushima Daiichi.

A total of 16 organisations of eight countries (France, Germany, Korea, Russia, Spain, Switzerland, the United States and Japan) participated.

An important contribution from BSAF was the elucidation of the accident evolution in the three units of the Fukushima Daiichi NPS that were in operation at the time of the accident, based on the best possible database. The first result from the BSAF project (phase 1) was a summary of the hypothesised evolution of the accident based on simulation results [1] (see Section 2). Based upon this evolution, open questions were then identified that can be resolved by post-accident examinations during decommissioning operations.

Phase 2 of the BSAF project began in early 2015. Its objectives are:

- To provide information and analysis results on the SA progression, fission product behaviour, source term estimation and comparison with measured plant data within the first 3 weeks in Fukushima Daiichi units 1 to 3 respectively to support safe and timely decommissioning at Fukushima Daiichi NPS.
- To raise the understanding of SA phenomena which took place during the accident, through comparison with participants' analysis results and with measured plant data.

- To contribute the above results to improvement of methods and models of the SA codes applied in each participating organisation, in order to reduce uncertainties in SA analysis and validate the SA analysis codes by using data measured through the decommissioning process.

It is evident that the decommissioning operation as currently described in the Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) strategic plan [2] will benefit from the outcomes of BSAF, which will suggest both the probable status of the three units and provide an estimate of the configuration of debris and melted material in- and outside of the RPV.

Taking the unfortunate consequences of the Fukushima Daiichi accident into full consideration, international collaboration should be organised in order to further facilitate the decommissioning activities of Fukushima Daiichi. The opportunity to gather unique insights into the severe accident phenomenology of boiling water reactors (BWRs) should also be utilised to further enhance nuclear safety, especially since there is much less BWR-specific knowledge of accident phenomena available than for pressurised water reactors (PWRs). Proper identification of relevant topics is key. Hence, the identification of such research topics constitutes a major element of the work of the Senior Expert Group (SEG) on Safety Research Opportunities post-Fukushima (SAREF).

However, without continued interaction with the relevant Japanese organisations, such an endeavour will remain without effect. SEG on SAREF addresses this important aspect by proposing establishing mechanisms for continued communication of the relevant experts beyond the mission of SAREF.

1.2. Mandate of SEG on SAREF

Recognising the broad international interest in information that could be gained from post-accident examinations and other activities related to Fukushima Daiichi, Japan recommended, in the 53rd meeting of the Committee on the Safety of Nuclear Installations (CSNI) in June 2013, that the CSNI develop a process to identify and follow up on opportunities to address safety research gaps. The CSNI members supported Japan's recommendation and set up a SEG on SAREF. The CSNI asked SEG on SAREF to develop its mandate that include the objectives and deliverables by itself and report back to the CSNI. Then, SEG on SAREF held the 1st meeting in November 2013 and discussed and developed it.

The members of SEG on SAREF are senior technical experts from technical support organisations and nuclear regulatory authorities representing Canada, Finland, France, Germany, Italy, Japan, Korea, Norway, Russia, Spain, Sweden, Switzerland, the United Kingdom and the United States. In addition, the lead Japanese organisations for the planning and execution of Fukushima Daiichi decommissioning are the members of SEG on SAREF.

Recognising that post-accident examinations should not impede Fukushima Daiichi decommissioning, and should preferably provide information on the status of the reactors that is helpful, the SAREF initiative has a twofold objective. That is, the Senior Expert Group is to develop a process for identifying and following up on opportunities to address safety research gaps while also supporting Japan in achieving safe and timely decommissioning and remediation.

Based on above discussion, the objectives of SEG on SAREF were proposed as follows: *To propose to the CSNI a process to identify and follow up on opportunities for addressing safety research gaps and advancing safety knowledge, based on information from Fukushima Daiichi.*

This objective will also support Japan in achieving safe and timely decommissioning.

The major deliverables of SEG on SAREF is a CSNI report that identifies safety research areas of common interest and recommends to the CSNI safety research activities including their relative priority.

The proposed mandate was approved in the 54th meeting of the CSNI in December 2013.

1.3. Overview of SAREF activities and report outline

To achieve the objective, SEG members have considered safety research opportunities in the areas of:

- severe accident progression:
  - in-vessel phenomena;
  - primary system and RPV failure;
  - ex-vessel phenomena;
  - containment failure and venting;
  - fission product behaviour and source terms;
  - hydrogen distribution and combustion;
  - pool scrubbing.
- system, structure and component (SSC) performance and condition:
  - salt water and concrete debris effects;
  - mission time and system survivability (further broken down into cables and sealing, instrumentation, reactor core isolation cooling (RCIC) system, relief valves and piping, and environmental degradation of metallic components);
- recovery phase:
  - long-term accident management (AM) and recovery;
  - debris and waste management.

Several other areas were also considered to be of interest, but no post-accident examinations were identified which would address them. For completeness, these areas are:

- external effects, multi-unit risk and loss of ultimate heat sink;
- robustness of electrical systems;
- spent fuel pools;
- human performance;
- seismic response (of buildings and components).

For each area the group identified the safety research interests with respect to safety knowledge gaps which need to be addressed for nuclear safety enhancement in the NEA member countries, the decommissioning interest, potential examinations including challenges (cost, timing, dose, etc.) and feasibility. Related R&D activities, either ongoing or planned, were also outlined. The safety experts assessed the safety research interest as high, medium and low; and the decommissioning representatives provided a similar rating for decommissioning interest.

Based on this information, the experts discussed and identified a number of safety research proposals that could be recommended to the CSNI.

Research proposals should typically address high priorities (e.g. debris sampling), but feasibility in terms of technical details, cost, etc. may not be known. Therefore, the proposal has to be conceptual and it will need to be kept updated. These proposals are called "**long-term considerations**" the details of which should be discussed in the future when sufficient information becomes available on the inside of the reactor buildings (RBs), PCVs, reactor pressure vessels (RPVs) and so forth.

In some cases, there needs to be "**near-term projects**" which can start in the near term as a preparatory phase such as to collect and analyse basic information and track and update "long-term considerations", maintain information channels between the CSNI and Japanese relevant organisations, check and test the feasibility of transportation, examination, etc. of the samples to be taken, and so forth.

The group's output is documented in this report; Chapter 2 describes the current status of the damaged units at Fukushima Daiichi NPS; Chapter 3 summarises safety research areas of common interest; Chapter 4 summarises the safety research activities recommended as short-term projects; Chapter 5 summarises those as long-term considerations; Chapter 6 supplies conclusions and recommendations. The appendix contains detailed information compiled by the SEG members on all safety research areas of interest as mentioned above.

2. Current status of the units 1 to 3 at Fukushima Daiichi end of 2015 and summary of the scenarios

2.1. Introduction to the severe accidents

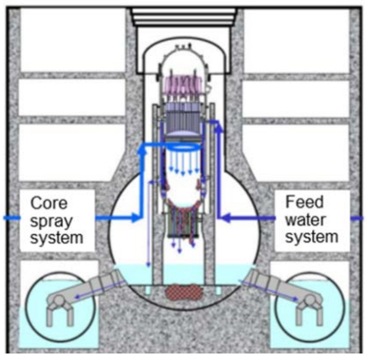
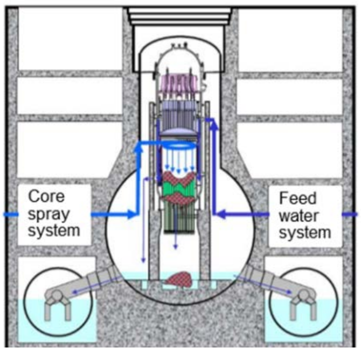
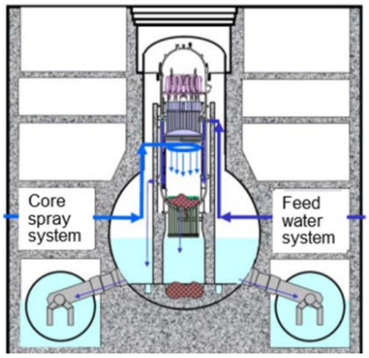
There are different sources of information available with regards to the current status of the Units at Fukushima Daiichi; three of them are mentioned herewith. On the one hand there is the information provided by the NEA BSAF project, phase I [1]. The objective was to analyse the accident progression of Fukushima Daiichi NPS within the first 6 days of the accident utilising the common information database to contribute analysis results on accident progression, status in the reactor pressure vessels (RPVs) and primary containment vessels (PCVs), and status of fuel debris distribution to assist in the planning of debris removal at Fukushima Daiichi. The second information source is the "Technical Strategic Plan 2015 for Decommissioning of the Fukushima Daiichi NPS of Tokyo Electric Power Company" published by NDF [2] in April 2015. Both documents are based on data and information published by the Tokyo Electric Power Company Holdings, Inc. (TEPCO). The plant status and the accident progression are reported continuously by TEPCO in its progress reports [3] on "The evaluation of the situation of the cores and containment vessels of Fukushima Daiichi NPS units 1 to 3 and the examination into unsolved issues in the accident progression". The 4th progress report was published in December 2015.

In the report published by the NEA [1], the cause of the accident is summarised as follows: "The Tohoku-Chihou Taiheiyou-Oki earthquake (Great East Japan earthquake) occurred on 11 March 2011 at 14:46 (Japan time zone). The three operating reactors were safely shut down at 14:47 followed by system isolation by main steam isolation valve closure. From TEPCO's observation of the plant's operation status, the main equipment with important safety control functions is assumed to have been operable after the earthquake. The earthquake was followed by a tsunami wave which, by reconstruction through videos and on-site post-measurements, is estimated to have reached the height of 14 m and which caused a large scale disaster in the Pacific Ocean coastal areas. The earthquake was designated with an intensity scale of 9.1, on the Richter scale, the index used to indicate the scale of an earthquake. It was the fourth largest ever observed in the world and the largest ever recorded in Japan. The tsunami waves which hit the Fukushima Daiichi NPS flooded not only the level 4 m above the Onahama Port construction level (O.P. 4 m), where seawater pumps had been installed, but also the 10-m level, where key buildings had been constructed. Consequently, motors and electrical equipment were flooded and important systems, such as emergency diesel generators and power panels, were directly or indirectly affected and disabled. The result for units 1 to 3 was the loss of the ultimate heat sink, loss or drastic decrease of plant instrumentation and the total inability or exceeding difficulty to operate the safety systems for cooling the reactor cores." Consequently, although they had been successfully shut down, units 1 to 3 lost functions related to cooling and ultimately experienced severe accidents.

2.2. Status as reported in NDFs Strategic Decommissioning Plan

From information provided by the NDF Strategic Plan [2], the following information on the status of the Fukushima Daiichi units 1 to 3 is drawn:

Table 1. Plant status and estimated fuel debris locations

Results from plant investigation		
Unit 1	Unit 2	Unit 3
- D/W water level approx. 3 m from PCV bottom. - S/C is mostly filled with water. - Leakage from the sand cushion piping identified. - Leakage from the expansion-joint cover of the vacuum break line connected to the wet well venting piping confirmed. - High dose rate (several Sv/h) spot in the southeast area of the reactor building 1st floor.	- D/W water level approx. 30 cm from PCV bottom. - S/C room water level close to the centre, almost the same water level to that of the torus. - No evidence of leakage in the torus upper section. - Photo of RPV pedestal inside taken from the opening confirms the structure of RPV lower region, which confirms portions of the RPV lower region are intact.	- D/W water level approx. 6.5 m from PCV bottom. (estimated from the pressure differential of D/W and S/C) S/C mostly filled with water. - Leakage from the expansion-joint of the main steam piping confirmed.
Estimated fuel debris location		
- Almost all molten fuel dropped down to the RPV lower plenum and no fuel debris remains in the core. - Dropped fuel debris into the lower plenum fallen on the RPV pedestal bottom. - Dropped fuel debris in the pedestal bottom flew outside of the pedestal (probable attack to the shell).	- Some molten fuel dropped to the RPV lower plenum and on the pedestal floor, and the remaining is retained in the core (no fuel debris estimated outside the pedestal).	
- 	- 	

2.3. Status as reported at the end of the NEA BSAF Project, phase 1

The NEA BSAF project, phase 1, was finished in 2015, with publication of their summary report [1]. Based on their simulation results, the participants have discussed possible accident scenarios of the Fukushima Daiichi NPS units 1 to 3. In these scenarios, several keywords are used to characterise how the related information was developed. The employed nomenclature refers to the following words:

- **consensual**: unique agreement is reached by all contributed simulations;
- **likely**: majority of the contributed simulations compute an event/trend/scenario;
- **possible**: minority of the contributed simulations compute an event/trend/scenario;
- **uncertain**: there is no unique definition on a certain event/trend/scenario and results are uniformly distributed among several different options.

For unit 1 the summary [1] contains among others the following information with regard to the core degradation and plant status:

- *Uncertainty* still exists related to the core degradation phase itself, and the wide variability of the results on the RPV depressurisation phase does not narrow the mechanism of RPV/reactor coolant system (RCS) failure and depressurisation. It is a *consensual* agreement that in unit 1 the pressure boundary failed during the core degradation phase but the effect on the RPV pressure history depends on the assumption of the location of the failure. In particular, main steam line (MSL) failure and safety relief valve (SRV) seizure tend to depressurise the reactor before the lower head failure, while penetrations and SRV gasket leakage tend to maintain a relatively large RPV pressure and lead to a high melt jet ejection into the pedestal region of the containment. The calculated high pressure lower head failure denotes the *possibility* of direct containment heating, which might have implications on the decommissioning activities.

  The detailed mechanism of core degradation and timing of events (rod collapse, melt relocation within the core and lower core plate failure or shroud melt and RPV lower head breach) are difficult to reconstruct from the available code calculations. The main reasons derive from differences in the details in the relocation models of the SA codes employed, the different nodalisation schemes adopted and uncertainty in physical process of the core material relocation inside the RPV due to lack of representative experiments. However, even though the calculated timing and sequences differ slightly and are *uncertain* to some extent, *consensus* was reached that a large fraction of the fuel, control blades and core structures (i.e. channel box, fuel supports and core plate) were destroyed/molten and relocated into the lower plenum (either through the shroud or directly from the core region through openings in the core plate) initially. Thereafter, the RPV failed *possibly* by either pipe melting and/or by RPV wall melt-through, which allowed a large amount of corium debris to move into the cavity, where the molten core concrete interaction (MCCI) phase started. The reactor might have reached this configuration within 15 hours after scram. Until this time, no water was successfully injected into the RCS/RPV or PCV.

- *Consensus* has been reached that the first significant pressure rise in the D/W (PCV) was associated with the RPV/RCS boundary failure described above, discharging steam and hydrogen into the PCV. It is likely that until that time (around ten hours from scram) a PCV failure did not occur. The time of PCV failure is still *uncertain*. It is *possible* that the PCV failed after failure of the lower head occurred, resulting in the second large pressure peak in the PCV, even though the location and the reasons have not been further substantiated by the analyses. The timing of this event is very scattered among results, but this result is supported by the high dose rate detected at the main gate on 12 March at 10:40. Further substantiation is expected as the result of evaluations undertaken during the second phase of the BSAF project.

  Water injection by means of fire trucks is *likely* to not have been effective until around 80 hours after scram. It is *likely* that at the time of water injection by fire trucks, the further core degradation progression (in the core and the pedestal) was terminated and long-term stable conditions were reached. This is not yet shown by all of the analysis with regard to the MCCI process which some analyses show is ongoing at this stage. It is clear that all simulations use large simplifications in the modelling of the debris released into the pedestal. The MCCI phase in the pedestal started after RPV failure, but before the first water injection by fire truck at 4:00 on 12 March and in all the simulations it is predicted to be ongoing at the end of the simulations six days after scram. Extensive oxidation of the metals contained in the cavity and of the reinforcing bars in the basemat is expected. The region of concrete erosion is supposed to be of a size comparable to the pedestal

> wall region in the radial direction, with the possibility that the pedestal walls had been weakened by the interactions with the corium. The cavern erosion in the vertical direction is *consensually* predicted not to be extended to the liner and no failure predicted in the current project phase.

It is not clear whether a larger amount of hydrogen was generated during core oxidation or during the MCCI phase, nevertheless the simulations are relatively coherent in the order of magnitude of the volume of generated flammable gases around the time of explosion of the reactor building (not an issue in phase 1 of the project). At the time of explosion indeed it is *likely* that the total mass of hydrogen generated was around 2 000 kg (in core production plus MCCI) while 1 000 kg of CO are expected to have been generated during the corium concrete interaction phase.

The qualitative description of the expected status of the plant is almost identical to the one shown in [2].

For unit 2 the summary [1] contains the following information with regard to the core degradation and plant status:

> During the uncontrolled operation of the RCIC system the D/W pressure increased slower than expected based on a simple energy balance. All simulations *consensually* needed to assume an energy loss from the S/C which is generally associated with tsunami water that entered the torus room after the inundation. (…) As a general conclusion regarding the prediction of the pressure transient in the containment, SA codes with more detailed nodalisation and multidimensional thermal hydraulic modelling features are necessary in order to simulate long station black out accidents where non-uniformity such as thermal stratification might be reached in the pool.
>
> There is *consensus* that water injection by means of fire truck started immediately after the RPV depressurisation and it is considered to have been effective in injecting water into the recirculation line, even though the quantity reaching the RPV was lower than the measured value of discharged water from the pump. It is *likely,* however, that the water level decreased to around the bottom of fuel (BAF) after the depressurisation, with a consequent increase of the fuel temperature. The attempt to reflood the core and the resulting zirconium-water reaction accelerates the temperature excursion and the core started melting. The evidence of the attempt to reflood is *likely* because of the repeated pressure transients visible in the RPV pressure trace. In the most positive scenario it is *possible* that from around 20% to 70% of the initial core inventory melted and relocated into the lower head with complete retention of the fuel. However, it also *possible* that almost the totality of the core mass relocated down into the pedestal with consequent onset of MCCI. It should be mentioned that the majority of the simulations does not predict RPV failure and relocation of core debris into the pedestal with a configuration close to what schematically is presented in from NDF report [2]. Given the simplicity of the models associated with the lower head, as hinted in unit 1, this is not indicating the impossibility of the RPV failure but that the MCCI phase is *likely* not expected for this unit and that the majority of the core debris are expected to be located into the RPV, both core region and lower head or as crust on lower head penetrations.

For unit 3 the summary [1] contains among others the following information with regard to the core degradation and plant status:

> The measured PCV pressure showed a much faster increase especially during RCIC system operation and SRV cycling which initially could not be reproduced by the computations. The reasons for the rapid S/C pressure increase were not identified and a wide debate exists. The two most widely accepted scenarios can be summarised as follows: the *possibility* of a steam bypass (incomplete condensation)

in the S/C due to the creation of stratification and local hot spots around the sparger in the pool, and the *possibility* of having a direct leak from the RPV to the D/W. From the available computations, a final answer cannot be derived but many useful insights can be obtained. In particular, the subdivision of the S/C in several nodes (contrary to the usual single node approximation) allows for calculation of the correct pressure trend using several assumptions. A single node approximation would need correlations to determine the extent of incomplete condensation depending on the temperature, whereas the assumption of a steam leak from the RCS into the D/W does not require any special treatment of the PCV into multiple nodes.

The rate of the temperature increase in the core varies among the different simulations at high RPV pressure; however, it is *likely* that the core degradation began around 40 hours after scram. This *possibility* is coherent with the evidence of neutron detection around this time. Around 9:00 on 13 March, the reactor was depressurised in about two minutes, *likely* by activation of the automatic depressurisation system, or a *possible* creep rupture of the MSL. At this time, water injection by fire trucks was started. As in the other two units, several leak paths might have played a critical role to decrease the effective mass of water sent to the RPV. The percentage of the water sent to the core is unknown. From the computations, it can be inferred that an assumed water injection rate of around 30% of the average mass flow rate to the core is *likely* to save the lower head, while lower values are plausible to trigger vessel breach.

Different from unit 1 and unit 2, some divergence in the prediction of the initiation of the core melt exists caused by *uncertainties* related to the degradation of high pressure core injection and the resulting core water level, and therefore related to the core temperature excursion phase during SRV cycling prior to RPV depressurisation. A considerable difference exists in the computations as to whether core relocation started before or after RPV depressurisation occurred.

Two plausible scenarios can be identified at the end of the time period of the computations, as a result of the several uncertainties still existing for this unit. (...)The first scenario predicts the RPV remained intact and the melt was retained into the lower head, so that MCCI did not begin and the pressure transient in the D/W depended on the balance between corium quenching in the lower head and PCV venting. In this scenario it is *possible* that from 40% to 60% of the initial mass inventory melted. The second scenario is an RPV breach and core debris transfer to the pedestal with onset of MCCI and gas generation as is shown in the NDF report [2]. In the second scenario the flammable gas generation is shown to be considerably larger than the in-vessel case. Indeed, in the first scenario there is *consensus* that between 1 000 and 1 500 kg of hydrogen were produced while in the second between 2 500 and 3 500 kg of hydrogen and more than 4 000 kg of CO are generated. The ex-vessel case predicts a total debris mass of from around 60% to 100% of the initial core inventory.

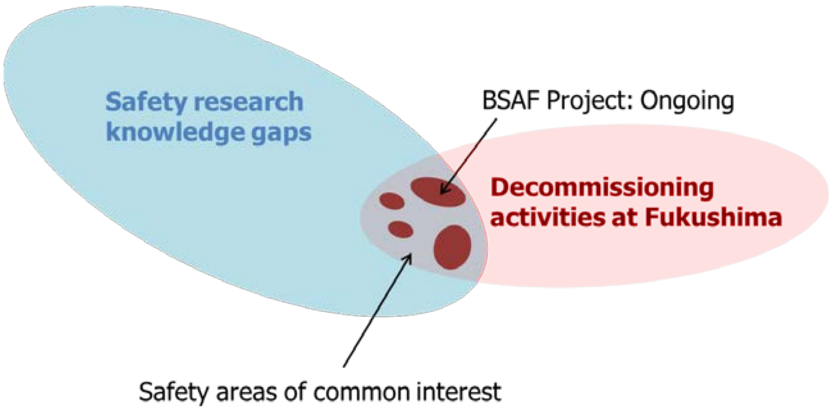
The information provided related to the multiple operations of PCV venting has some *uncertainties* especially as the area of the opened valves is almost unknown. Most partners modelled a first venting due to the RPV depressurisation and the PCV pressure increases until the burst disc in the already opened venting line failed. Thereafter, the timing of the venting processes in the analyses shows *uncertainty* in relation to which vent was effectively performed. In particular, *consensus* exists that the first two ventings were performed while no agreement exists for the later four vents.

In the conclusion of the NEA BSAF project, phase 1 it is mentioned that, regarding the actual progression of the accident in the three units and the expected current plant status, several common understandings were reached in the computations. Nevertheless, various differences and uncertainties exist in the computations regarding the extent of core degradation in the units, the possibility of in-vessel or ex-vessel scenarios in unit 2 and unit 3, the effectiveness of alternative water injection in all the units and the actual S/C venting operation in unit 3. It has been underlined however, that differences are introduced once the core geometry is altered during the relocation process and in the attempt to stabilise the plant with external water injection and venting. The differences are attributed to each code's modelling approaches. An attempt has been made to identify the influence of the employed models during relocation. The main physical modelling uncertainties are the RPV failure mechanism at high core temperatures (e.g. penetration failure or creep rupture), computation of the debris surface area once the core changes configuration, creation of possible paths for the debris to move from the core region to the lower head through the core lower structures and core plate, failure mechanisms of the lower head and mechanisms for leak/failure of the containment system. Further work is needed to include physics insights from new experimental and analytical activities as well as to try to close knowledge gaps through results gained from the Fukushima Daiichi units decommissioning process.

3. Identification of safety research areas of common interest and recommended examinations

The domain of interest for SEG on SAREF is activities that address safety knowledge gaps and that address the needs of Fukushima Daiichi decommissioning. This intersection is illustrated in Figure 1. On the one hand, there are a number of safety research knowledge gaps, particularly those arising from the accidents at Fukushima Daiichi – shown as the blue oval. On the other hand, there is information required to ensure safe and timely decommissioning of the reactors at Fukushima Daiichi – shown as the pink oval. Then there are areas where activities generate information that is of common interest for addressing safety research knowledge gaps, and that support decommissioning activities – shown as the overlap between the two ovals. One example is the BSAF project that is generating information on the predicted accident progression in the Fukushima Daiichi reactors to support decommissioning decisions and to improve to severe accident modelling codes.

Figure 1. Illustration of the SAREF domain of interest



With this framework in mind, SEG on SAREF identified the following areas where there may be research opportunities of common interest to addressing safety knowledge gaps and to support Fukushima Daiichi decommissioning:

- severe accident progression:
  - in-vessel phenomena;
  - primary system and RPV failure;
  - ex-vessel phenomena;
  - containment failure and venting;
  - fission product behaviour and source term;
  - pool scrubbing.

- SSC performance and condition:
  - salt water and concrete debris effects;
  - mission time and system survivability (further broken down into cables and sealing, instrumentation, reactor core isolation cooling (RCIC) system, relief valves and piping, and environmental degradation of metallic components).
- recovery phase:
  - long-term accident management (AM) and recovery;
  - debris and waste management.

Several other areas were also considered to be of interest, but no post-accident examinations were identified which would address them. For completeness, these areas are:

- external effects, multi-unit risk and loss of ultimate heat sink;
- robustness of electrical systems;
- spent fuel pools;
- human performance;
- seismic response (of building and components).

For each of these areas, SEG on SAREF assessed the status of the existing knowledge base and identified what information could be gained during Fukushima Daiichi decommissioning to address any knowledge gaps. Ongoing R&D activities in the CSNI member countries that could also address the gaps or would complement information from Fukushima Daiichi were also reviewed. SEG on SAREF assessed safety research interest, on a consensual basis, as high, medium or low based on the significance of the knowledge gap and on the importance of information from Fukushima Daiichi to address the gap[1]. For example, the existing knowledge base on severe accident evolution in BWRs would benefit greatly from information gained from Fukushima Daiichi, which is therefore of high interest. At the same time, for some areas it may not be possible to gain useful information from the damaged Fukushima Daiichi reactors. A case in point is the area of hydrogen distribution and combustion. While the hydrogen combustion events at Fukushima Daiichi were unique, it is not possible to determine what the hydrogen distribution was, and it is difficult to distinguish the effects of the hydrogen combustion from other damage mechanisms. Moreover, there are a number of research activities underway in the CSNI member countries that are addressing open questions related to hydrogen distribution and combustion, therefore, the safety research interest is medium.

Also for each of these areas, SEG on SAREF reviewed with experts from Japanese organisations what information could be gained to support planning and execution of decommissioning, in particular fuel debris retrieval at the Fukushima Daiichi NPS. Again, the SEG assessed on a consensus basis the decommissioning interest of high, medium or low reflecting the importance of the information to being able to make decisions or to undertake activities safely. Decommissioning needs information that supports:

1 It has to be stressed that this assessment of interest does not necessarily line up with the general priority of the area from the perspective of importance to severe accidents, nor to understanding the accident at Fukushima. There can be areas that are of high priority from these perspectives for which not much information can be obtained during Fukushima decommissioning, and therefore the overall safety research interest is assessed as low or medium.

- evaluation of the distribution and risk assessment of fuel debris and fission products remaining in PCV;
- safety and risk assessment in planning and implementing fuel debris retrieval (structural integrity, re-criticality, radiative material dispersion and dose exposure, etc.);
- design and development of technologies and systems for fuel debris retrieval and storage and other decommissioning processes.

For example, getting detailed information on the final state of reactor internals and the location of the damaged fuel (in-vessel vs ex-vessel) is of high safety research interest and of high interest for decommissioning as it is critical to deciding how to retrieve the fuel debris while ensuring worker's safety, and hence is of high interest to decommissioning. On the other hand, knowing more about the phenomenology of pool scrubbing in the wetwell water pool would be of high safety research interest, as the current models in codes are based on old experiments, are rather conservative, but the phenomenon is of high importance for source term predictions for BWR. Here there exists only a low interest to decommissioning, as the decommissioning work may start years after the accident and the contamination in the water pool will be decreased.

The details of the assessments of the status of the existing knowledge base, potential examinations at Fukushima Daiichi, ongoing R&D activities, safety research interest, and decommissioning interest are provided in Appendices A.1-A.4. The results are summarised in Table 2, with the safety research interest, decommissioning interest and potential examinations summarised for each research area.

In terms of determining which examinations should be recommended, those that are of high safety and high decommissioning interest are clearly higher priority than those with lesser interest. At the same time, SEG on SAREF recognised that it may be possible to examine reactor systems and components or obtain samples that, while of low interest for decommissioning, are of high safety research interest and can be gained without adversely impacting decommissioning. These examinations cannot easily be identified in advance and must be undertaken as the opportunity arises (i.e. they can be characterised as opportunistic examinations). Examples are visual examinations that contribute to an understanding of RCIC system performance, or to understanding the performance of primary system safety relief valves (SRVs).

The SEG on SAREF decided to focus on the areas that are of high decommissioning interest, and either high or medium safety research interest, and on mission time and system survivability where there could be opportunistic examinations. These areas are (with the safety research interest and decommissioning interest shown in brackets):

- severe accident progression:
  - in-vessel phenomena (H-H);
  - ex-vessel phenomena (H-H);
  - containment failure and venting (H-H);
  - fission product behaviour and source term (H-H).
- stress corrosion cracking performance and conditions:
  - mission time and system survivability (H-L).
- recovery phase:
  - long-term AM and recovery (M-H).

To address the knowledge gaps and decommissioning needs associated with the above areas, a series of conceptual outline of potential examinations and sampling regimes is outlined in the Appendices. While this information is desirable, SEG on SAREF recognises that other factors also need to be considered before deciding on the examinations to be undertaken at Fukushima Daiichi. In some cases, it may not prove feasible to perform an examination, or it may be very challenging in terms of impact on decommissioning, additional cost or worker dose. At this point, it is not possible to properly assess the feasibility or challenges for the detailed examinations and sampling regimes. For example, a more detailed plan for decommissioning of the reactors is required before the impacts in terms of schedule and cost can be determined. Likewise, more information on the final core end states and contamination distribution is needed to assess potential worker doses. SEG on SAREF therefore recommends that as decommissioning evolves and more information becomes available, feasibility should be confirmed and challenges should be characterised and addressed, while recognising that continuous support from the international community to address the challenging decommissioning of the Fukushima Daiichi will encourage a number of local stakeholders in Fukushima with a hopefully foreseeable future of safe and stable conditions at the Fukushima Daiichi site.

With this background, SEG on SAREF is recommending four long-term research considerations (see Section 5 for details). There are two long-term research considerations on fuel debris retrieval and characterisation from inside and outside reactor pressure vessels that directly address the areas of in-vessel and ex-vessel phenomena. For the first long-term consideration, RPV failure mechanisms will be investigated with in-vessel phenomena recognising that while its decommissioning interest is medium, there may be opportunities to investigate the source of any RPV failure without much adverse impact on the decommissioning schedule. Moreover, mechanisms for RPV failure may be closely tied to in-vessel phenomena such as melt formation in the lower head, and to ex-vessel phenomena such as melt relocation. Similarly for the second long-term consideration, the mechanisms for containment failure and venting are somewhat tied to ex-vessel phenomena such as molten core concrete interaction. There is a long-term research consideration to compile a database on fission product measurements that addresses fission product behaviour and source term. The last long-term research consideration is aimed at opportunities for examinations of reactor components that may arise as decommissioning proceeds to address areas of high safety research interest. Recognising that these long-term research considerations require additional information and development of decommissioning planning and processes, two near-term research proposals are also recommended (see Section 4 for more details). The first proposal on preparatory studies on debris analysis is aimed at ensuring processes are in place for the long-term consideration on debris retrieval and analysis. The second proposal covers examinations inside the reactor buildings and water sampling, and addresses, in part, the safety research area on long-term AM and recovery. The near-term research proposals will also ensure ongoing interaction between international safety research experts and experts from Japanese organisations relevant to Fukushima Daiichi decommissioning, and co-ordination with other activities, notably the BSAF project.

Table 2. Summary of SEG assessments

Research area	Safety research interest	Decommissioning interest	Potential examinations	Challenges/feasibility	Ongoing R&D activities
A.1. *Severe accident progression*					
A.1.1. In-vessel phenomena	High: knowledge gaps in BWR severe accident progression, corium formation and relocation within RPV, key to improve understanding and modelling of late severe accident progression.	High: the final location and composition of material in the reactor is of high interest; re-criticality issue needs consideration; support for decisions on how to decommission damaged reactors.	End-state of core materials, evidence of material interactions and relocation paths into lower plenum, evidence of stratification in corium.	Costly, from both a schedule and dose impact perspective. Visual inspections will provide a baseline of information, detailed debris sampling may be difficult. Pre-work required on simulated debris.	Complements ongoing modelling activities (e.g. BSAF). Experimental work on in-vessel severe accident phenomena ongoing in Europe, US, Canada, Japan and Korea, mostly related to PWR.
A.1.2. Primary system and RPV failure	High: addresses gaps in reactor system, RPV or component failure mechanisms of BWRs; key to improve understanding and modelling of late SA progression.	Medium: knowing the reactor failure points may be helpful in determining the melt relocation and fission product release into the containment; this contributes to decommissioning decision.	What component failures led to depressurisation (MSL, SRV or others). RPV failure location. Behaviour of vessel penetrations and interaction with melt. Samples from reactor vessel.	Costly and possibly time consuming as it may be difficult to identify system failure. New information may become available as RPV is removed from containment. Visual inspections helpful, but limited.	Complements ongoing modelling activities (e.g. BSAF), and analytical study (Components and Structures under Severe Accident Loading – COSSAL). Experimental work being done by the Institute of Applied Energy (IAE) and the Korea Atomic Energy Research Institute. JAEA developing analytical methods on BWR's RPV failure.
A.1.3. Ex-vessel phenomena	High: knowledge gaps in melt relocation from RPV, in interaction with BWR typical ex-vessel structures, in molten core concrete interaction and melt coolability by water; is important to SAM strategies and mitigation technologies (e.g. melt flooding); key to improve understanding and modelling of ex-vessel severe accident progression; focus is on melt relocation and MCCI and not on fuel coolant interaction or direct containment heating phenomena.	High: the final location and composition of core material in the containment, and impacts on ex-vessel structures and concrete basemat are of high interest; Re-criticality issue needs consideration; support for decisions on how to reactors.	With regard to core material similar to in-vessel phase: amounts, morphologies and distributions of core material, interaction with ex-vessel structures, stratification of melt pool and erosion of containment basemat, mechanisms for melt decommissioning damaged cooling.	Costly, can impact decommissioning. schedule, challenges with dose and contamination. Visual inspections will provide a baseline of information, accessing, mapping and sampling ex-vessel melt may be difficult, pre-work required on simulated debris.	Some out-reactor information available, but limited for BWR core materials. Relevant experimental ex-vessel corium research (MCCI) being done in Europe, US, Japan. The Nuclear Regulation Authority and JAEA developing methodologies to evaluate ex-vessel debris coolability. JAEA analyses concrete samples from reactor buildings of units 1&2.
A.1.4. Containment failure and venting	High: knowledge gaps in containment failure mechanisms under SA conditions and high containment pressure and temperature loads, could provide information to SAM and modelling.	High: there is an ongoing reliance on containment structures during decommissioning, particularly important to establish boundary condition for debris removal.	Containment leakage paths for water and gas, integrity of key regions such as containment penetrations and bellows, defects in head flange sealing.	Costly and potentially dose intensive. Some sampling and examinations for leak paths may be very difficult.	R&D related to improvement of SA codes or ongoing modelling activities (e.g. BSAF) could complement. Some information available from national programmes on testing of penetrations, bellows and gaskets. Inspection technology being developed for detecting and sealing leakage points. Further PCV thermal-hydraulics experiments being done in various countries.

Research area	Safety research interest	Decommissioning interest	Potential examinations	Challenges/feasibility	Ongoing R&D activities
A.1.5. FP behaviour and source term	High: Supports public dose assessment, EQ, Reducing uncertainty in dose significant FPs.	High: Key to actual public and worker dose. Need to evaluate characteristics of FP residues and associated dose. Will also need to assess potential releases during debris retrieval.	Off-site release data available, need in-plant distribution and overall speciation, water chemistry.	Accessibility of deposition data and isotope decay since the accident will contribute to uncertainty in analysis. Some limited information from dose measurements.	Supporting info. from NEA joint projects (STEM2, BIP3,THAI3) and national programmes. Modelling efforts such as BSAF Phase 2 are important.
A.1.6. Pool scrubbing	High: Fission product (FP) transport through the reactor coolant system and into the containment (especially through the suppression pool), as well as potential FP release from the containment, are of major importance when assessing the release of activity to the environment.	Low: phenomena not of decom-missioning interest, but info support estimates of FP deposits.	FP distributions in water pools and on containment surfaces.	Determination of surface distributions (including bottom of pools) will be time consuming and costly. Elapsed time and redistribution makes estimation of earlier conditions hard.	Some work on FP retention in water pools ongoing (NEA Thermal-hydraulics, Hydrogen, Aerosols, Iodine [THAI-3] Project, the EU Passive and Active Systems towards enhanced Severe Accident source term Mitigation [PASSAM] Project, and various national programmes).
A.1.7. Hydrogen distribution and combustion	Medium: Gaps in knowledge on hydrogen combustion in the containment and other areas of the power plant are key to proper assessment of hydrogen risk.	Low: Information on hydrogen combustion mechanisms and phenomenology may be of limited use due to other affecting factors, even though there is potential risk of hydrogen residue in piping.	Estimates of total hydrogen, migration paths, combustion sites, consequent damage through visual observation and plant specific info.	Could affect schedule, personnel dose if pursued separately from other activities. May be quite feasible to gain additional information on damage as decommissioning proceeds.	Some data already collected for unit 4. Large amount of separate effects data available, and new information coming from NEA joint projects (THAI-3, Hydrogen Mitigation Experiments for Reactor Safety [HYMERES]) and various national programmes.
A.2. *SSC performance and condition*					
.A.2.1. Salt water and concrete debris effects (primary system and spent fuel pool)	Low: Seawater (or raw water) may be used for emergency cooling in other plants affecting corrosion (& radiolysis and behaviour of FPs).	Medium: Long-term corrosion mechanisms may be important to decommissioning, also effects of sea water on SSCs and fuel debris and formation of biological deposits.	Analysis of water pools over time, corrosion samples for key reactor components and systems, samples of fuel for leaching, determination of distributions of precipitates.	Costly, since materials sampling and assessing precipitates could possibly have schedule impacts. Difficult to assess corrosion rates, and changes with time, particularly as now using deionised water.	- Some separate effects tests being conducted in Japan - Determination of corrosion rates - Spent fuel pool issue is closed for salt water, but concrete debris is making pool water alkaline
A.2.2. Mission time and system survivability					
A.2.2.1. Cable and Sealing	High: Ability of containment seals and electrical systems to continue to perform its function as designed is key to SAM.	Low: While ensuring the containment function of the PCV boundary is crucial for planning and implementing fuel debris retrieval methodology, research on seal material degradation mechanisms is not critical, as the potential of leakage from seals should be considered in any way. Damaged cables and seals could be repaired/replaced if needed for decommissioning activities.	Visual inspection, sampling and testing of cables and seals in various places to determine effects of SA conditions and continued exposure in a harsh environment.	Estimating exposure conditions (dose, temp., etc.) is difficult.	Separate effects tests conducted in various national programmes (effect of fluence, temperature, etc.).

Research area	Safety research interest	Decommissioning interest	Potential examinations	Challenges/feasibility	Ongoing R&D activities
A.2.2.2. Instrumentation	High: Key instruments are required for SAM.	Low: Instruments replaced or not continued to be used.	Instrumentation status and modes of failure (visual inspection, sampling etc.).	Costly and dose intensive, but may become available as decommissioning proceeds. Information may not be available (instrument destroyed) or be difficult to inspect, so data from Fukushima may be of limited use.	Various national programmes taking measure to address SA survivability of instrumentation.
A.2.2.3. RCIC	High: RCIC system performance had a major influence on accident progression at Fukushima.	Low: RCIC system no longer required.	Status of RCIC systems in various units, including determination of possible damage and failure mechanisms.	Data from Fukushima Daiichi may be of limited use. Probably need to wait until decommissioning removes RCIC systems, since it is currently immersed in radioactive water and is difficult to access.	Out-reactor testing and analytical model development being considered in various national programmes.
A.2.2.4. Relief valves and piping	High: performance of valves and / or piping failure key to RPV depressurisation.	Low: potentially important for establishing a water-tight boundary.	Status of relief valves and integrity of relief piping (visual inspection, photos/videos, etc.).	May be able to visually inspect piping.	
A.2.2.5. Environmental degradation of metallic components	Medium: unique opportunity to examine materials exposed to environmental conditions that cannot be reproduced in labs.	Low: no interest concerning decommissioning.	Visual inspection, sampling and testing of metallic components in various places to determine effects of SA conditions and continued exposure in a harsh environment.	Estimating exposure conditions (dose, temp., etc.) is difficult, but sample examination could become fairly feasible as decommissioning proceeds in a long-term process.	Research projects with the aim of determining materials behaviour under different environmental conditions are being carried out in many countries.
A.3 Recovery phase					
A.3.1. Long-term AM and recovery	Medium: some interest in how to manage contaminated water, structural integrity.	High: need to ensure continued safety of damaged reactors and minimise radioactive releases. Behaviour of contaminated water is of particular interest.	Water sampling, status of key systems (e.g. containment).	Costly, but required for safe decommissioning. Water sampling may be challenging.	Japan evaluating integrity of Fukushima Daiichi spent fuel pools, and developing evaluation method of dose rate distribution in PCV. The NEA Working Group on Analysis and Management of Accidents is working on long-term AM.
A.3.2. Fuel debris and waste management	Low: some interest in managing damaged fuel or components (handling, packages etc.), but info specific to Fukushima.	High: options needed for safe-handling and long-term management.	Characterisation of various debris and their radionuclides (particles, metallic and oxidised melts, relatively intact components, etc.).	Costly, but a prerequisite for decommissioning. Preliminary work performed on simulated debris. Solutions for handling and management should be straightforward.	Research on debris characterisation and criticality being done in Japan, Some work being done under NEA Radioactive Waste Management Committee.

Research area	Safety research interest	Decommissioning interest	Potential examinations	Challenges/feasibility	Ongoing R&D activities
A.4. General/already addressed					
A.4.1. External effects and multi-unit risk and loss of ultimate heat sink	Medium: Many countries have interest in estimating multi-unit risk including external effects and loss of ultimate heat sink.	Low: Information not really needed for decommissioning.	Nothing	-	The NEA Working Group on Risk Assessment organised a workshop on multi-unit probabilistic safety assessment, and the NEA Working Group on External Events will start a work on external flooding. Some countries have national programmes on these topics.
A.4.2. Robustness of electrical systems	Low: except for systems identified as important for mission time and system survivability.	Low: Information not really needed for decommissioning.	Nothing	-	The CSNI established new working group on safety of electrical systems.
A.4.3. Spent fuel pools	Low: no significant damage in spent fuel pools.	Low: no significant damage in spent fuel pools.	Nothing	-	Air-spent fuel pool in Nugenia and Phenomena Identification and Ranking Table (PIRT) exercise in CSNI ongoing. Some analytical and experimental studies being done in national programmes.
A.4.4. Human performance	Medium: supports improvements to severe accident management.	Low: Information not really needed for decommissioning.	Interviews with staff, data on conditions experienced, etc.	Enough time has passed that people will not remember details well. Low-cost, but not much additional information will be available. For this reason, no new proposed activities have been defined.	The NEA Working Group on Human and Organisational Factors and the Halden Reactor Project are performing some works on human performance under extreme conditions. Most countries have national programmes on these topics.
A.4.5. Seismic response • building • components	Low: Better data from other plants (not subject to accident).	Low: Information not really needed for decommissioning.			The NEA Working Group on Integrity and Ageing of Components and Structures is working on this topic.

4. Safety research activities (near-term projects)

4.1. Preparatory studies for fuel debris analysis

4.1.1. Objectives

It is imperative to undertake appropriate preparations in order to enable fuel debris retrieval work (including fuel debris sampling work prior to full-fledged fuel debris retrieval work) under safe and stable condition with steadily applicable technology. With this view, it is important to implement fuel debris characterisation (form, composition and physical characteristics of fuel debris) through experimental analysis using simulated fuel debris. It is also indispensable to develop radiation dose and exposure evaluation methods during fuel debris retrieval operations such as fuel debris sampling, retrieval and management.

The major objective of this project is to share the latest knowledge on the fuel debris characterisation, and the radiation dose and exposure evaluation methods during fuel debris retrieval operations, and to discuss future projects on fuel debris analysis techniques. The outcome of this project will be useful for sharing and accumulating knowledge and expertise of fuel debris retrieval work and for further deepening understanding SA progression for each member country.

4.1.2. Scope

1. In this project, several meetings and workshops will be organised to share the knowledge on fuel debris characterisation and radiation dose and exposure evaluation methods during fuel debris retrieval operations. Proposed topics are as follows:

   - fuel debris characterisation methods (analytical, experimental);
   - estimated fuel debris properties; physical, chemical and thermal properties;
   - experimental and analytical techniques.

   Note: Simulated fuel debris characterisations (R&D laboratory scale experiments) are underway within the Japanese national R&D projects as well as in several other national or European projects.

   Other possible activities could include:

   - joint R&D with BSAF-2 project.

   There is a possibility to extend the ongoing SAFEST European round-robin on MCCI debris sample characterisation to Japanese and international laboratories and to other samples/ analyses techniques.

   Workshop activities, that are proposed as phase I activities of the long-term consideration “in-vessel phenomena and RPV failure” mentioned in Section 5.1 and “ex-vessel phenomena” mentioned in Section 5.2, within the framework of this short-term project.

2. Radiation dose and exposure evaluation methods during fuel debris retrieval operations:
   - radiation dose and exposure evaluation methods;
   - criticality control and subcriticality monitoring techniques and evaluation methods;
   - public and worker protection techniques and methods.

   Note: R&D on evaluation methods of radiation dose and exposure as well as criticality control/monitoring and evaluation measures in the fuel debris retrieval operation are underway within the Japanese national R&D projects.
3. Discussion on possible future R&D topics that will use actual fuel debris samples:
   - possible international joint-research topics by making use of actual fuel debris to be retrieved;
   - knowledge sharing on fuel debris analysis and characterisation technologies.

   Note: R&D on actual fuel debris analysis techniques and technologies are underway within the Japanese national R&D projects.

4.1.3. *Tentative schedule*

- Duration:
  - 3 years (spring 2017 – spring 2020).
- Schedule:
  - 1st workshop and/or invitation of experts in 2017 for intensive discussion;
  - project group meetings and workshops from 2017 to 2019;
  - final report to NEA/CSNI by the end of 2020.

4.1.4. *Interested participants*

- operating organisation: JAEA
- possible participating organisations:
  Japanese organisations such as the Nuclear Regulation Authority, the Ministry of Economy, Trade and Industry, NDF, TEPCO and organisations from the CSNI member countries

4.2. Examinations inside reactor buildings (RBs) and primary containment vessels (PCVs), and water sampling

4.2.1. *Objectives*

Although understanding of status inside RBs and PCVs is indispensable for planning of safe and timely decommissioning, it is still very difficult to conduct even visual examinations because of highly limited accessibility. The Japanese organisations such as TEPCO and International Research Institute for Nuclear Decommissioning (IRID) have attempted and will continue to attempt to examine inside RBs and PCVs using robotics and other means.

The major results from the examinations having been done so far are summarised in Table 1 in Section 2. For example, in November 2013, the lower part of the PCV was examined from the torus room in unit 1 using a robot (surface boat) and the leakages from the sand cushion drain pipe and protective cover of an expansion joint on vacuum break line were confirmed. In April 2015, a snake-shaped robot equipped with a camera was sent into inside PCV of unit 1 and took the first look inside unit 1 PCV. In October 2015, water sample was taken from inside PCV through the penetration X53 of unit 3 and the concentrations of radionuclides and others were analysed.

It is important to understand the status of structures and components, dose and temperature distributions, etc. to establish the basic strategy for decommissioning. It is also important to identify the coolant leak paths from RPV to PCV and then to the turbine building (TB) in each unit and to better understand its mixing behaviour with highly contaminated water accumulated inside PCVs and TBs.

The major objectives of this proposal are to collect, compile and analyse basic data information obtained by the Japanese organisations on the status inside the RBs and PCVs organisations and to track and update the future long-term project proposals of the CSNI. The objectives also include testing the feasibility of transportation and examinations of the various samples to be taken.

The basic data information provided by these studies is essential to understanding the severe accident progression that actually took place, and therefore this project be closely linked with the BSAF project phase 2. This project is also expected to contribute to maintaining information channels between the CSNI and Japanese relevant organisations.

4.2.2. Scope

The major activity in the project is information exchange through workshops to be held periodically on examinations inside RBs and PCVs and analyses of the samples of water, concrete debris and so forth that are expected to be done by the Japanese relevant organisations. The workshop is also expected to provide Japan with some advice and recommendations to future examination/sampling activities. The topics to be discussed are shown as follows:

1. The status inside RBs and PCVs:

   - Discuss results from examinations including visual inspection by small robots, measurement of dose distribution, temperature distribution, size/shape of debris and others discuss, based on results of analyses on samples of concrete wall, rubbles and so forth and measurements on the distribution of radionuclides deposited and others, where feasible.

2. Behaviour of leaked water and that of with highly contaminated water inside TBs, RBs and PCVs:

   - Discuss based on results of analyses on samples of the water flowing, where feasible, (e.g. from PCVs before it mixes with the highly contaminated water in the torus rooms and measurements on the concentrations of fission products (FPs) and others to see if the contamination level of the water is considerably less than the water in the torus rooms).
   - Discuss based on results of analyses on the flow paths and mixing behaviour of contaminated water accumulated in PCVs and TBs, and FP leaching behaviour from the fuel debris by using existing measured data having been taken by TEPCO. As knowledge on the mechanism of dissolution of radionuclides from the damaged nuclear fuel into the water is very limited, measurements on the release rate and species of radionuclides dissolved in the cooling water would result in significant scientific advancement.

3. Proposal of for future long-term projects:

   - Discuss and propose, if feasible, a long-term project to the CSNI based on examinations and analyses to be done taking into account the decommissioning process of the Japanese government.

4.2.3. Tentative schedule

- duration:
  - three years (spring 2017-2020).
- schedule:
  - the 1st "technical meeting" in summer/autumn 2017;
  - project group meetings and workshops from 2017 to 2019;
  - final report to NEA/CSNI by the end of 2020.

4.2.4. Interested participants

Operating organisation: JAEA

Possible participating organisations: Japanese organisations such as the Nuclear Regulation Authority, the Ministry of Economy, Trade and Industry, NDF, TEPCO and others and organisations from the CSNI member countries.

5. Safety research activities (long-term considerations)

The long-term research considerations detailed here are recommended to the CSNI as they represent future project proposals which, in the judgement of SEG on SAREF, provide the best opportunities for addressing identified gaps in the safety-related knowledge post-Fukushima Daiichi, and at the same time support Japan in achieving safe and timely decommissioning. All of the examinations activities identified in these sections are considered by SEG on SAREF as having or adding value and worth considering for opportunistic inspections incidental to or with minimal adverse impact on decommissioning activities. As Japan proceeds with planning, preparation and implementation of decommissioning of the Fukushima Daiichi damaged reactors, SEG on SAREF recommends the planning activities be mindful of these other examinations and implement these examinations if the opportunity arises to do so with minimal or no adverse impact on decommissioning operations.

5.1. In-vessel phenomena and RPV failure

5.1.1. *Objective*

The objective of this long-term consideration is, on the one hand, to collect, analyse and evaluate data relevant to in-vessel progression of accidents and reactor vessel failure in the three damaged plants during the decommissioning operations and provide the scientific community with these data in order to fill knowledge gaps and be consequently able to improve severe accidents analyses and management in member countries, on the other hand, to support the Japanese organisations in preparing the retrieval of damaged fuel from the Fukushima Daiichi units with regard to in-vessel core melt progression and reactor vessel failure.

5.1.2. *Scope*

As mentioned in Section 1, the decommissioning of the damaged units of the Fukushima Daiichi NPS provides a unique opportunity to collect data related to the progression of severe accidents in BWRs at full scale, so that our understanding of the physical phenomena involved in core melt progression in the vessel and reactor vessel failure and their corresponding modelling in simulation codes be significantly improved. Also, the analyses of the collected data should, as was the case at TMI-2, contribute to the optimisation of the clean-up actions.

The following key data have been identified for in-vessel phenomena and RPV failure (Sections A.1.1 and A.1.2):

- End state (mass, density, composition[1], distribution, morphology and mechanical properties of various phases (crusts, debris, re-solidified molten phases), debris sieving) and peak temperatures of undamaged, damaged, and relocated core

1 If possible weight fractions and oxidation states of main components in oxide and metallic phases, search for boride phases, indications of FP distribution and association in oxide and metallic phases.

materials. With respect to RPV failure, special attention should be paid to material relocated in lower plenum, in control rod guide tubes (CRGTs) and at positions close to penetrations. Prior to detailed material analyses, inspections by detectors or cameras may provide highly valuable information such as fuel debris distribution with existence of blocked zones and flow by-pass zones, degradation state of upper, lower structures, RPV walls and penetrations, evidence of fuel foaming and melting.

- Evidence of interactions between fuel, cladding, fuel channel, control blades, structure and instrumentation materials.
- Evidence of interactions between relocated material and vessel walls in the lower head with a special attention to be paid to positions in the vicinity of penetrations and melt pool interfaces.
- Evidence of stratification within once-molten materials particularly in the vessel lower head.
- Physical characteristics affecting fuel debris coolability (particle shape and porosity, cracks, gaps in crusts).
- Evidence of effects due to seawater injection (salt precipitation, effect on melt formation).
- Evidence of core plate failure (manner and location).
- Evidence of degradation of structures in the upper plenum and lower plenum.
- Evidence related to vessel deformation and failure mechanisms (e.g. visualisation of global lower head deformation, of hot spots, of damages to instrumentation tubes, drain line, and CRGT penetrations and of attacks on RPV walls).
- If possible, analysis of mechanical properties of RPV samples.

The available knowledge and simulation tools be used as best as possible to predict the actual status of the damaged cores and vessels and therefore prepare the decommissioning operations: this is the objective of the ongoing BSAF project.

We propose therefore a long-term consideration by taking a phased approach which would accompany the successive steps of the decommissioning of the Fukushima Daiichi damaged units in the next decades.

5.1.2.1 *Phase I: Preparation of the decommissioning operations*

As a first step (within a year after completion of this report), it is proposed to organise a workshop between Japanese organisations and member countries to share knowledge about in-vessel core melt progression and RPV failure in the Fukushima Daiichi damaged units, discuss the degraded material retrieval operations and develop a possible programme of collection of samples and data during these operations.

These discussions could be based on: (1) the main lessons learnt from the TMI-2 examination programmes with regards to in-vessel accident progression, (2) the insights of the research programmes on in-vessel core degradation and RPV failure performed worldwide since the TMI-2 accident and, (3) the outcomes of the BSAF project (phases I and II) and further SA code crosswalk exercises.

This workshop could also be the opportunity to discuss ongoing and planned experimental programmes in this domain (e.g. SAFEST and the in-vessel melt retention funded by the European Commission, JAEA's investigations related to core degradation for BWR configurations and the Korea Atomic Energy Research Institute's investigations related to BWR vessel failure mechanisms) and to discuss to what extent they address knowledge gaps and how they are complementary of the possible examination programme to be developed for Fukushima Daiichi. The issue of the manufacture of corium and debris prototypes for planning fuel debris retrieval methodologies could also be evoked at this workshop.

5.1.2.2. *Phase II: Fuel debris retrieval operations*

During the fuel debris retrieval operations, considering the experience of TMI-2 examination programmes, it would be worthwhile to iteratively discuss and update the implementation of the examination programme developed in the previous phase, according to the actual status of the damaged cores and the actual carrying out of the retrieval operations.

These discussions would be based on knowledge gained through data acquisition and analyses and progressively more detailed simulations, based on these data and analyses.

5.1.2.3. *Phase III: Analysis of fuel debris samples and reconstitution of accident progression in units 1-3*

Currently, it is planned that the fuel debris and eventually RPV samples will be analysed in a specific laboratory set up by Japanese organisations in the vicinity of the Fukushima Daiichi NPS. Member countries could provide some of the expertise needed for these analyses and result interpretation. The possibility to send Fukushima Daiichi samples to foreign laboratories for independent analysis, as it was done for TMI-2 samples, is to be discussed. The expertise gained in several material analysis laboratories during severe accident experimental programmes could be useful to obtain the best information from the samples.

Moreover, these results should make it possible to improve the understanding and modelling of in-vessel and RPV failure phenomena in simulation codes, in particular with regards to BWRs, and to share common views on the progression of the course of the accidents which occurred at Fukushima Daiichi units 1-3.

5.1.3. Tentative timeline

The timeline of this research proposal is linked to the schedule of the decommissioning operation roadmap of the Fukushima Daiichi power station.

The workshop proposed in phase I could take place after the completion of the phase II of the BSAF project (i.e. around 2018), as it would provide very valuable inputs to the workshop discussions. This can be integrated into the near-term project.

The beginning of phase II should be consistent with the beginning of the phase 3 of the decommissioning roadmap (namely start of fuel debris retrieval operations, currently planned from 2021).

5.2. Ex-vessel phenomena

5.2.1. Objective

The objective of this long-term consideration is to support the Japanese organisations in preparation of the decommissioning of the Fukushima Daiichi units with regard to ex-vessel phenomena especially related to core melt relocation, spreading and MCCI and the collection of relevant data to fill knowledge gaps.

Looking from the scientific or technical perspective the Fukushima Daiichi accident may provide a variety of valuable data on the MCCI process and associated phenomena and may help to solve open questions especially related to the melt relocation from the RPV and the distribution in the lower part of the containment, the concrete ablation process, the erosion front behaviour, the impact of water on the coolability of the melt and the termination of the erosion process and the final melt composition at that time. To gather all the information related to MCCI from the decommissioning process, the involved Japanese experts and engineers attach great importance to a careful preparation of the task and ask for support by the international experts' community. As a follow-up activity a more close co-operation in preparation of the decommissioning activities at the Fukushima site could be proposed.

5.2.2. Scope

The decommissioning of the damaged units of Fukushima Daiichi provides a unique opportunity to collect data related to the progression of severe accidents in BWRs at full scale, so that our understanding of the physical phenomena involved in core melt progression, both in- and ex-vessel and their corresponding modelling in simulation codes be significantly improved. The key data to be gained from or through the decommissioning phase have been identified; major ones are similar to those in the in-vessel phase described in chapter A.1.1, others are repeated here:

- Melt distribution: core melt either frozen at components below the RPV or accumulated onto the pedestal floor including the sumps.
- Melt spreading into adjacent areas to the pedestal including determination of a possible containment liner attack.
- Melt relocation into tubes connecting the drywell and the wetwell.
- Characterisation of the morphology of the melt/debris on the pedestal/ containment floor and discrimination between a fraction of the corium which is obviously fragmented and a fraction of melt that can be described as a more or less continuous pool or cake and the geometrical configuration of both fractions (e.g. fragmented fraction on top of the frozen pool), presence of "eruptive cone", evidence on possible anchoring of crusts on the pedestal or sump walls; in a second stage, sieving of particulate debris would provide useful insights on fragmentation processes.
- Research of potential deformations that could be attributed to a mild steam explosion (a violent explosion would have already been detected), by visual inspections and also sieving of particulate debris (fine debris – median size of a few tens of millimetres – are an indicator of steam explosion).
- Characterisation of the distribution of melt composition throughout the debris by sampling (composition data at several locations, e.g. at the corium/concrete interface, at the corium/water interface and a variation of measurement location in vertical and lateral direction within the bulk of the corium and the crust) to see whether some stratification/segregation phenomena (e.g. refractory/non-refractory material or metal/oxide) occurred, analysis of radionuclide in samples.
- Finally, to collect any information related to the erosion process/progress in axial and radial direction at the different locations of the containment floor/pedestal; this information can be provided through systematic bores and/or successive removal of the melt/debris from the containment floor.

5.2.2.1. Phase I: Preparation of the decommissioning operations

A workshop – as a first step – could bring together experts from member countries including Japan to exchange knowledge gained with regard to MCCI and data acquisition and to support the preparation of the related decommissioning activities. Further it allows defining and discussing main important areas of interest related to MCCI data to be gained through the Fukushima Daiichi decommissioning process.

These discussions could be based on: (1) the main lessons learnt from the Chernobyl examination programmes as regards ex-vessel accident progression, especially melt relocation, (2) the insights of the research programmes on ex-vessel core degradation performed worldwide since the TMI-2 accident and, (3) the outcomes of the BSAF project (phases I and II) and further SA code crosswalk exercises.

This workshop could also be the opportunity to discuss ongoing and planned experimental programmes in this domain and to discuss to what extent they address knowledge gaps and how they are complementary of the possible examination programme to be developed for Fukushima Daiichi. The issue of the manufacture of corium and debris prototypes for planning fuel debris retrieval methodologies could also be evoked at this workshop.

5.2.2.2. *Phase II: Fuel debris retrieval operations*

During the fuel debris retrieval operations, considering the experience of TMI-2 examination programmes, it would be worthwhile to iteratively discuss and update the implementation of the examination programme developed in the previous phase, according to the actual status of the units and the actual carrying out of the retrieval operations.

These discussions would be based on knowledge gained through data acquisition and analyses and progressively more and more detailed simulations, based on these data and analyses.

5.2.2.3. *Phase III: Analysis of fuel samples and reconstitution of accident progression in units 1-3*

This proposal is almost identical to the one related to the in-vessel phase. Currently, it is planned that the fuel samples will be analysed in a specific laboratory set up by Japanese organisations in the vicinity of the Fukushima Daiichi NPS. Member countries could provide some of the expertise needed for these analyses and result interpretation. The possibility to send Fukushima Daiichi samples to foreign laboratories for independent analysis, as it was done for TMI-2 samples, must be discussed. The expertise gained in several material analysis laboratories during severe accident experimental programmes will be useful to obtain the best information from the samples.

Moreover, these results should make it possible to improve the understanding and modelling of in-vessel phenomena in simulation codes, in particular as regards BWRs, and to share common views on the progression of the course of the accidents which occurred at Fukushima Daiichi units 1-3.

5.2.3. *Tentative timeline*

The timeline of this research proposal is linked to the schedule of the decommissioning operation roadmap of the Fukushima Daiichi NPS.

The workshop proposed in phase I could take place after the completion of the phase II of the BSAF project (i.e. around 2018), as it would provide very valuable inputs to the workshop discussions. This can be integrated into the near-term project.

The beginning of phase II should be consistent with the beginning of the phase 3 of the decommissioning roadmap (namely start of fuel debris retrieval operations, currently planned from 2021).

5.3. Fission product database compilation and measurement

5.3.1. *Objective*

Although gaseous fission product releases from Fukushima Daiichi are broadly understood, there are several semi-volatile species such as Ru, Sb and Sr whose behaviour is either not well explained, or that were not observed in sufficient quantities to deduce their behaviour. In addition to this uncertainty, fission product releases by water vectors had a significant impact on the source term from Fukushima Daiichi. These releases have not yet been fully analysed. Fission product leaching from the fuel and aqueous pathways for their release are important parameters to understand both from the point of view of treatment of heavily contaminated water in the reactor buildings, and prevention of further releases to the environment during decommissioning activities. The proposal below addresses some of these knowledge gaps.

5.3.2. *Scope*

The proposed scope covers three main activities:

- compilation of a data base of radionuclide sampling;
- re-measuring archived samples (if any) for specific longer-lived radionuclides;
- performing additional measurements of heavily contaminated areas and water for long-lived radionuclides.

5.3.3. *Proposed work activities*

5.3.3.1. *Database compilation.*

The database compilation activity would involve two sub-activities:

1. Assembling data taken soon after the accident that has already been reported (on the TEPCO website, and from other sources), into tables according to measurement location and date. The data sources ideally would include air measurements performed by various national laboratories, and soil and water measurements close to or at the Fukushima Daiichi site. Table 3 contains a transcription of activities in order of total activity of chemical element, along with activities measured in the soil at the Fukushima Daiichi site, and in the turbine hall water in unit 2, at similar dates. An additional tabular form could be adopted for aqueous measurements from the trenches and the seawater.
2. A second component of the compilation would be to provide radionuclide measurements from current activities to decontaminate the waste water on-site, and sampling of water from highly contaminated areas (reactor basements) that will be performed in the near future. There are limited data currently available on the radionuclide concentration prior to treatment with the multi-nuclide removal equipment, although these measurements have been taken. Information on the radionuclide distribution on the ion exchange resins used for water decontamination would also be valuable.

5.3.3.2. *Re-measuring archived samples for specific longer-lived radionuclides.*

Depending upon the availability of samples, re-measurement of archived soil samples could be performed to detect some of the longer-lived isotopes without interference from short-lived species. ^{106}Ru, ^{125}Sb are two examples. Archived samples of the water samples taken in the turbine basements could also be measured for these radionuclides. Given that ^{90}Sr appears to be more abundant in water samples, the archived turbine basement samples could be measured for Sr as well.

Table 3. Comparison of activities of fission products observed at similar times after the Fukushima accident

Nuclide	$t_{1/2}$	Air activity(1) Mar 22-23 (Bq/m^3 air)	Fixed point 1(2) soil activity Mar 21 (MBq/kg wet soil)	Unit 2 turbine(2) hall water Mar 26 (MBq/cm^3)
^{131}I	8.025 d	2.18	5.8	13
^{132}I	3.204 d(3)	0.0998	0.61	
^{127}Te	9.35 h	0.00516		
^{129}Te	33.6 d	0.0347		
^{129m}Te	33.6 d	0.0549	0.25	
^{132}Te	3.204 d	0.100	0.61	
^{134}Cs	2.605 y	0.0363	0.34	2.3
^{136}Cs	13.16 d	0.00581	0.072	0.25
^{137}Cs	30.08 y	0.0354	0.34	2.3
^{99}Mo	65.976 h	0.00411	0.021	0.03
^{140}La	12.2527 d(3)	0.000464	0.033	0.19
^{125}Sb	2.759 y	0.000321		
^{95}Nb	34.991 d	0.000288	0.0017	
^{140}Ba	12.2527 d	0.000286	0.013	0.49
^{86}Rb	18.642 d	0.000282		
^{110m}Ag	249.83 d	0.0000655	0.0011	
^{103}Ru	39.247 d	0.0000495		
^{113}Sn	115.09 d	0.0000467		

(1) Information taken from Reference [4]

(2) From the TEPCO website.

(3) Half-life of released parent nuclide.

5.3.3.3. *Performing additional measurements of heavily contaminated areas and water for long-lived radionuclides.*

There are two components to this activity:

1. The first is re-measurement of soil samples in areas previously characterised. Depending upon the availability of archived samples, additional soil sampling could be undertaken to assess the radionuclide concentration in areas previously sampled, again with the aim of detecting longer-lived isotopes.

2. Extensive characterisation of any contaminated water from the reactor buildings (wet well, water leaking from pressure vessels and water found in turbine basements) will

provide invaluable information about fission product leaching from fuel. Note that decommissioning needs have also identified the need for careful measurements of radionuclide concentrations in water leaked out of the PCV and of contaminated water in the buildings.

5.3.4. *Tentative timeline*

A portion of these activities are likely to start in the short term, and indeed have already started as part of TEPCO and JAEA sampling campaigns for measuring environmental radioactivity. However, it is anticipated that measurements to support worker dose assessments will be ongoing for several years, hence the proposed activities above have been designated as long-term activities.

5.4. Mission time and system survivability

5.4.1. *Objective*

The severe accident performance of certain components and systems in conditions leading up to and during a beyond design basis event (BDBE) has a major impact on overall reactor survivability and the mitigation of consequences. The data obtained during decommissioning process could help address reactor safety technology knowledge gaps associated with these components and systems [5]. The overall objective of this activity is to support TEPCO and other organisations towards implementing possible examination of these key Fukushima Daiichi components and systems while planning and implementing safe and timely decommissioning. Components and systems of most interest are:

- cables and sealing;
- instrumentation;
- reactor core isolation cooling (RCIC) system;
- primary system safety relief valves (SRVs);
- environmental degradation of metallic components.

Survivability of electrical cables is important to severe accident management (SAM). The performance of cables is needed for sensors transmitting the measured data to the control room for activation of operator actions according to SAM Guidelines. Actions during the early stage of BDBE include opening specific valves (e.g. cavity flooding), maintaining the desired valves function, assessing performance of hydrogen igniters and measurement of different plant parameters like pressures, temperatures, gas concentrations and water levels. Cable material may be exposed to high temperature, radiation and mechanical loadings resulting from core melt progression itself and from various external events and internal fires. The cables may also become submerged into water contaminated with fission products and other chemicals. Survivability of cables applies for the containment and reactor building as well as to separate spent fuel pool storages. Recent information about the cable performance suggests that different irradiation history with the same cumulative dose may result in different cable survivability conditions. The information from the post-Fukushima examinations would provide information about the conditions of cables in authentic, complex severe accident conditions and would support development of more resistant jacket materials and the evaluation of reliability of existing electrical systems in severe accident situations.

Polymer materials are employed as sealants in numerous containment penetrations such as electrical penetrations, pipeline penetrations, material and personal hatches. In addition to polymers, a light-water reactor (BWR in particular) may also employ other organic compounds in lubricants and paints. The technical qualifications of the seal components in the containment are tested for normal operation and under design basis accidents (DBA) conditions with temperature ranging typically from room temperature to about 170°C. Much higher temperatures can be expected during severe accidents relating to e.g. superheated conditions, hydrogen combustion or radiative heating from core melt. In addition to temperature degradation the seal materials can be exposed to high radiation dose rates and high cumulative doses. The elastomer seals are susceptible for ageing. The degradation of strength and elasticity properties has been studied for several decades, mostly under normal and DBA conditions. The seals may also be submerged into water for long periods of time as a consequence of SAM measures. Many of the penetration seals are typically qualified for atmospheric conditions. The leak-tightness of the penetrations is crucial for the containment long-term performance as a barrier against spreading of radioactivity. Current research of polymer behaviour under radiation and temperature loading conditions suggests that different material may perform quite differently. Relatively little information is available on the capability of the mechanical penetration structure itself to protect the sealing elastomer from radiation and temperature loads. Fukushima Daiichi post-examinations would provide valuable information about the status of the seals for various types of penetrations. The knowledge gained from the examinations (visual and material property analyses) would support in knowing the extent of reliability in post-severe accident leak-tightness of the nuclear containments and also in development of more resistant seal materials.

Instrumentation data are critical during severe accidents to enable operators to diagnose the status of the plant and evaluate the efficacy of accident management actions. During the Fukushima Daiichi accident, numerous instrumentation measurements were suspect or unavailable as a result of loss of alternating current (AC) power and direct current (DC) power. Further, harsh BDBE conditions adversely impacted instrumentation. After power was restored, inaccurate and inconsistent values and/or trends across instruments measuring the same parameters were observed. As a result, operators struggled to discern reactor status and the impact of operator actions.

The reactor core isolation cooling (RCIC) system is the major long-term heat removal system employed under a wide range of transients and accidents in BWR designs. Probabilistic risk assessments indicate that the dominant accident sequences for BWRs are BDBEs involving extended loss of AC power or extended loss of alternating current power (ELAP) events and would involve RCIC system operation. Thus, extended performance of RCIC systems under BDBE conditions is important to overall plant safety in terms of reducing both the likelihood and the consequences of core damage events involving ELAP. Available information suggests there is significant margin in RCIC systems that has been neither quantified nor qualified. Determining the actual operating envelope under BDBE conditions could enable operators to expand the required time before transition to other safety measures. Post-accident examination information could improve understanding of accident progression and support development of more effective accident management and response procedures. Further, the similarly configured turbine driven auxiliary feed water (TDAFW) systems are the major long-term heat removal systems in PWR designs so better understanding of RCIC system performance benefits the operating PWR fleet as well as the BWR fleet.

Primary system safety relief valves (SRVs) are essential components for controlling RPV pressure in accident management procedures. After reactor trip, SRVs are used to reduce primary system pressure and remove decay heat. Under BDBE conditions there are no relevant data aside from what may eventually be revealed from the Fukushima Daiichi accident decommissioning activities. Data from the Fukushima Daiichi accident indicate that protracted SRV cycling took place in all three reactors and under the added duress of extreme temperatures cause by core degradation processes. Reducing uncertainties in relief valve and associated piping performance under BDBE conditions would provide a better understanding of accident progression. New models could be developed and validated with appropriate test data to characterise relief valve and associated piping performance under BDBE conditions. Post-accident examination information could also improve our understanding of accident progression, support development of more effective accident management and response procedures, and identify potential plant system enhancements.

Concerning environmental degradation of metallic materials, one way of monitoring this degradation is to correlate the evolution of microstructure and material damage with applied loadings and conditions and this is particularly useful in the case of infrequent transients. And the "transient" at Fukushima Daiichi is unique on its kind. In this sense, metallic materials coming from the decommissioning process of Fukushima Daiichi could be of high interest for the international community, as these materials were exposed to conditions that cannot be reproduced without large efforts. The study of the characteristics of the materials under these accident conditions could help in the understanding of the degradation mechanisms expected in operating nuclear power plants and in addition to gain some insights into the response of structural materials under accident conditions. It is acknowledged however that in order to do this, significant reconstruction of the transient conditions likely to have existed is required.

5.4.2. Scope

The severe accident performance of key components/systems in conditions leading up to and during a BDBE has a major impact on overall reactor survivability and the mitigation of consequences. Examination of these systems at Fukushima Daiichi could be helpful in severe accident safety enhancement. SEG on SAREF recommends long-term considerations on the following examinations:

- Visual/photographic examinations of selected cables in the PCV and in the RB.
- Cable samples be collected from locations with cables in the atmosphere environment and from submerged conditions if applicable. Operability of the cable samples be tested. If possible the changes in the cable jacket material and other parts of the cables be examined and tested.
- Visual/photographic examinations of the penetrations and the elastomer materials be carried. Penetrations cover different types: electrical penetrations, material and personnel hatches and bellows if applicable.
- Samples of different seal materials be collected from penetrations. Material properties and operability be tested.
- Visual/photographic examination, selected sampling and selected operability assessments of in-vessel and ex-vessel sensors, support structures and associated cables [6].
- Photos/videos of condition of RCIC valve and pump before drain down and after disassembly at affected units at Fukushima Daiichi (units 1 and 2).
- If these RCIC systems are disassembled as part of decommissioning:
  - check for evidence of damage to the RCIC turbine caused by extended operation in the self-limiting mode;
  - determine RCIC system failure mode(s) for RCIC system on unit 2.
- Photos/videos of main steam lines and automatic depressurisation system lines to end of SRV tailpipes, including instrument lines.
- Visual inspections of SRVs, including standpipes (interior valve mechanisms).

- Photos/videos of TIP tubes and SRV/intermediate range monitor tubes outside the RPV.
- Photos/videos, probe inspections, and sample exams of main steam lines (MSLs); Interior examinations of MSLs at external locations.
- Direct observations (visual inspection) and collecting metallic materials from SSCs for testing at labs. Matrices identifying the eligible materials, the stressors and degradation mechanisms to be investigate be prepared to elaborate the harvesting programme and the direct observations programme. Co-ordination with other safety research areas where similar examinations are proposed is recommended, as some overlap exists with previous proposals.

The RCIC systems at Fukushima Daiini were also subjected to accident conditions and information on RCIC system status could potentially be more easily obtained at a lower cost and with less radiation exposure to personnel. As such, SEG on SAREF suggests operators consider examination of the Fukushima Daiini RCIC units through:

- Complete forensic disassembly of the RCIC turbine and pump with measurements of clearances and visual conditions via pictures. This could include:
  - Inspection of RCIC pump impeller/wearing rings for cavitation related damage.
  - Oil sampling from RCIC turbines operated under event conditions or sample results if sampling and analysis are already completed.
  - Pump mechanical seal condition
    If these detailed examinations of the RCIC systems at Fukushima Daiini are implemented, operators could document RCIC operational history during the event to include; length of time RCIC operated during post event response, trips/restarts of RCIC, actions taken to preserve RCIC operation, local operation of RCIC performed during event response, and temperature history of RCIC suction during post-ELAP operation.

Some of these components may be readily accessible, but others may be difficult to evaluate. Operators could take a graded approach, determining what components to visually examine, and what components, if any, to sample and subject to operability assessments based on the ease of examinations, location and noticed trends or other issues of interest. Information available related to the various instruments cables and support structures could be collected and categorised first to help guide these efforts.

Note: The major criteria for SAREF's recommendation of an examination activity is that it be a high priority both in terms of the safety research interest as well as the decommissioning interest (i.e. how well it helps facilitate decommissioning activities). While all four of these items in this mission time and system survivability area are a high priority in terms of future reactor safety, they are a low priority in terms of decommissioning. However, SEG on SAREF concluded these topics merited consideration for examination because of the direct relevance to mission time and system survivability and given a graded approach to the examinations.

5.4.3. *Tentative timeline*

Some examinations could likely be done in the near term, as these items may be readily accessible. For example, a number of the proposed examinations are basic photographic/video/visual examinations such as of the cables, sealings, piping and valves inside the primary containment vessel. Many of these visual examinations can probably be done with minimal adverse impact during initial entry and examination of inside of the PCV with cameras. Other examinations are likely to be long-term activities as they require considerable decommissioning progress, such as draining the RCIC rooms or removal and close examination of instruments. Some instrument examinations may require near-proximity evaluation, such as in situ calibration evaluations, etc. and sampling/removal to examination facilities. As such, the schedule of these examinations is necessarily linked to the Fukushima Daiichi decommissioning roadmap and reactor examination planning activities would need to be based on the decommissioning activities of the roadmap.

SEG on SAREF recognises some of these suggested examinations may be prohibitive in terms of cost, dose, or impact on ongoing decommissioning activities which are critical decommissioning priorities. The priority of such examinations will need to be evaluated in light of these issues.

6. Conclusions and recommendations

The CSNI undertook the SAREF initiative to provide guidance on potential examinations of the damaged reactors at Fukushima Daiichi, with the twofold goal of addressing safety research gaps and of supporting Japan in achieving safe and timely decommissioning.

A group of senior experts in nuclear safety research as well as decommissioning was assembled to identify areas of common interest and recommend to the CSNI safety research activities including their priority. The group reviewed research areas relevant to three categories: 1) severe accident progression, 2) Stress corrosion cracking performance and condition, and 3) the recovery phase of an accident – and included a fourth category – 4) general or already addressed. For each research area consideration was given to the international safety research interest, the decommissioning interest, potential examinations, challenges (e.g. cost, timing, dose, etc.), feasibility and ongoing R&D activities. The results of the Senior Expert Group's deliberations are summarised in the Table 2 in Section 3.

In the table, a research area is identified as having high safety research interest if information that can be gained during Fukushima Daiichi decommissioning addresses significant knowledge gaps. If the knowledge gaps are not significant, or the information could be better gained another way, the safety research interest is only medium or low. For example, the safety research interest in information on hydrogen combustion is rated medium even though the combustion events at Fukushima Daiichi were the first for a nuclear plant during a severe accident. This rating was chosen because it is difficult to retroactively determine how the hydrogen behaved, and because the effects of the combustion events cannot be easily separated from other sources of pressurisation and damage.

Similarly, decommissioning interest in a research area is high if the information gained is necessary to support safe and timely decommissioning. For example, information on the survivability of instruments during and following the accident is of low interest to decommissioning as any malfunctioning equipment is either not required to ensure ongoing safety or can be replaced without determining how it survived the accident.

In terms of determining areas for further work, those with both high safety and high decommissioning interest are clearly higher priority than those with lesser interest. Nevertheless, other factors also need to be considered. In some cases, it may not be very feasible to get the desired information, or it may be very challenging in terms of cost or worker dose. Another consideration is the anticipated timing for a proposed examination. Those that are not planned or possible for a number of years will benefit from the additional information that becomes available, and from the opportunities offered by the preparatory work to improve the success of the eventual examination. An example of such preparatory work are the studies on fuel debris analysis that will prepare for actual debris samples from Fukushima Daiichi, described below.

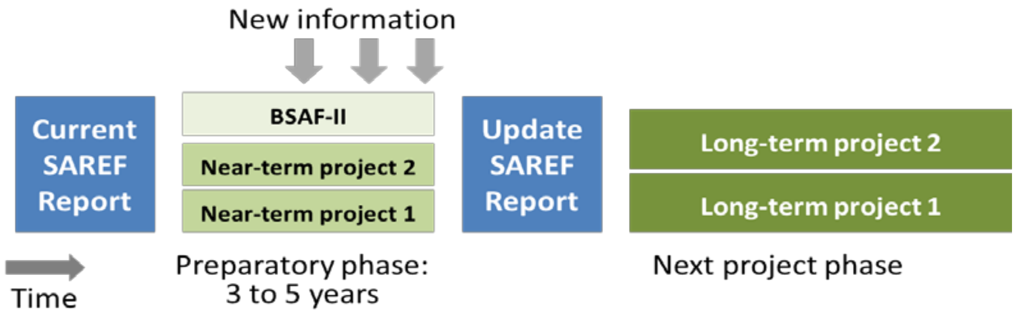
With these factors in mind, the SEG on SAREF is recommending four long-term considerations (see Section 5 for details). There are two long-term considerations on fuel debris retrieval and characterisation from inside and outside reactor pressure vessels that directly address the areas of in-vessel and ex-vessel phenomena. The ex-vessel proposal also addresses in part containment failure and venting. There is a long-term consideration to compile a database on fission product measurements that addresses fission product behaviour and source term. The last long-term consideration is based on opportunities for examinations of reactor components that may arise as decommissioning proceeds to address areas of high safety interest under mission time and system survivability. Recognising that these long-term considerations require additional information and development of decommissioning planning and processes, two near-term proposals are also recommended, and the motivation is summarised here:

1. *Preparatory studies for fuel debris analysis*: It is important to determine what happened to the cores of the Fukushima reactors, to confirm and improve modelling of accident progression, and to provide information to support decisions on options for decommissioning the reactors. Having said that, sampling of reactor internals is not possible at the moment and not planned for some years to come. It is also anticipated that special tooling and sampling technology will be required to remove and retrieve pieces of material from inside the containments of the Fukushima reactors. Challenges include providing protection against radiation fields and contamination, and how best to characterise samples that have been removed. As a result, preparatory studies are recommended to develop sampling technology and characterisation methodologies in advance of obtaining actual samples from reactor internals.
2. *Examinations inside reactor buildings and primary containment vessels, and water sampling*: Further examinations inside the reactor buildings and containment vessels are similarly required to confirm and improve modelling of accident progression, and to determine the status of components and structures in support of decommissioning decisions and options. Information can be gained in the near term from remote visual inspections and temperature and radiation measurements made with purpose built robots. In addition, samples of various water bodies and rubble can be removed with robots to provide information on the distribution of radionuclides.

A third activity that is already underway and will continue over the near term is the benchmark study of the accident being conducted under the Benchmark Study of the Accident at the Fukushima Daiichi NPS (BSAF) joint project. This project is providing valuable information on the possible end-states inside of the damaged reactors and is indicating where further information from the reactors would help improve the modelling and predictions.

Over the next few years, additional information will be gained from the progress in decommissioning, from the examinations at Fukushima Daiichi, from the development of sampling methodology and from the BSAF modelling studies. This information will have an influence on the ongoing near-term activities, and therefore it will be important to have a mechanism for sharing the information gained between organisations in Japan relevant to Fukushima Daiichi decommissioning and international experts in reactor safety research. Moreover, the new information will also have an impact on the assessment that has been performed by the SEG on SAREF and documented here, and an update will be required. As a result, SAREF can be seen as an ongoing process, with the assessment driving near-term activities, then feedback from new information feeding into near-term activities and leading to an update in the SAREF assessment, which will then drive the next phase of activities as shown in Figure 2. An important element of the SAREF process will be a mechanism that allows for ongoing interaction between international experts and relevant organisations within Japan. As one of the possible mechanisms, the SEG on SAREF recommends that a joint annual meeting and/or workshop be held by inviting international experts including those in the CSNI working groups where the updates be presented from the near-term activities and from the decommissioning activities at Fukushima Daiichi. The SEG on SAREF also recommends the CSNI to monitor the ongoing process of SAREF.

Figure 2. Overview over the near- and long-term activities following the SAREF report



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Appendix: Safety Research Area of Interest

The details of the assessments of the status of the existing knowledge base, safety research interest, decommissioning interest, potential examinations at Fukushima Daiichi, and ongoing R&D activities, are provided in the Appendix, while the results are summarised Table 1 in the chapter 3. Through the course of the meetings of SEG on SAREF, a number of research topics were raised and discussed in terms of identifying safety research areas of common interest and recommended examination. The contents below in this appendix are based on a compilation of written contributions by SEG members and a series of extensive discussions.

A.1 Severe Accident Progression

A.1.1 *In-Vessel Phenomena*

A.1.1.1 Background

The accident that occurred in 1979 at Three Mile Island Unit 2 (TMI-2) was the starting point of a large effort worldwide to gain knowledge on the phenomenology of a core melt accident, based on the examination programmes performed at TMI-2, on numerous experimental programmes – analytical and integral ones, performed either out-of-pile or in-pile – and on the development of computer codes aimed to simulate the progression of a core melt accident and its consequences.

A.1.1.1.1 Experience from TMI-2 examination programmes

After the accident, several examination programmes have been performed under the auspices of US DOE and NRC. An R&D agreement was signed in 1980 between GPU, EPRI, NRC and DOE (GEND) to obtain information about the accident sequence and its effects. An R&D characterisation plan was developed that sought extensive data about many areas of the plant [1-4] and followed by an NEA/CSNI programme in which TMI-2 samples have been analysed in international laboratories [5, 6].

As it was recognised that significant fuel relocation to the lower head occurred, a new NEA/CSNI programme was initiated: the ''Reactor Vessel Characterisation''. It began with analyses of accident scenarios using computer codes and continued with water sample analysis, video examinations, radiation and instrumentation readings, gamma scanning of an in-core detector, debris sampling, topographical mapping by sonar, core stratification drilling, and removal of samples of the reactor vessel itself [6, 7].

The vital importance of data was recognised early with two objectives:

(1) to support ongoing clean-up operations;

(2) to extract information of scientific value.

Management policy was that research work could not significantly interfere with clean-up work. Nevertheless, research work often furthered the progress of clean-up by providing crucial and unique information to the planning of the clean-up operations.

A key and representative example was the ''Core Stratification Sampling Program'', sponsored by the DOE in 1986, in order to obtain samples of the ''hard layer'' below the upper debris bed and from the formerly ''unknown'' regions below it. The motivations for the data acquisition programme were a mixture of R&D and support to the clean-up operation:

- Extract representative material samples from the core region that were suitable to support research needs; it was especially important to the research community that samples taken before the original post-accident conditions of the core were altered by defueling operations;
- Provide supporting data on conditions in the lower head to answer questions about the extent of material relocation;
- Provide the recovery team with data that would help in defueling operations.

A commercial drilling machine was modified to enable drilling of samples from the core regions and to open access to the lower head region. A fairly accurate picture of post-accident conditions could be derived that was helpful for defueling planners. Finally, the core boring machine was also used for defueling operations.

The information gathered through the various examination programmes performed at TMI-2 during the clean-up operations permitted a significant improvement of our understanding of the evolution and the related physical phenomena of a core melt accident in PWRs. Combined with many analytical experimental programmes performed worldwide and with the development of physical models and computer codes, these examination programmes proved essential to enhance our understanding of core melt accidents and therefore our related managing capabilities [8, 9].

A.1.1.1.2 Status of knowledge on in-vessel severe accident progression

After the TMI-2 accident, a large effort has been devoted worldwide to gain knowledge on core melt accident phenomenology. Several experimental programmes, including large scale tests, have been performed; new physical modelling has been developed and implemented in integral accident simulation or more specific computer codes. Large scale tests dedicated to core degradation are summarised in the table below (derived from [30]).

Among integral test programmes related to in-vessel severe accident progression, one can mention the CORA and QUENCH out-of-pile experiments and the PHEBUS Severe Fuel Damage and PHEBUS FP in-pile experiments. These experiments are large scale, with typically 20-25 fuel rods of up to 1 m heated length enclosed by an insulating shroud, with control rod material present (Ag-In-Cd or B_4C). The QUENCH experiments are more specifically focused on the study of water re-flooding of (slightly) degraded cores.

Table A.1. Overview of PWR-related experimental programmes

Test	Description	Phenomena explored
PWR		
Loss Of Fluid Test (LOFT)		
FP-2	Large scale fuel bundle severe damage test with reflood	Fuel heat-up, cladding oxidation, H_2 generation, quench behaviour
Power Burst Facility Severe Fuel Damage Tests (PBF SFD)		
SFD ST, 1-1, 1-4	Small scale and fuel assembly severe damage tests with boil-off and steam flow	Fuel heat-up, boil-off cladding oxidation, H_2 generation
SNL Annular Core Research Reactor (ACRR)		
MP & ST series	Small tests with irradiated clad fuel; simulation of the heat-up of PWR in-core debris bed	Fission product release from irradiated fuel; debris bed melting
Full-Length, High-Temperature (FLHT)		
FLHT-2, 4, 5	Heat-up of full-length PWR fuel assembly; coolant boil-off	Boil-off, fuel heat-up and damage, H_2 generation, noble gas release
CORA		
CORA-2, 3, 5, 7, 9, 10, 12, 13, 15, 29, 30	Fuel assembly with electrical heater rods, Inconel spacers, Ag-In-Cd absorber	Fuel heat-up and damage, cladding oxidation, H_2 generation, reflood and quench
CORA-W1,W2	Fuel assembly VVER type with electrical heater rods, B_4C absorber	Fuel heat-up and damage, cladding oxidation, H_2 generation, effect of B_4C rod on bundle degradation
PARAMETER		
PARAMETER-SF1, SF2, SF3	Fuel assembly VVER type with electrical heater rods	Fuel heat-up and damage, cladding oxidation, H_2 generation, quenching from top and bottom and simultaneous quenching from top and bottom
PARAMETER-SF4	Fuel assembly VVER type with electrical heater rods	Fuel heat-up and damage, cladding oxidation, H_2 generation, air ingress phase, oxygen starvation, nitriding and bottom quenching
PHEBUS		
B9+ (SFD), FPT-1 to 4 (FP)	Fuel assembly and integral severe fuel damage tests: steam generator deposition, containment aerosol/chemistry; melt progression in debris bed geometry with irradiated fuel	Fuel heat-up, liquefaction, collapse, eutectic behaviour, H_2 generation, FP release, speciation and volatility, transport and deposition, containment chemistry and deposition, and iodine partitioning; late phase melt progression and low volatility FP release
QUENCH		
QUENCH-1 to 15	Small simulant fuel assembly with electrical heater rods, Ag-In-Cd absorber	Fuel heat-up and damage, cladding oxidation, H_2 generation, quenching

Table A.2. Overview of BWR-related experimental programmes

Test	Description	Phenomena Tested
BWR		
ACRR Damage Fuel Tests (ACRR DF)		
DF-4	Small bundle test that included fuel, channel box and SS control blade with B_4C	Fuel heat-up, cladding oxidation, H_2 generation, B_4C-SS eutectic interaction, fuel liquefaction, fuel rod collapse
CORA		
CORA-16, 17, 18, 28, 31, 33	Small electrically-heated fuel assembly (UO_2) with channel walls and B_4C/SS control blade; steam/Ar flow	Fuel heat-up, damage, cladding oxidation, H_2 generation, quenching (1 test)
XR		
XR1-1, 2; XR2-1	Fuel assemblies, channel walls and B_4C/SS control blade	Full scale BWR core cross-section with core-plate structures represented; response of lower core structures to prototypic relocating liquid materials from upper core

Based on the set of these experiments, the general phenomenology of PWR core degradation is now considered as well understood. The following main stages can be identified [10-12]:

- 720–820 °C: Creep deformation (ballooning) and failure of the Zr-alloy clad in low pressure sequences, and melting in situ of Ag-In-Cd absorber alloy;
- 1200–1400 °C: Start of rapid Zr oxidation by steam leading to uncontrolled temperature excursion and extensive hydrogen production; liquefaction of Inconel grid spacers and absorber rod materials due to chemical interactions, producing metallic melts which initiate core melt progression (localised fuel rod and control rod damage, metallic melt relocation and freezing in colder zones, partial flow blockage);
- 1750–2000 °C: Melting of the remaining metallic Zr-alloy and/or Zr(O) with subsequent chemical dissolution of UO_2 fuel, leading to the formation of metallic and ceramic melts which relocate and form large blockages upon solidification (extended core damage);
- 2150-2600 °C: Fuel rod collapse and meltdown of core materials, formation of ceramic melts and a melt pool expanding upon temperature increase. Fuel liquefaction occurs at much lower temperature (2 350 °C ± 200 °C) than the melting temperature of UO_2 (2 830 °C) primarily due to additional UO_2-clad interactions. Fuel swelling and oxidation may also contribute to fuel melting at lower temperatures;
- 2 600 °C and above: Large meltdown of core materials and formation of a large melt pool (large core destruction) eventually resulting in molten materials flows to the reactor vessel lower head and formation of a molten pool at this location.

Other experimental programmes with prototypic materials have been devoted to the study of corium behaviour in the lower head. They are summarised in the table below (derived from [30]). These programmes have provided valuable insights on the stratification of the molten pool in several layers with different composition, having a direct impact on vessel lower head failure mechanisms [13, 14]. They also have provided information of Fuel-Coolant Interaction phenomena and their impact on debris bed formation [15, 17].

Table A.3. Overview on experimental programmes related to corium behaviour in the lower head

Experimental Programme	Description	Phenomena Tested
FARO (Fuel melt And Release Oven)		
L-5 to 33	Prototypic materials relocating through water	Melt stream breakup and quench
KROTOS		
K-21 to 58	Prototypic materials poured into a water pool	Melt stream breakup and quench
KS1 to 6	Prototypic corium compositions poured into a water pool	Melt jet breakup and quench, steam explosion
TROI		
TROI-1 to 40	Prototypic materials poured into a water pool	Melt stream breakup and quench
TROI-49, TROI-50 2	Prototypic material test with induction heating to observe molten corium materials	Stratification in molten pools
RASPLAV		
AW-200-1 to 4	Prototypic material test with electrical heating to observe molten corium materials	Natural convection, stratification in molten pools
MASCA		
RCW-1 (RCW); MA-1 to 9 (RASPLAV-2); small scale STF tests	Prototypic material test with inductive heating to observe material interactions, stratification, natural convection, and fission product distribution in stratified corium materials both in inert and oxidising atmosphere	Natural convection, stratification, and fission product distribution in molten pools

A number of separate-effect programmes have also been and are being performed to help a more quantitative understanding of individual phenomena. Of special interest for BWRs is the effect of B_4C control rod materials on core degradation, which is recognised to be significant due to its intensive eutectic reactions with the surrounding metal at low temperature. These reactions lead to the formation of less viscous melts and possible melt projections, increasing the extent of degradation and relocation. They also contribute to hydrogen production. The separate-effect Becarre programme has been dedicated to this issue and provided an in-depth knowledge about the involved mechanisms [18].

Another separate-effect programme of interest is the (POMECO) programme performed at the Royal Institute of Technology (Sweden) in order to investigate the debris bed coolability, using particulate beds with multi-sized spheres or irregular sand particles [19, 20].

Composition and properties of corium determine its behaviour and its potential interactions with the reactor vessel: this requires detailed information on the phases present at specific temperature and how the phases are formed. Therefore, additional data on chemical systems at high temperature should be obtained and integrated in thermodynamic data banks in order to construct accurate equilibrium phase diagrams [21, 22].

A.1.1.1.3 Ability of current advanced codes to simulate in-vessel accident progression

For many years, computer codes have continuously been developed. These codes aim at modelling the most important physical phenomena involved in severe accidents in an integrated manner: one can mention MELCOR, MAAP, ASTEC and SOCRAT codes. Their models are based on the results of separate-effect experiments and validated on the results of integral experiments as well as the results of the examination programmes conducted at TMI-2.

A means to assess our current level of knowledge on severe accidents is the benchmark exercise recently performed within the CSNI/WGAMA: the results of several advanced severe accident codes have been compared on ''TMI 2-like'' scenarios in order to assess their ability to predict in-vessel core melt progression and degraded core coolability [23].

With regards to in-vessel accident progression, the following recommendations are made:

- New and/or improved models based on additional experiments are needed for the description of late-phase degradation phenomena. These experiments should mainly focus on:
  - debris bed and melt coolability inside the core region (in particular for areas with very low porosity);
  - melt interaction with core support structures and slumping into the lower head;
  - melt behaviour in the lower head.
- The fact that the amount of hydrogen generated during the core reflood differs considerably in the simulations requires further in-depth analysis of factors that impact zirconium oxidation rate in various severe accident scenarios, including applicability of correlations to non-rod-like geometries (debris, melt).
- This benchmark showed that some "cliff-edge" effects still exist, e.g. for the quenching of a much degraded core where some codes predict success of quenching whereas other codes predict the impossibility of stopping melt progression and thus, the occurrence of vessel failure. Consequently, uncertainty and sensitivity analyses are strongly recommended to support and complement code applications for deterministic plant analyses.

A.1.1.2 Safety Research Interest

Some recent studies performed in Europe and in the US and in Japan identify current gaps in our knowledge on core melt accidents, in order to define research priorities. A general outcome is that most of experimental programmes have been devoted to PWRs and, as a consequence, core melt computer codes are less validated for BWRs than for

PWRs. Therefore, it is worthwhile to first point out the specific features of BWRs with regards to core melt accidents.

A.1.1.2.1 Specific features of BWRs as regards core melt accidents

Compared to the situation for PWRs, the knowledge and the simulation capability of severe accident progression in BWRs are more limited. As explained above, the results of the examinations performed at TMI-2 permitted to have an accurate view of the whole picture for PWRs, which has been completed by many ''separate-effects'' experimental programmes; on this basis, computer codes have been developed and validated. Moreover, as shown in Table A.2, there are fewer BWR large scale tests (~10) in comparison to PWR tests (~40). Furthermore, all 10 of the BWR experiments were initiated in a dry environment unlike the events at Fukushima.

The progression of severe accidents in BWRs could be significantly different from what has been established for PWRs. This is due to some specific features of BWRs (Figure A.1) compared to PWRs [24-26]:

- Fuel assemblies are enclosed in channel boxes, that hinder cross flows of steam during core dewatering and of molten material during the core degardation;
- The inventory of zirconium is significantly higher as the channel boxes are made of zirconium as well;
- The control rods are cruciform control blades, with a significant metallic mass;
- Boron carbide (B_4C) is used for neutron absorption in control rods (instead of silver-indium-cadmium - SiC- used in most PWRs);
- There are massive steel structures above the core (steam separators, steam dryers…);
- There are massive steel structures in the lower plenum (control rod drives, control rod guide tubes);
- There are a large number of lower head penetrations (control rod drives and instrumentation) which upon weld failure or melting through of structure may lead to vessel failure.

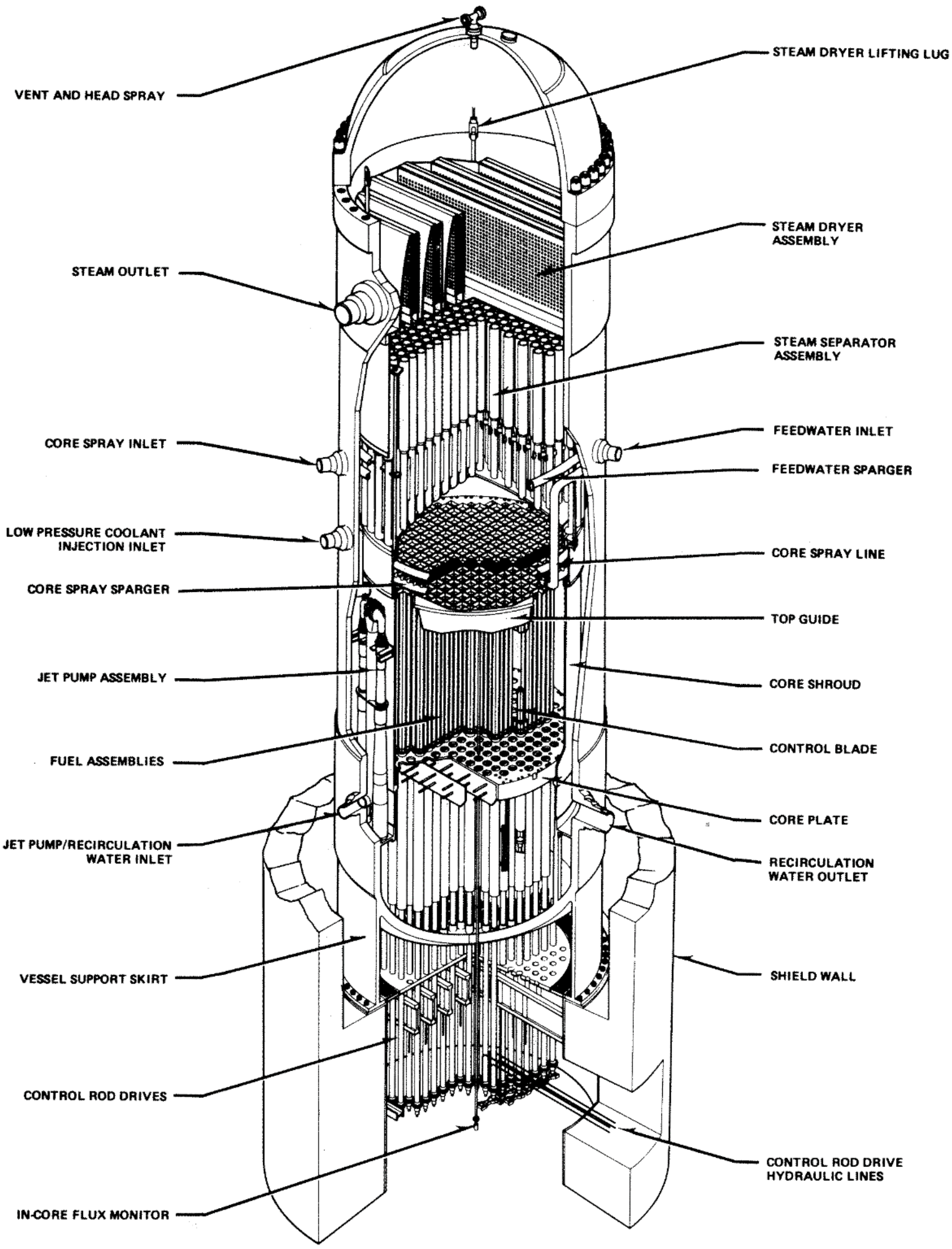
The initial stage of accident progression is probably affected by the specific geometry of the core (channel boxes, control blades in the bypass between the fuel channels) and by the presence of massive steel structures above and below the core.

Then, the boron carbide use as absorber material is likely to form eutectics with the constituents of stainless steel at low temperatures, leading to melted materials early during core heat-up, which will flow downwards to cooler regions of the core bypass.

At higher temperatures, steel structures, claddings and fuel pellets will melt and relocate. Some models indicate that these liquefied elements do not accumulate as a molten debris pool within the core region, as it is supposed to occur in PWRs (and eventually occurred at TMI-2). Rather, relocation is likely to extend to the lower core support plate and into the lower reactor plenum. Different flow paths are possible, either within the open space between the control blade guide tubes or directly into the guide tubes. As a consequence of the low power in the outer core region, the outer fuel assemblies may well suffer less damage and could even stay almost intact.

It is also envisaged that melted materials flow progressively into the lower plenum, instead of large ''pours'' of molten debris falling onto the lower head as in PWRs. The massive steel structures in the lower plenum and the remaining amount of water are likely to temporarily quench liquefied core debris, which will re-melt if no water injection occurs and attack structures in the lower plenum – such as instrumentation tubes and penetrations, the drain line and control blade drive penetrations – and the vessel lower head itself.

Figure A.1. Schematic view of a BWR RPV with internal jet pumps similar to the ones in Fukushima



Due to the large inventories of zirconium and steel in the core, it is unlikely that it is totally oxidised to form zirconium dioxide (ZrO_2) in the core region. In BWRs, steam starved conditions during core degradation are also possible in the core. Therefore, the molten pool in the lower plenum will contain a large proportion of metallic constituents; this has an impact on the stratification of the pool in separate layers, which in turn has an influence on the vessel failure mechanisms.

The assumptions above have an influence on the amount and timing of hydrogen generation that result from core degradation and hence, on the containment integrity due to the increasing pressure. Hydrogen deflagration or detonation inside the containment is not possible as the containments are nitrogen inerted during normal operation, but as the hydrogen gets stored in the containment, leakages to the surrounding rector building will challenge its integrity. This was seen in Fukushima Daiichi by the explosions in the reactor buildings.

Based on these considerations, one can expect significant differences between the in-vessel progression of a severe accident in a BWR and in a PWR. It would therefore be of prime importance to perform similar examinations at the Fukushima Daiichi damaged plants as those performed at Three Mile Island, in order to confirm or invalidate these assumptions and produce accurate modelling of core melt accidents in BWRs.

A.1.1.2.2 NUGENIA research priorities on core melt accidents

In the framework of the European Severe Accident Network of Excellence (SARNET) – now integrated into the NUGENIA platform on Generation II and III reactor research – a Severe Accident Research Priority (SARP) group was set in order to rank priorities of research in this field. This work took also benefit from the outcomes from the European ASAMPSA2 project, dedicated to Level 2 PSA studies [27].

The conclusions of that work were published in 2014 [28]. The research priorities identified for in-vessel accident progression are the following:

"The 1st issue concerns the in-vessel hydrogen generation (by oxidation of metal-rich melt mixtures) during the core re-flooding accident phase in a slightly degraded core. During this phase, the hydrogen is generated rapidly and may not be totally recombined by passive autocatalytic recombiners in the containment in this short time frame. This may increase the risk of hydrogen combustion and the potential consequences of an early containment failure (remark: relevant for PWR, BWR have been assumed to be nitrogen inerted). This issue is of <u>low priority</u> as several validated models exist to properly evaluate the phenomenon.

In a later accident phase with highly degraded core or in case of melt relocation into water in the RPV lower head (2nd issue) the prediction of hydrogen generation during in-vessel re-flooding is much more uncertain. Therefore, this issue is still of <u>medium priority</u>. It has to be pointed out that the risk significance of both situations depends strongly on the considered sequence and the degraded core geometry.

The 3rd issue concerns the core and debris coolability (rod failure, molten pool formation, molten pool and debris cooling, crust failure) and thermal-hydraulics within particulate debris during re-flooding. This research item addresses the re-flooding of a core not yet totally degraded, with the potential of stopping the core degradation process. This topic is still a point of discussion inside Level 2 PSA studies. Because of multidimensional effects, the models are quite poor. Therefore, users have to be cautious in using the models for reactor applications. The topic is still of <u>high priority</u>.

The 4th issue concerns the corium behaviour in a RPV lower head. This research item clearly addresses the details of the phenomena to be investigated for in-vessel melt pool behaviour. These are preconditions for possible in-vessel melt retention considerations. The in-vessel melt retention aspects and the improvement of the predictability of the thermal loading are a matter of high interest, especially for BWRs because of the control rod guide tubes (CRGTs) that influence the behaviour of melt in the lower head significantly and for reactors with low power density. Two *main phenomena are still of interest: heat flux to metal layer in layered melt configuration and 3-layer configurations including the transient stages. Further points are the evolution of the thicknesses of light and dense metallic layers and the formation of an insulating oxide crust at the upper surface. The simulation of the oxide pool is quite understood. However, the simulation of transient corium behaviour inside BWR and PWR lower heads must still be improved. It was suggested to change to* <u>*high priority*</u> *level, due to the increased current interest on IVR, especially after Fukushima."*

As a summary, regarding in-vessel accident progression and considering the large amount of research performed on core melt phenomenology, the main area of uncertainties remaining concerns the capability of re-flooding a degraded core, either in the core region or in the lower vessel head. This implies to get detailed knowledge of the core degradation mechanisms in the late phase of degradation, in particular corium relocation into the lower head. This knowledge is also important with regards to the RPV failure mode, which is recognised to be poorly understood for BWRs with many penetrations. Another research priority concerns the hydrogen generation in the late degradation phase of the accident, in particular in case of core re-flooding. The question of a recriticality especially in BWR cores were non-borated water might be used for reflooding, was not discussed.

These conclusions have been endorsed by NUGENIA in its Research Roadmap, which was issued in 2013 and revised in 2015 [29].

A.1.1.2.3 DOE reactor safety gap evaluation on severe accident analysis

In the aftermath of the Fukushima Daiichi accident, the DOE conducted a technology gap evaluation on accident tolerant components and severe accident analysis methodologies with the goal of identifying any data and/or knowledge gaps that may exist, given the current state of light water reactor (LWR) severe accident research [30], and additionally augmented by insights obtained from the Fukushima Daiichi accident. The process used a panel of US experts in LWR operations and safety with representatives from the DOE staff, DOE laboratories (ANL, INL, ORNL and SNL) and industry (EPRI, BWR Owners Group, PWROG, etc.) to identify and rank knowledge gaps, and also to identify appropriate R&D actions that may be considered to close these gaps. Representatives from the NRC and TEPCO participated as observers in this process.

Panel deliberations led to the identification of 13 knowledge gaps that were deemed to be important to reactor safety and are not being currently addressed by industry, NRC, or DOE. The panel noted that information from the damaged Fukushima Daiichi reactors provides the potential for key insights that could be used to address virtually all the identified gaps (i.e. 11 out of 13). As documented in [31], a subsequent DOE sponsored effort is underway to provide consensus US input related to desired examination information from the damaged Fukushima Daiichi reactors.

It is noteworthy that the first, second, and fourth-ranked gaps are all under the category of in-vessel core melt behaviour. In particular, the highest ranked knowledge gap is associated with assembly/core-level degradation; the second is core melt behaviour in the lower head; and the fourth is lower head failure. This is principally due to the fact that there are currently large differences between the two US codes for plant-level analyses (MELCOR and MAAP) in the prediction of core degradation behaviour for a scenario similar to Fukushima Daiichi Unit 1.

The panel concluded *that "These modelling differences reflect uncertainty that persists in the understanding of severe accident phenomena, principally due to a lack of experiment data that can be used to resolve such differences, particularly for BWRs. From a reactor safety viewpoint, this is an important issue as the modelling differences lead to large disparities in predictions related to the balance of the accident including in-vessel hydrogen production, lower head behaviour, and finally ex-vessel behaviour that affect thermal loads on containment and long-term debris coolability.*

During the early phases of in-core degradation, the two codes have adopted similar modelling approaches and, for a given scenario, produce similar results regarding initial fuel heat-up, oxidation, formation and relocation of molten core debris. The debris accumulates in the originally open flow channels, and the rod-like geometry is lost. The primary modelling differences arise when fuel assembly collapse begins. Both codes utilize time-dependent models to determine when collapse occurs, but the models are quite different and lead to differences in the timing of assembly collapse for a common scenario. Additional modelling deviations arise when considering particle bed formation and core-wide melt zone propagation.

The primary knowledge gap during this phase of the accident progression relates to the different methods used to model assembly blockages, the resultant porosity of these blockages, and how these formations influence the overall progression of in-core melt front propagation. There is currently insufficient experiment data or insights from reactor accidents that can be used to assess these modelling differences.

While the XR2-1 experiment was testing the response of lower core structures in a BWR, the core plate did not fail. Therefore, a knowledge gap exists on how the core plate might fail and how core material (particulate or molten) enters the lower plenum. Furthermore, very limited effort has been spent on the response of the upper internals in a BWR during a severe accident."

Regarding corium behaviour in the lower head, the panel concluded that "there is a lack of prototypic data for characterizing melt relocation phenomena, such as melt/water interactions, debris coolability, heat transfer from core materials relocating to the lower head, and the effects of raw water addition on these phenomena. This lack of data and the associated increase in uncertainty lead to significant differences in late phase models in severe accident analysis codes. Such differences significantly impact model predictions of subsequent accident progression phenomena, including ex-vessel behaviour. However, at this time, the consensus of the expert panel was that uncertainties in late phase lower plenum phenomena are dominated by uncertainties related to in-core behaviour, such as the timing, mass, composition, temperature, morphology, and heat capacitance of relocating core materials."

A.1.1.2.4 Conclusion

Regarding safety research interest of in-vessel examinations, we can conclude that prototypic full-scale data obtained from Fukushima Daiichi Units 1, 2, and 3 offers the unique opportunity to resolve many of the identified modelling uncertainties for BWR. Available information suggests that post-accident examinations could provide significant insights into key late phase core degradation and lower plenum phenomena**,** which should be considered as **high priority examinations from a safety research point a view.**

A.1.1.3 Decommissioning Interest

A.1.1.3.1 NDF's strategic plan

NDF has established a Strategic Plan for the decommissioning of Fukushima Daiichi NPPs [34]. A paramount challenge is the retrieval of the fuel debris from Units 1 through 3. Several approaches are under consideration for the fuel debris retrieval, depending on the actual situation which will be found in each unit: full submersion method (in which the whole Primary Containment Vessel (PCV) will be flooded) or partial submersion method (in which only the bottom of the PCV is flooded) with two options as regards the access to the debris (from the top or from the side).

The process and state of fuel debris formation as well as the location and distribution of fuel debris are keys to selecting fuel debris retrieval methods and approaches, which also ensure safety during retrieval, transfer and storage operations.

Before launching retrieval operations, it is of prime importance to conduct an overall assessment of the situation in the Reactor Pressure Vessel (RPV) and in the PCV of each Unit (1-3) and to understand the location and distribution of fuel debris. This is the objective of the first part of the Strategic Plan, which considers the combination of several means:

- evaluations by using external detectors (e.g. myon technology),
- investigations inside the PCV and then RPV by using robotics equipment and,
- analysis by using severe accident computer codes, in particular in the framework of the NEA BSAF project (see below).

Table A.4. Overview on relevant information important for the planning of the fuel retrieval operation, based on according to NDF's strategic plan (In-vessel phenomena)

Key issues	Information to be analysed/obtained
1． Fuel debris location and distribution	
Un-melted fuels and fuel debris remaining in the core region	Mass, form (crust or particulate), size, morphology (thickness, adherence to the structures and particulate size), and composition of fuel debris
	Possibility of un-melted (stub-shaped) fuels
	Impact of control rod melting on fuel melting
	Fuel debris formation phenomenology by the interaction with cladding and control rods
3. Criticality control	
Possibility of un-melted (stub-shaped) fuels in the core	Location and condition of un-melted fuel, and amount of fuel assembly
Conditions of various types of fuel debris	Mass, form, size, morphology, and composition of fuel debris at each location (element distribution, water content etc.) - Core and lower RPV - PCV bottom - S/C
	Chemical properties of fuel debris in terms of interaction with neutron-absorbing materials
4. Maintaining cooling function	
Decay heat of fuel debris	Mass, form, size, morphology, and composition of fuel debris at each location (surface area, porosity etc.) - - Core and lower RPV - PCV bottom - S/C
	Fuel debris properties (thermal properties etc.)
7. Feasibility and adoptability of fuel debris retrieval method	
Feasibility of cutting and removing methods of fuel debris	Fuel debris properties at each location (hardness, toughness etc.) -Core and lower RPV -PCV bottom -S/C, etc.
	Long-term stability during fuel debris retrieval operation (chemical activation by laser etc.)
Design of storage canister of fuel debris	Long-term stability of fuel debris during storage
	Fuel debris properties (possible dehydration and drying)
	Possible chemical reaction of fuel debris (gas generation etc.)

An iterative use of these means should progressively lead to a more accurate picture of the situation in the damaged reactors and to define the best approaches for fuel debris retrieval.

NDF's Strategic Plan identifies the information that is considered as important to obtain in order to plan and manage the fuel retrieval operations. The following table presents the information related to in-vessel phenomena.

A.1.1.3.2 NEA BSAF project

In order to support decommissioning operations, the CSNI has been conducting a benchmark study of the accident progression for the Fukushima Daiichi NPS units 1-3 accident has been conducted by some of the NEA member countries with a long experience in developing and using severe accident computer codes and methods of analysis, constituting an international project named Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) on the basis of a proposal by METI [35].

The objectives of the project are:

- To analyse accident progression of Fukushima Daiichi NPS utilising a common information database,
- To raise the understanding of severe accident phenomena, which took place during the accident, through comparison with participants' analysis results and with measured plant data,
- To contribute towards the improvement of methods and models of the severe accident codes applied in each participating organisation, in order to reduce uncertainties in SA analysis and validate the severe accident analysis codes by using data measured through the decommissioning process, and
- To contribute with analysis results on accident progression, status in the reactor pressure vessels and primary containment vessels, and status of debris distribution to a future debris removal plan."

The phase I of this project was launched in November 2012 and completed within two years after its start in 2012. Full scope analyses have been performed by several organisations, using currently available integral severe accident codes. In-vessel accident progression is analysed as well as the molten core concrete interaction (MCCI), but phenomena associated with fission product release from fuel and transfer within the installations is not analysed (it will be done in phase II). The time span for analysis of accident events is about six days from the occurrence of the earthquake.

Current analyses have been carried out against comparison of major measurements in the plant during the accident, such as core water level, pressure in reactor pressure vessel (RPV) and primary containment vessel (PCV) of units 1-3.

The general outcome stated in the BSAF Phase I final report is that, due to uncertainty introduced by the limited physical comprehension, several drastically different accident scenarios can reproduce the measured values relatively well. The main uncertainties identified during the benchmark and related to in-vessel accident progression are the following:

- *''The main uncertainty is to establish the precise location of the integral masses of the core debris, in the core region, lower head and in PCV. This aspect is entwined with information retained by the benchmark, that to say that models and assumptions for the core relocation deeply affect the progression of the accident, such as the capability of core re-flooding, lower head failure and the amount of corium relocating into the pedestal. It is believed that composition of the debris in the core region and lower plenum, particulate size could* *provide important insights to understand the progression and improve severe accident codes modelling;*

- *Beyond the localisation of the integral masses, the form (crust or particulate), morphology (crust thickness, crust adherence to structure and particulate size) and composition of core debris;*
- *Methods of penetration failure and ejection, hole ablation and debris retention in the lower head;*
- *Fuel crust on the penetrations in the lower head;*
- *Hydrogen generation (whether in core or during the MCCI phase) was found to be a major uncertainty in the simulations. This aspect might be strictly connected to the first point."*

Table A.5. Preliminary list of data needs from decommissioning operations, ranked based on their importance and feasibility given the methods of decommissioning

#	Item
1	Localisation of core debris masses
2	Samples of corium in the core region (composition, size) – Extent of the various corium layers
3	Samples of corium in the lower head region – Extent of the various corium layers
4	Samples of debris in the pedestal region – Extent of the various corium layers
5	PCV head flange conditions (gasket, flange deformation)
6	Lower head failure, pipe, wall conditions
7	Corium composition and size in the DW - Sample of corium debris if any
8	MSL creep rupture evidence
9	SRVs gasket conditions
10	SRVs state (e.g. stuck open in Unit 1)
11	Unit 2 RCIC turbine-pump
12	Unit 3 high-pressure coolant injection turbine-pump
13	Samples of concrete such as: solidus and liquidus temperature, depth of ablation, composition; ablation shape
14	Melt spreading area
15	Pedestal integrity
16	Estimation of corium level change
17	Presence of wall crust and frozen droplets
18	State of crust surface (cracks, volcanic moulds, debris bed)
19	Sampling of the crust in various locations
20	Temperature map of the crust by top view
21	Effect of the heat radiation on the surface
22	Aerosol deposited on different surfaces of the drywell
23	Evidence of hydrogen absorption by metal structures

On this basis, the report proposes a preliminary list of data needs from decommissioning operations, ranked based on their importance and feasibility given the methods of decommissioning.

''The data needed from the decommissioning are useful in particular for post-accident studies and severe accident codes improvement. Obviously all plant data represents an invaluable source of knowledge so that each data source should be retained and analysed. However information should be filtered to those considered fundamental. In the present chapter the data needs are organized in table form in a similar way for what is done in the Phenomena Identification and Ranking Table. Items are listed and ordered based on their importance and feasibility given the methods of decommissioning. It appears clear that data needs, relevant uncertainties and decommissioning procedures are all three topics dependent on each other that can proceed forward based on an iterative update and reconsiderations.''

The phase II of the NEA BSAF project has been recently launched; it will extend the previous analyses to fission product release and behaviour and will cover a longer time span.

A.1.1.3.3 Conclusion

Considering the key challenge of the fuel debris retrieval from Units 1-3, **the priority of in-vessel issues in terms of decommissioning interest is high** as long as it updates and validates the estimation of the final location and composition of fuel debris in the reactor vessels and the assessment of safety and risk of fuel debris remaining in the PCV, including potential recriticality issues.

A.1.1.4 Potential Examinations

The experience gained at TMI-2 shows that data acquisition in the reactor vessel will be essential both for properly designing fuel debris removal operations and for safety research purposes, i.e. improving our knowledge about in-vessel severe accident progression particularly for BWRs, in order to enhance our ability to manage core melt accidents.

On the basis of the previous considerations on BWR specificities and BSAF phase I outcomes, it would be useful to obtain, as far as possible, the following data related to in-vessel phenomena during the decommissioning process [31, 36-37]:

- End state (mass, density, composition[1], distribution, various morphologies : crusts, particulate debris, re-solidified molten phases) and peak temperatures of undamaged, damaged, and relocated core materials; particulate debris sieving;
- Evidence of mixing between fuel, cladding, fuel channel, control blades, structure and instrumentation materials;
- Evidence of stratification within once-molten materials;
- Physical characteristics affecting debris coolability (particle shape and porosity, cracks, gaps in crusts, research for the presence of boride phases, mechanical properties of debris);
- Evidence of effects due to sea water injection;
- Evidence of core plate failure (manner and location);
- Evidence of degradation of structures in the lower plenum;

1. If possible weight fractions and oxidation states of main components in oxide and metallic phases, indications of FP distribution and association in oxide and metallic phases.

- Evidences related to reactor pressure vessel failure mechanisms including analysis of mechanical properties of RPV samples;
- Evidence of crust anchoring.

The experience of TMI-2 examination and clean-up programmes showed that it was challenging to plan and prepare these operations before any data acquisition, through visual inspections for instance. As the end state of each of the three degraded cores is probably different from TMI-2 and from each other, plans and retrieval methods should be iteratively designed according to the information progressively gained during the operations. Schedule, feasibility and methods of samplings will notably very much depend on quantity, location and mechanical behaviour of materials remaining in the vessel and current observations and analyses do not provide sufficient information on the state and location of remaining materials in the vessel (with maybe an exception for reactor 1 for which there are indications that the core was severely degraded and for which little material remained in the vessel). Another challenge is the complexity of proposed detailed analyses that are to be performed for improving the understanding of the accident progression in-vessel in the three damaged reactors.

A.1.1.5 Ongoing R&D Activities

A.1.1.5.1 In Europe

Some experimental research programmes have been launched in accordance with the priorities determined by the NUGENIA Severe Accident Research Priority (SARP) group mentioned above:

- The PROGRES programme (IRSN), in collaboration with EDF, is dedicated to the study of degraded core coolability, using small and medium scale facilities (PRELUDE, PEARL), focusing on debris bed configurations [38]. The objective is to develop a model of reflooding of severely damaged cores in which 2D effects will be considered (e.g. by simulating flow bypasses around a compact debris bed, a configuration that probably existed during the TMI-2 accident). IRSN intends to investigate in the future the cooling of complex debris bed configurations which are reactor prototypic (e.g. heterogeneous debris bed, debris beds including compact zones) and the hydrogen production resulting from the reflooding;
- The CORDEB programme (NITI for IRSN, CEA, EDF & Areva) is a follow-up of the MASCA programme. It is devoted to develop models which are able to evaluate the heat flux profile along the RPV wall in contact with molten material during a severe accident in view of assessing safety margins for in-vessel melt retention. These models would describe the transient mass transfers between layers due to the changes of density of metal and oxide phases taking into account local conditions at the interface and not just a global equilibrium approach. An initial programme of about 20 tests was conducted between 2012 and 2015 using simulant corium materials in various configurations to quantify main involved processes: physical-chemical phase separation, layers inversion by turbulent instability, interactions between debris and layers;
- The In-vessel Melt Retention project involving 20 partners (European Commission, funded in the H2020 framework) was launched in June 2015 and aims at defining in-vessel melt retention management strategy for existing and future NPPs. It will address in particular in-vessel corium debris and molten pool behaviour through specific experiments and detailed modelling analyses to appreciate on deterministic grounds if in-vessel melt retention strategies are safe for all conceivable melt-down accidents configurations in LWR concepts relying on such strategies for powers of 1 000 MWe or above.

It is also to be noted that a State-of-the-Art report on in-vessel core degradation is being elaborated in the framework of the NUGENIA/SARNET CoreSOAR project.

A.1.1.5.2 In Japan

For fuel rod degradation, JAEA is designing an in-pile test to clarify conditions of fuel rod degradation due to oxidation and melting. A test fuel rod of about 300 mm long is heated in a research reactor simulating loss of coolant conditions in the irradiation experiments. The onset conditions of fuel degradation behaviour will be investigated by changing peak temperature and atmosphere mainly [39].

Because there is less information to evaluate in-vessel phenomena in BWRs, it is necessary to investigate melting and relocation behaviour of BWR fuel assemblies, to provide data and models to estimate accident progression, debris composition and core damage structure. Such data and information would also be useful to decommissioning at the Fukushima Daiichi NPS. Thus, METI/JAEA/CRIEPI/NSSMC (Nippon Steel and Sumitomo Metal Corp.) have started developing an apparatus and multi-scale model for investigating detail mechanism of BWR fuel rod and fuel assembly degradation. METI/JAEA conducts mid-scale experimental and analytical studies to clarify the degradation behaviour of BWR control blade and core support apparatus [40]. There are a lot of uncertainties for melt relocation in the BWR lower plenum, from the core plate to the lower head of RPV. Then, JAEA is preparing mid-scale experiment for melt relocation in the BWR lower plenum. Molten simulated core materials are poured onto the short BWR fuel assemblies and the core plate structure including control blade driver similar to the geometry of the XR2 experiment at Sandia National Laboratories. Tests with UO_2 may be conducted depending on the results with simulant materials. In addition, METI/JAEA/NSSMC is updating thermodynamic database by introducing a steel/slag database of the Japanese smelting industry, which includes a development of simplified version suitable for severe accident -analysis code.

For investigating impacts of seawater injection to fuel bundle coolability, NRA has conducted the seawater and boric acid mixture precipitation tests employing a partial length mock-up fuel bundle. In this test, flow blockage patterns in the rod bundle and the heater rod surface temperature were measured while injecting controlled concentrations of the mixture fluid and while maintaining the water level. Based on these data, a numerical method for predicting the precipitation pattern and the peak temperature under postulated injection scenarios under accidents are developed.

To obtain information about basic thermal-hydraulic characteristic of seawater and estimate the influence of seawater injection for alternative cooling during accidents, JAEA performs thermal-hydraulic experiments by using an artificial seawater, NaCl solution and pure water with and without boiling conditions. In these experiments, an internally heated annulus was used. As a result, it was confirmed that the thermal-hydraulic behaviour of the artificial seawater was evaluated by existing correlations (for example, Dittus-Boelter correlation for single phase heat transfer) with the physical properties of the artificial seawater [41].

A.1.1.5.3 In Korea

An experiment on the mechanism of penetration failure for APR1400 is under progress at Korea Atomic Energy Research Institute. The experiment is being performed in the VESTA (Verification of Ex-vessel STAbilisation) [42] facility using prototypic corium melt.

An experiment on the mechanism of penetration failure for Fukushima Daiichi nuclear power plant unit 1 is under progress at Korea Atomic Energy Research Institute. The experiments are being performed in the VESTA (Verification of Ex-vessel STAbilisation) [35] facility using prototypic corium melt.

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A.1.2 *Primary System and RPV Failure*

A.1.2.1 Background

The pressure vessel and the primary loop of light water reactors is one of the fundamental safety barriers. In case of a severe accident, the ability to cool the fuel within the pressure vessel is of highest importance. Therefore, detailed understanding of the thermophysical mechanisms of core degradation and relocation, corium molten pool formation and coolability has direct implications for safety analysis and severe accident management (SAM) strategies (see A.2.1.1).

Our current understanding of mechanisms with regard to RPV failure modes, melt release from the RPV or any other failure of the primary system and severe accident conditions is built on decades of extensive experimental and modelling efforts mainly related to PWR in addition to experiences gained from the TMI-2 (PWR) severe accident. BWR-related experiments investigating severe accident behaviour are rather limited (some for early core melt phase, see chapter A.1.1). None have been done with regard to RPV failure beyond some investigations of BWR drain line failure and instrument failure. The CORVIS (COrium Reactor Vessel Interaction Studies) project, conducted at the Paul Scherrer Institute in the mid-1990s, investigated the penetration failure of BWR in a series of experiments [1]. Fauske Associates, Inc. (FAI), under the sponsorship of the Electric Power Research institute (EPRI), performed experiments [2, 3, 4, 5] to investigate instrument tube penetration failure in PWR. Failure of other primary system components under high temperature and pressure loads is an issue of common interest in the non-nuclear industry and a large number of publications exist; experiments for nuclear reactor pipelines or components under severe accident typical loads are unknown.

In the case of the Fukushima Daiichi (BWR) accidents in units 1-3, the integrity of the pressure vessels is not expected to have been retained and RPV failure (not clear today if in any case RPV melt through) is believed to have occurred in all three units. Besides the question of the RPV failure mode – either a local one, one of the various penetrations or global one due to creep rupture – the questions of the RPV pressure at its failure and the melt relocation mode are of main importance – either at high pressure causing high pressure melt ejection phenomena followed by a challenge to the containment integrity through direct containment heating (DCH) phenomena or at low pressure with moderate melt relocation downwards from the RPV bottom along the multiple structures connected to the RPV bottom (penetrations, instrumentation pipelines, control blade driving rods and mechanisms and its supporting equipment, etc.).

High pressure melt ejection and DCH are characterised by the interaction of molten core material, blowdown gases from the RPV, and the containment atmosphere. The time scale for the interaction is typically short (of the order of tens of seconds). During this period, the energy deposition to the containment atmosphere could result in high containment pressures and temperatures, which could potentially lead to the failure of the containment and subsequent release of radionuclides into the atmosphere. Larger amounts of melt particles would be transported with the gas to locations outside the cavity (pedestal in a Mark I) within the containment. In addition large amounts of hydrogen would be generated by oxidation reactions of hot metals with steam. With regard to the current status of analyses in the BSAF phase 1 project, a high pressure RPV failure was predicted in some analyses in unit 1 of Fukushima Daiichi [6]. A detailed study for a Mark I containment is presented as well by [7] and shows another effect which is described in the following.

A failure of a pipe line connected to the RPV or a component (e. g. a safety valve or gaskets of it) under high temperature and pressure loads caused by the hot gases released from the core in a severe accident and flowing through the pipeline and/or the component is another issue of special interest. It is of safety significance, as a high pressure failure of a pipeline before a high pressure failure of the RPV would lead most probably to much less loads and challenges to the containment and would therefore bear less challenges to its integrity. With regard to the current status of analyses as of BSAF phase 1 project, a high pressure main steam line failure or a safety valve gasket failure was predicted by several simulations for all three units before RPV failure was predicted [6]. Such issues are typically analysed in severe accident analyses performed as part of Risk Studies or in other studies like the SOARCA project [8]. The results of the recently finished ASAMPSA2 European project [9] underlines that the issue of "Induced failure of reactor circuit components in high pressure sequences" is one of other important severe accident phenomena in the event tree of a PSA study.

As an example a short summary of related modelling information from the recently published SOARCA report [8] follows. By applying modern analysis tools and techniques, the SOARCA project developed a body of knowledge regarding the realistic outcomes of severe reactor accidents. This study focused on providing a realistic evaluation of accident progression, source term, and off-site consequences for the Peach Bottom Nuclear Power Station (BWR) and the Surry Nuclear Power Station (PWR). The focus here is on the BWR plant, as it is generally similar to the plants in Fukushima Daiichi. The analysis comprises a short term and a long-term station blackout (SBO) scenario. There are two main different issues which needs attention in a BWR:

"The possibility that a cycling SRV would randomly fail to open, or fail to reclose after opening under the conditions to be expected in a severe accident, needs to be considered. Several hundred SRV cycles were calculated for SBO accident sequences [6, 7] which raised concerns that random failures should not be ignored. Probabilistic modelling of a SRV failure was done.

The potential for creep rupture of a BWR main stream line (i.e. piping or RPV nozzle) is another issue of importance. The creep rupture model in MELCOR was applied; it is a direct translation of the hot leg creep rupture model used in the Surry MELCOR simulations in the same project. The typically used Larson-Miller criterion is used (see below). Creep rupture of the structure is assumed to result in an opening to the drywell atmosphere equivalent to the full open area of the main steam line."

The use of integral codes, such as MELCOR or ASTEC or others (see chapter A.1.1), provide typically only simplified models to consider such effects. In addition specialised structural mechanics codes are used to analyse the behaviour of the pipeline using thermal and pressure boundary conditions provided by the integral codes. The problem arising is how to predict a rupture time for components submitted to a given time dependent stress and temperature load. Considering the different consequences of different rupture locations, it is not only of interest whether a rupture occurs at all but it is also important to know which component will fail first. Such predictions rely on tools classically used in mechanics. A damage function may be associated to each structure. Its value is in the range [0; 1] and rupture occurs when the damage value is one. For creep phenomena, the damage is commonly computed through the Larson-Miller correlation linking stress and strain. In the Larson-Miller correlation, the Larson-Miller Parameter (LMP) is correlated to the logarithm of the piping stress using fitting parameters. These parameters depend on the material properties of the hot components.

Sampling and metallurgical inspection of the TMI-2 vessel and in-vessel debris bed provided new and detailed insights regarding melt pool formation and cooling in pressure water reactor for the specific accident scenario. The importance of crust formation and the formation of cooling channels between the vessel and the crust was previously not known (see chapter A.1.1). Following those investigations, a large number of scaled integral experiments, separate effect experiments and numerical simulations have been carried out in order to improve the understanding of RPV failure. In the following, some of the most important experimental and modelling activities are described.

The Lower Head Failure (LHF) programme [10] was carried out at Sandia National Laboratories between 1998 and 2002. It comprised a total of 8 tests and was initially motivated by the inability at the time of simulating the outcome of the TMI-2 accident, namely that the vessel did not fail. The programme focused on high internal pressures as was the case in TMI-2, and scaling was applied to achieve representative membrane stresses. The vessels were heated from the inside by radiation, which could be varied along the height of the lower head so as to achieve centre peaked, uniform or edge-peaked distributions.

Important results from the LHF programme were:

- *"For vessels without penetrations failure occurred locally at overall vessel deformations of 10 – 30 %. With local heating, the location of severe deformation and failure coincided with the location of peak temperature.*
- *With penetrations, the vessel typically failed at the weld of a penetration due to global deformation.*
- *A consistent material database was developed. Existing databases for tensile properties and creep were critically reviewed and correlated, and a modified creep law was proposed and fit to the relevant database.*
- *Additional material property measurements were carried out, covering, e. g., elastic moduli, yield stress and creep.*
- *Severe accident code methodologies, employing simplified "engineering" methods to predict creep and time to failure were critically assessed and a number of recommendations were given.*
- *Finite Element Model (FEM) methods were found to predict the failure location reasonably well. However, the time of failure was generally overestimated, a deviation that could be attributed to uncertainties in the modelling of creep."*

The NEA Lower Head Failure (OLHF) project [11] was a continuation of the LHF programme. Here, lower RPV pressure was studied (2-5 MPa) compared to the LHF tests (10 MPa). Scaling was applied so that representative through-wall temperature gradients were reached, i.e. 200 to 400 K, compared to about 25 K in the LHF tests. The actual pressure in the OLHF experiments was increased to preserve membrane stress.

Important results from the OLHF programme were:

- *"Use of representative through-wall temperature gradients demonstrated the importance of stress redistribution on vessel deformation. The net effect was that failure occurred at higher temperatures than in the corresponding LHF tests. Since penetration failure was governed by global deformation, these tests also showed failure at higher temperatures.*
- *Failures were typically local, where the ratio of membrane stress to yield stress was largest.*
- *Failure was consistently found to occur at roughly 30 % global strain, for uniformly heated vessels without penetrations. With penetrations, failure occurred at much lower global strains of ~10 %. It was recommended that metallurgical and materials properties at weld locations should be addressed in more detail.*
- *It was found that variations in material data between different vessels, as well as laboratory-to-laboratory differences in measured material properties were large enough to influence model predictions substantially.*
- *The chemical composition of the RPV steel may influence the mode of failure. Characterisation of failure sites performed by CEA showed that at high temperatures, the LHF material exhibited brittle failure whereas the OLHF material exhibited ductile failure. The difference was attributed to the higher sulfur concentration in the LHF material.*
- *The influence of the ferrite to austenite phase transition (~1 000 to 1 100 K) was seen in the experiments. However, it was concluded that the effect of this transition is less significant*

than previously assumed, due to the fact that it occurs at different times through the RPV wall.

- *Benchmark simulations related to the OHLF project led to the following results [12]:*
  - *The time to failure could in general be quite well predicted (within 15 min). However, such accurate predictions cannot be expected in a realistic scenario, since the experiment were conducted with a heating rate in which long term creep did not have sufficient time to develop.*
  - *The geometry at failure, e.g. the maximum vertical displacement, was not well reproduced in calculations. This was attributed to the use of too simple failure criteria based on 1D strain.*
  - *The failure location was quite well predicted, as in the LHF case.*
- *Sensitivity studies showed that failure time is extremely sensitive to creep and damage parameters."*

It was concluded by the OHLF Project Committee that the LHF and OHLF experiments have laid a sound foundation for modelling of lower head failure. Some recommendations given were [11, 12]:

- *"Materials data correlations should be reassessed and the influence of chemical composition should be taken into account.*
- *The material databases should be extended to higher temperatures (T > 1 300 K)*[2].
- *Mechanical properties should be derived in biaxial testing, in order to assess the appropriateness of using uniaxial test data in failure criteria.*
- *FEM should be used to benchmark experiments with penetrations.*
- *Separate effect experiments on penetrations should be considered."*

The FOREVER experiments were performed at the Royal Institute of Technology (KTH) between 1999 and 2002 [13]. Unlike the LHF and OLHF experiments, simulant melt material was used, with internal electrical heating, and again PWR geometry without lower head penetrations was modelled. Pre- and post-test calculations were done using FEM. The experiments showed, as expected, that the maximum lateral heat flux occurred at the upper boundary of the melt pool. This is also where the failure occurred. Following the FOREVER experiments, analytical work was carried out in order to improve the detail of understanding of various phenomena.

Some conclusions were:

- *"Creep occurs when high temperatures (T > 600 °C) and high pressure (p > 1 MPa) are present.*
- *Regions of maximum heat flux (hot focus) exhibit the highest creep strains, which leads to a reduction in load-bearing cross section and eventually failure.*
- *The level of temperature and pressure has an influence on the failure timing but not on the failure position.*
- *The failure time can be predicted with an uncertainty of 20-25%. This uncertainty is connected to the uncertainties in viscoelastic properties of the RPV steel at high temperatures.*
- *The lower region of the vessel head experiences lower temperatures and higher strength compared to the hot focus region.*

2. Limited material properties data exist at temperatures up to 1500 K (see "TMI-2 Vessel Investigation Project Integration report", NUREG/CR-6197, March 1994.)

- *Further research has been conducted within the SARNET European project [14] on the interpretation of creep failure experiments. Specific high temperature creep measurements [15] have shown the importance of the sulphide particles in steels having high sulphur contents, within the range of the considered steel grades."*

Other experimental programmes with prototypic materials as RASPLAV and MASCA have been devoted to the study of corium behaviour in the lower head [16]. They are summarised in chapter A.1.1. It has been found that the corium influences the RPV wall not only by raising the temperature and eventually melting material. There is also corrosion and formation of eutectoid phases with much lower melting point than the two separate materials. This interaction has been studied separately in the METCOR experiments [17] with VVER and PWR steels in a temperature range limited by external vessel cooling. Experimental data on corrosion kinetics were used in analysis of in-vessel melt retention in some VVERs and PWRs. Calculations have shown that, if heat fluxes to the cooling water do not exceed CHFs, such internal vessel corrosion has a minor effect on the vessel strength since relatively thin but cold external surface layer of the vessel steel can withstand thermomechanical loads, at least during low pressure severe accident scenarios. Again, this assumes successful external cooling.

Specific issues in BWRs

The progression of a severe accident in a BWR is expected to be slightly different to that in a PWR in terms of core degradation, melt relocation and formation of a debris bed in the lower plenum. This is caused by the differences in core geometry, materials and components as CRGT located in the lower plenum of the RPV (see chapter A.1.1). As mentioned already in A.1.1., there are some specific design features that are expected to create differences in the accident progression [18].

- The lower head is geometrically complex with a forest of CRGTs cooled by water in normal operation and of IGTs. The free volume in the lower plenum of the RPV is significantly larger than that of a PWR and partly occupied by CRGT and IGT. This may change melt fragmentation, in-vessel debris formation/coolability and delay formation of a melt pool. Relocation of material into the CRGT cannot be excluded. Also, when a molten pool is eventually formed, it may contain much more Zr metal and different mass of molten steel in comparison with PWR case.
- BWR lower head is composed of various types of materials of RPV, weld-overlay cladding, control rod guide tubes, stub tubes, welds, etc.
- The vessel wall is thinner in a BWR than in a PWR, while the bottom head is thicker due to the large number of penetrations. The operating pressure in a BWR is lower than in a PWR.
- There are many different penetrations through the lower head in a BWR, which construction details and location of welds differing between BWR designs. Some PWR even have a drain line connected to the lowest point of the bottom head.
- Most of the penetrations are not empty, as there are materials like the control rod driving piston or instrumentation cables in it.
- The expected mode of RPV failure (while not fully justified) is through a penetration failure, e. g. by the weld melting. IGTs are considered to be vulnerable because they are smaller than CRGTs, some of them might be even empty (not used).
- The effect of corium on the tube weld, under conditions of debris reheating and global vessel deformation, has not been studied experimentally. Recent thermomechanical analyses of BWR failure modes carried out at the Royal Institute of Technology (Sweden) indicate earlier failure of IGTs than other failures. It was found that in several reference scenarios, the mass of liquid accumulated in the debris bed prior to the IGT failure is limited. This will naturally limit flow rate of melt from the vessel lower head into the reactor pit.

In conclusion one should consider all possibilities to gain additional information from Fukushima Daiichi decommissioning about severe accident progression in BWRs in particular, about RPV failure and failure locations by detailed inspection and sampling.

A.1.2.2 Safety Research Interest

After termination of the operation of the emergency core cooling systems (ECCS), core degradation started under different conditions in all 3 units at Fukushima Daiichi and fission products were released from the fuel. It is speculated that fission products were primarily released through one steam line with the operating safety valve and the connected pipe line down into the water pool in the wetwell through spargers. Deposition is possible in the water pool and on this way in the cooler regions, such as the steam separators or dryers in the RPV or the steam lines.

Even though Fukushima Daiichi plant examination results so far do not indicate a safety valve failure in either position or indicate a high pressure main steam line failure as predicted specifically for Unit 1 by some of the partners [6], the question of a failure of a main steam line, or a safety valve or even its gasket is of high interest.

The response of the RPV following a severe accident will depend on:

- the pressure in the RPV during melt relocation and vessel/penetration heat-up,
- the water level in the RPV lower head at the time of met relocation,
- the amount, composition, and distribution of the melt (e. g. molten pool, debris bed) which is relocated to the lower plenum,
- the system induced stresses in the RPV wall (low or high pressure scenarios) or the welds of the penetrations,
- the resulting thermomechanical load on the RPV wall, and
- the precise steel composition, in particular the Sulphur content, as it affects the creep law.

The mode of RPV failure together with the RPV failure pressure in turn has important consequences for the further development of the accident (e. g., energetic melt release vs gradual release, and the risk of steam explosions in case of flooded cavity as well as of high pressure melt ejection and direct containment heating).

Considering research performed during the last decades, it seems clear that the thermo-mechanical response of the RPV is complex but uniform static loads are relatively well known, at least for PWR vessels with no or small penetrations. Open issues of high safety significance for BWR are:

- The RPV failure mechanism.
- Possible uniformities and transient behaviour of vessel/penetrations during interaction with debris bed/melt under conditions of gradual temperature increase.
- Oxidation effects in the system influencing material degradation and producing additional “chemical” heat.
- The effects of corrosion and ablation of the RPV steel, penetrations and welds at high temperatures.

Clearly, investigations of the lower head, visual or by sampling will be of **high importance** for the advancement of safety research of LWRs in general and of BWRs specifically. Although the behaviour of the vessel itself is relatively well understood, investigations of its state following a severe accident can improve our understanding mainly of the late stage of in-vessel accident progression. The global deformations of the vessel are very indicative for understanding debris and molten pool behaviour in the RPV lower plenum. The place of the vessel or penetration opening will clarify the failure mode and time. Material studies of local samples cut in different parts of the vessel will reflect temperature history; thermo-chemical interactions that have taken place and possible non-uniformities in the system. Similar to the TMI-2 vessel investigation, mapping of the (maximum) temperatures of the vessel internal layers could be produced based on the examination of the samples [19] and this will address the question about a possible hot spot, its location and size. Material relocation down from the RPV into the cavity along the various structures is discussed in chapter A.1.3.

Regarding the possible failure along one of the main steam lines, it is of **high interest** from a safety research perspective, to know not only if, but also where and when, such a failure occurred.

A.1.2.3 Decommissioning Interest

From the perspective of decommissioning, there is **medium interest** in estimating the reactor failure points since it may be helpful in determining the melt relocation and fission product release into the containment; this contributes to the decommissioning decision. On the other hand, there is less interest in cutting samples from the lower head region of the pressure vessel to achieve that objectives, as it will take long times before some metallic components of the reactor pressure vessel can be sampled. Once the location and extent of failure of the vessel has been established, efforts will be focused on characterising the form and morphology of fuel debris in the structures below the lower head. Therefore, the detailed investigation of the pressure vessel itself has **low priority** from a decommissioning point of view.

Regarding the possible failure of one of the main steam lines, or one of the corresponding safety valves, it should be of **medium interest** from a decommissioning perspective, since it would represent a major source of fission product release. Therefore high activity is expected along this path, something that needs to be considered in the planning of decommissioning.

Overall, from the perspective of decommissioning, the interest is **medium** with regard to knowing the reactor failure points. It may be helpful in determining the melt relocation and fission product release into the containment; this contributes to decommissioning decision.

Table A.6. Overview on information possibly gained from the decommissioning operation

Key issues	Information to be analysed/obtained
1. Fuel debris location and distribution	
2. Fuel debris remaining in the RPV lower head	Mass, form, size, morphology, and composition of fuel debris
	Relocation behaviour of fuel debris at core support plate
	Relocation behaviour of fuel debris and debris bed formation in the RPV lower plenum (Including interaction with a “forest” of CRGT etc.)
	Lower head failure mode (penetration etc.)
3. Fuel debris remaining around the control rod drives (CRD) under the RPV lower head	Mass, form, size, morphology, and composition of fuel debris
	Amount of fuel debris remaining on and inside of the control rod penetration.
	Amount of fuel debris adhere to the outer surface of the CRD housing

A.1.2.4 Potential Examinations

Possible leaks in the primary system at the main steam line with the operating safety valve(s) should be identified. This may be supported by radiological investigations or other inspections.

Most likely, the modes of RPV failure can be preliminarily determined once the RPVs of reactor 1-3 can be inspected. In addition, as explained above, the following examinations and samplings are proposed:

- Visual inspections of the pipe lines connected to RPV within PCV and the safety valves. Radiological investigations of the primary system may thus reveal details, e.g. about the timing of fuel failure, fission product release and transport rates, radioactive aerosol transport, deposition and resuspension.
- Characterisation of failure(s) and hot spot(s) by visual observation of RPV vessel external surface and penetrations. The determination of the failure location may provide information on late in-vessel accident progression.
- Characterisation of the deformation possibly first by visual inspection. The degree of global creep is of interest in understanding the late stages of the accident, before RPV failure.
- If possible, representative full cut samples of RPV wall, bottom and other main components inside PCV should be acquired. Later mechanical or metallurgical investigations may provide information on peak temperatures experienced during the accidents.
- Samples from the pressure vessel steel ingots, if they have been stored from the construction period, could be used to determine creep laws for the actual deformed or failed vessels.
- Samples from the interface between melt/crust and the RPV wall would be highly valuable for later chemical, mechanical and metallurgical examination. Special attention should be paid to positions close to penetrations, and to the upper part of the melt pool, if it was formed.

Possible leaks in the primary system at the main steam line with the operating safety valve(s) should be identified. This may be supported by radiological investigations or other inspections.

Examinations with regard to material relocation outside the RPV are discussed in chapter . A.1.3.

A.1.2.5 Ongoing R&D Activities

There is an ongoing CSNI activity within its working group WGIAGE. Under the headline “Components and Structures under Severe Accident Loading (COSSAL)”, the group compares structure mechanical analysis methods for integrity assessment of metallic components of selected pressurised and boiling water reactors under severe accident loading, especially high temperatures which may occur during core melt scenarios. The main safety-related issues to be treated from generic point of view are:

- Which metallic component of the pressure boundary loaded by selected severe accident scenarios with high temperatures fails first in selected PWR and BWR with/without consideration of ageing aspects?
  - PWR: surge line, main coolant line, steam generator tube and reactor pressure vessel
  - BWR: main steam line, relief line to safety valve and reactor pressure vessel

- What are the uncertainties for the integrity assessment of the metallic components and quantification of safety margins against failure?

Melting test of the penetrating tube through the BWR-RPV lower head was implemented by the Institute of Applied Energy (IAE). Test sections with a penetrating tube had been prepared by IRID/Hitachi-GE and tests using real corium was performed by Korean Atomic Energy Research Institute (KAERI). The main purpose of this test is to investigate the behaviour of the penetrating tube through the BWR-RPV bottom wall which is attacked by the corium relocated to the lower plenum for the validation of the analytical model. It is recommended to make use of the results obtained in earlier studies done at the Paul Scherrer Institute (PSI) [1] and Fauske Associates, Inc. [3, 4, 5].

JAEA is developing a detailed analysis method to predict time and location of rupture for RPV lower head of BWRs considering creep damage mechanisms based on three-dimensional thermal-hydraulics (TH) and thermal-elastic-plastic-creep analyses. Such method may contribute to assess progress of severe accident and to estimate situation of inside of the reactors of NPPs, and to improve existing severe accident codes, e.g. MELCOR and THALES2, which judgement methods for RPV rupture are simple such as temperature and/or stress criteria.

In the study performed by JAEA in order to expand materials database and verify the creep constitutive equation and rupture model, materials data is obtained under uniaxial and multi-axial stress conditions at high temperature near melting point. To investigate the inhomogeneous temperature and stress distribution by geometrical complex of BWR lower head, a detailed 3D model of RPV lower head with control rod guide tubes (CR_GTs) and shroud supports are constructed and a 3D thermal hydraulic analysis of simulated molten pool and creep deformation analysis of lower head are performed using ANSYS Fluent/Mechanical finite element code. It is found that the possibility of failure mode for BWR lower head are both the penetration failure which is melt-through of the guide tube, local rupture and global rupture of lower head by creep deformation mechanism and the melting collapse mechanism.

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A.1.3 *Ex-Vessel Phenomena*

A.1.3.1 Background

Despite the highly efficient accident prevention measures adopted for the current Generation II nuclear power plants (NPP) within Accident Management Programmes implemented after the TMI-2 accident, where there are more demanding ones for the Generation III plants (e.g. external RPV cooling, core catcher, core spreading and cooling), some accident scenarios with a low probability may remain resulting in severe accidents with core melting, RPV damage (at low pressure), melt relocation into the cavity and finally molten core concrete interaction (MCCI). As mentioned and discussed in chapter A.1.1 and A.1.2, the research activities in the various countries started after the TMI-2 accident.

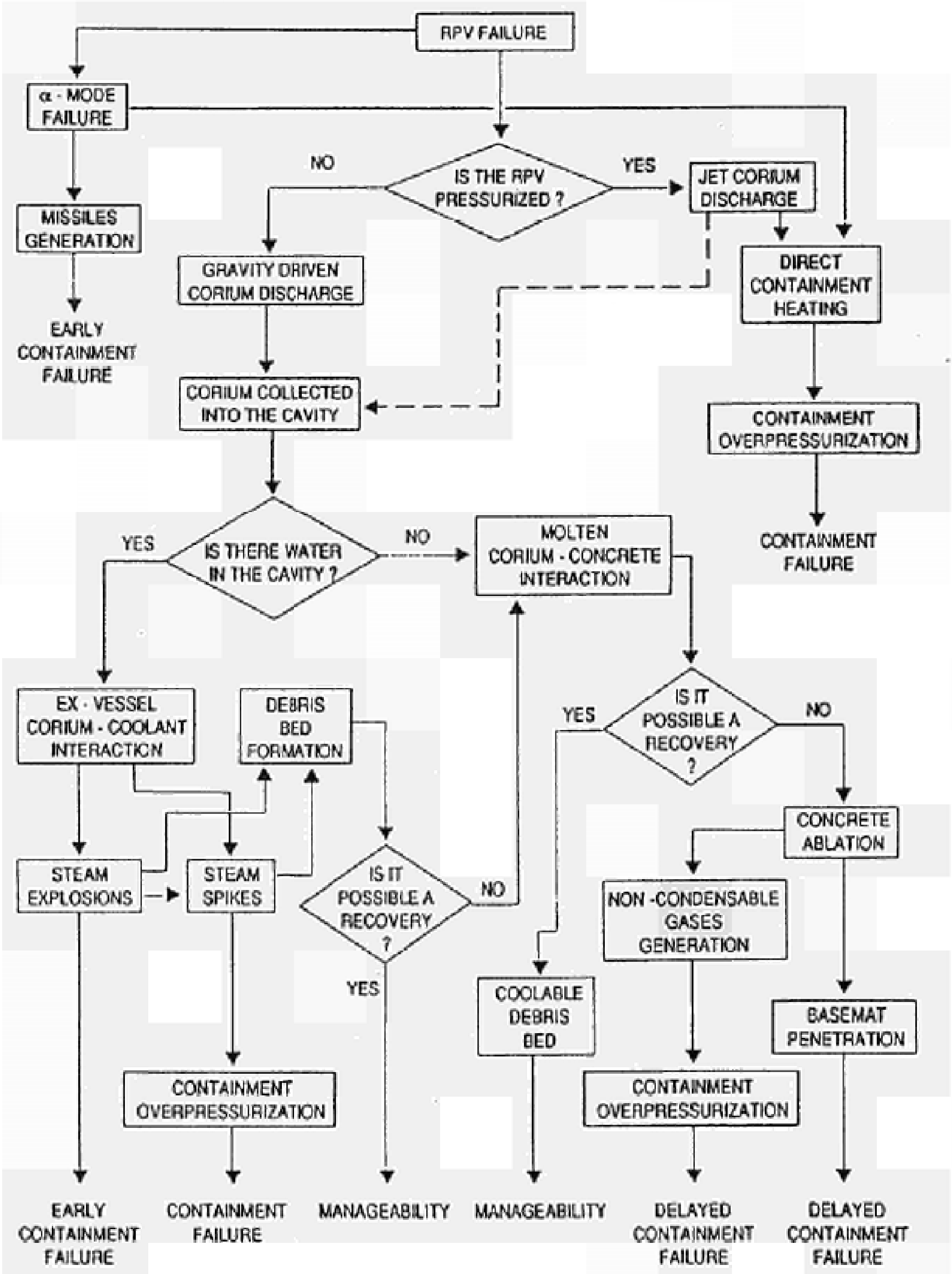
A.1.3.1.1 Basic phenomena of ex-vessel phase except hydrogen related issues

Studies for relevant phenomena of the ex-vessel phase of a severe accident, defined to start after RPV failure occurred, are discussed here. An illustrative description of severe accident phenomena in the ex-vessel phase in LWR plants with inerted containment atmosphere (as in the BWR in Fukushima Daiichi) was found as expected in earlier work summarised in [19] (Figure A.2). This means that hydrogen related aspects (e.g. generation, release, combustion and mitigation) and possible containment challenges are not discussed here; these are covered in chapter A.1.6 of this report.

The ex-vessel phase starts with melt release from the RPV (here considered only under low pressure) into the reactor cavity together with any water, steam and non-condensable gases. Depending on the presence or absence of water in the cavity and on the configuration of the cavity and other connected rooms, different sequences can occur:

- In a dry cavity, dry melt spreading of the core melt to adjacent rooms is possible, if e. g. open connections at floor level exist. The contact between molten core debris and the concrete of the cavity floor/basemat finally leads to Molten Corium Concrete Interaction (MCCI) if the corium is not adequately cooled, with the consequent decomposition of concrete and challenge to the containment integrity by various mechanisms. Subsequent pours of corium may affect the phenomena.
- In a sufficiently flooded (wet) cavity, molten fuel (corium) will interact with coolant (water) (called Fuel Coolant Interaction – FCI) leading to partial or total jet fragmentation. Another phenomenon called steam explosion may occur in which heat is so rapidly exchanged from the corium droplets to water/steam that the medium cannot accommodate the increase of volume without shock waves. Even if the energy of this explosion does not lead to containment failure, it will affect the debris bed particle distribution and its coolability. If the debris bed is not sufficiently cooled, corium will interact with the concrete and MCCI is started. In the cases with incomplete melt jet fragmentation (mainly shallow water configurations) there is underwater corium, spreading into adjacent rooms and MCCI may also start there, if the melt is not cooled. In summary, many factors affect the development of the MCCI phenomenon such as the availability of water in the reactor cavity, the containment geometry, the composition, size distribution and amount of the core debris, the thermodynamic condition of the core debris and the type of the concrete used for the basemat construction [27].

Figure A.2. Main severe accident phenomena in the ex-vessel phase as expected in LWR plants with inerted containment atmosphere



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A.1.3.1.2 Link to Fukushima Daiichi accident progression after RPV failure

As indicated in the previous chapter, in the case of the Fukushima Daiichi accidents in units 1-3, the integrity of the RPV could not be retained, and vessel failure is believed to have occurred in all three units. Melt release from the RPV is believed to have occurred at least in units 1 and 3, the units with the largest extent of core degradation. All codes applied to the analyses within the NEA BSAF project phase 1 [8] assumes that the melt released from the RPV relocates completely down to the cavity (pedestal) bottom; melt retention at the various structures below the RPV and in the pedestal of the BWR cannot be calculated. The pedestal melt either remains or may occupy the two pump sumps located there and MCCI starts or melt may spread further out of the pedestal through an opening at floor level to the circular drywell floor and MCCI starts in all locations. If melt has spread to the drywell floor it may come into contact with the containment steel wall and challenge its integrity. Further, in many of the analyses performed in the BSAF project phase 1, successful melt cooling was not predicted; MCCI hence continued until the end of the analysed time span.

In Fukushima Daiichi, since a certain level of water was observed at the bottom of the PCV, no large containment failure due to MCCI was predicted to date; at least no melt has been found outside of the containment. Water leakages are obvious for all three Units, e.g. for the Unit 1 containment through a sand cushion drain pipe [9, 23].

The further analyses in the BSAF project should show if temporal MCCI could be one reason for this damage causing containment leakages. Visual inspections of the units 1 and 3 executed by TEPCO could so far not answer the question if melt was released from the RPV and has accumulated in the pedestal, or if melt is retained on the various structures below the RPV, or at least if melt has spread outside the pedestal (cavity). Based on this expected accident progression, the focus of the research activities and needs should be on the phenomena of melt spreading, melt/debris coolability and MCCI here to a lesser degree on FCI.

After the accident at Fukushima Daiichi, international research predicted lower head failure and corium-concrete interaction at Fukushima Daiichi Unit 1 (1F1) using several different system-level code analyses, including MELCOR v2.1 and MAAP5. Although these codes capture a wide range of accident phenomena, they do not contain detailed models for ex-vessel core melt behaviour. However, specialised codes exist for analysis of ex-vessel melt spreading (e.g. MELTSPREAD) and long-term debris coolability (e.g. CORQUENCH). The U.S. DOE funded an analysis to further evaluate the ex-vessel core debris location and extent of core concrete interaction for Unit 1 using MELTSPREAD and CORQUENCH to provide rigorous "best estimate" predictions and analysis of ex-vessel core melt accident progression and final debris configuration [28, 29, 30]. This was followed by another analysis, funded by US NRC, to investigate why the MELCOR v2.1 code, compared to the MELTSPREAD and CORQUENCH 3.03 codes, yield differing predictions of ex-vessel melt progression [31].

These studies provided an initial assessment of ex-vessel core debris spreading and long-term debris coolability for unit 1F1 using tools that were considered to be state-of-the art at that time. Based on lessons learnt from these studies as well as ongoing findings from the Fukushima forensics work, these two models are currently being extensively upgraded to account for additional factors that include: i) melt stream interaction with below vessel structure, ii) melt stream breakup with water present on the pedestal floor, iii) melt spreading under the presence of an overlying water layer, iv) providing the ability to perform multi-nodal core-concrete analysis to examine local cavity ablation behaviour and coolability, and v) detailed water inventory boil-off modelling to optimise severe accident water management strategies for boiling water reactors (BWRs). Based in part on these results, the user community is in the process of updating systems codes; specifically MAAP and MELCOR, to improve and reduce conservatism in their ex-vessel core melt models.

A.1.3.1.3 Summary of knowledge on basic phenomena of ex-vessel phase except hydrogen related issues

The issues of FCI, debris coolability, melt spreading and MCCI have been extensively studied in the past decades in NEA member countries and abroad [4], [10] - [19] , even though some gaps in knowledge remain.

The main sources of R&D work performed in NEA member countries on FCI are related to NEA projects SERENA 1 and 2. The in-vessel results from SERENA-1 project [10] show that steam explosion loads are far below the capacity of an intact vessel. On the contrary, the spread of ex-vessel steam explosions load is often below the typical strength of containment structures.

Thanks to SERENA 2 [11], better consistency among the various FCI codes was achieved in the prediction of ex-vessel steam explosion, and the calculated loads were somewhat less than those reported previously. However, there remains still a large scatter in the prediction of ex-vessel steam explosion loads. It is believed that the new data from KROTOS and TROI experiments [10] will be useful in improving further the FCI models and codes and also validating them, which may help resolve the discrepancies between various code predictions. Also new results on steam explosion in shallow waters have been provided by the PULiMS and SES experiments (using simulant materials) in Sweden [11] and have drawn renewed attention on the risk of violent steam explosions during melt spreading in shallow water.

In the 1980s, corium spreading had been studied in the United States [12] and Japan[13] and later in Europe in support of the EPR spreading core catcher concept within several European projects (CSC [14], COMAS [15], ECOSTAR [16]). Dry spreading is quite well understood. It is controlled by the melt viscosity which increases as corium is cooled and solidifies. On the contrary, spreading in underwater situations has not as much been studied and many unknown issues remain. It seems to be controlled by the surface crust strength [15, 16].

A debris bed can be formed by corium water interaction. Coolability of corium debris bed has been extensively studied within SARNET phases 1 and 2 [4, 6, 7]. Whereas cooling of a 1D monodisperse debris bed is quite well modelled, the current modelling focus is on the study of 2D effects (e.g. Layered beds, conical beds, side flows…) and to the effect of particle shape and size distribution [17]. Advances in modelling have led to a novel dimensionless formulation of the debris bed dry-out problem [18].

Investigation of MCCI related phenomena including the release of fission products from the melt started in the eighties and late nineties with a significant number of experiments performed worldwide (BETA, SANDIA Large Scale, ACE/MACE experiments) thus increasing the general understanding of most of the phenomena and allowing to develop and assess MCCI codes like CORCON and WECHSL. A summary of all these activities is available in a 1995 CEC report in a framework of the MCCI Programme [19]. The research led to a basic understanding of the MCCI process: At typical corium temperatures, the concrete is ablated and steam and non-condensable gases are generated, hydrogen originating from the oxidation of corium metals and concrete rebars by steam, and steam and carbon dioxide from the concrete itself. The influence of the different concrete compositions (Table A.7) on the erosion behaviour and the non-condensable gas generation was already pointed out [19].

Table A.7. Chemical composition of typical concrete species used in LWR; Serpentine only used in Russian RBMK and VVER [19]

	Type of Concrete			
Species	**Siliceous**	**Limestone/ comm. sand**	**Limestone**	**Serpentine**
SiO_2	76.6	35.8	3.6	33.5
CaO	9.2	33.1	51.6	7.0
MgO				29.3
Al_2O_3, MgO. Fe_2O_3, K_2O,...	5.3	5.2	3.2	10.3
CO_2	2.9	21.2	35.7	1.1
H_2O_{bound}	1.8	2.0	2.0	~13.8
H_2O_{free}	4.2	2.7	3.9	~5.0
H_2O_{total}	6.0	4.7	5.9	18.8

In most cases, structural concrete used in NPPs can be grouped into one of the three categories, namely siliceous, limestone/common sand, and pure limestone concrete. The main difference in the first 3 types is the ratio of $Si0_2$ to CaO. The latter exists originally mainly in the form of limestone ($CaCO_3$). At temperatures above 800 °C limestone decomposes into CaO and gaseous CO_2. Therefore, concrete with high limestone content generally produces high degassing rates, which exceed the degassing rates of pure siliceous concrete by a factor of 2 to 3. The degassing rates have in addition considerable influence on the dynamics and heat transfer of the melt pool [19].

In 1987, the importance of the condensed phase chemistry was recognised by the SURC-4 experiment which gave the first quantitative results for the interaction of metallic zirconium with the silica from the concrete according to the exothermal reaction:

$$Zr + SiO_2 \rightarrow ZrO_2 + Si$$

This reaction leads to the formation of elemental Si, which will be oxidised later on by the gases H_2O and CO_2 from the concrete decomposition. For temperatures exceeding 2 200 K, the Zr oxidation may be more and more influenced by the reaction:

$$Zr + 2\ SiO_2 \rightarrow ZrO_2 + 2\ SiO$$

The steam and the carbon-dioxide from the concrete oxidise the metallic phase of the corium melt according to:

$$Me + H_2O \rightarrow MeO_X + H_2$$

$$Me + CO_2 \rightarrow MeO_X + CO$$

The sequence of the metals to be oxidised is Zr, Si, Cr, Fe. Another important effect of the chemical reactions in the melt is their influence on the oxygen potential in the melt which may determine the chemical form of the fission products and hence their release rates [19].

It is further known that, even if the erosion rate of the basemat is low, the pressure increase in the containment due to the gas generation from the decomposing concrete may be significant, particularly for small-volume containments, such as BWR Mark I and Mark II containments. Long-term decay heat and chemical power generation can lead to basemat melt-through in the absence of corium cooling. Moreover, most of the tellurium and some of the low-volatility fission products, such as cerium, barium strontium, ruthenium and lanthanum, could be released to the containment during the ex-vessel phase of the accident. The question of the generation of combustible gas by core oxidation and MCCI is of special importance for Fukushima Daiichi, as combustion processes caused by hydrogen released from the containment damaged the reactor buildings of three different units [19]. It should be mentioned that the results of earlier work sometimes suggested conclusions that in light of the latest experimental results can no longer be maintained; hence, care should be applied in using the results published in [19].

In the years 2000, new experiments were launched, among them CCI [20] and VULCANO [21, 22] to study the 2D ablation effects and the role of the metallic layer. More recent R&D work performed in NEA member countries on MCCI and debris coolability is related to the European SARNET (Severe Accident Research NETwork of excellence) project (FP6 and FP7 respectively of Research and Development of the European Commission, EC) [4, 6, 7].

Within SARNET, the addressed situation related to MCCI was the presence of melt/corium in an initially dry reactor pit with the possibility of water injection later during the MCCI phase. The work on concrete erosion was based on the analysis of recent 2D real material MCCI tests performed in Argonne National Laboratory, United States, within the MCCI projects, VULCANO experiments performed by CEA, France, with an oxide pool and VULCANO and MOCKA experiments performed at CEA, Cadarache and KIT, Germany, with oxide and metal melts (simulant for MOCKA). The following topics are of interest here:

- A major question was why an anisotropic ablation of basemat was observed in several experiments with siliceous concretes and an isotropic one with carbonaceous concretes [2], [3].
- Another potential factor that might strongly affect MCCI is the possible metal/oxide stratification and the spatial distribution of metallic and oxide phases. Experiments provided new insights at the large-scale MOCKA facility performed at KIT, Germany, using ~1 ton of corium consisting of simulant materials. It also has the unique ability to study the effect of steel rebars in concrete on the ablation by corium. Further experiments have been performed at the VULCANO facility (CEA,

France) and the SICOPS facility (Areva GmbH, Germany) facilities using prototypic materials [5].

- As for corium coolability, recent data suggest that early water flooding i.e. at a time when the molten concrete fraction is still low, has a good potential for the coolability. However, this remains to be confirmed by further experiments. The efficiency of late top flooding seems more doubtful, especially for siliceous concrete. Bottom flooding is being studied as an efficient alternative to the top flooding [4].
- The corium composition and properties determine the interactions both with the reactor vessel and in the later phases with the concrete basemat. Separate effect and larger scale tests supplied new data on melt-liquidus composition for the NUCLEA thermodynamic database [1].

Within SARNET, a State-of-the-Art-Report on MCCI in dry conditions summarising analysis of experiments, modelling and reactor applications was written. This report contains a description of insights from the Chernobyl accident in 1986 and data sampling related to MCCI and melt progression. This part could be taken as an example related to the decommissioning interest with regard to MCCI [2].

Within SARNET the R&D priorities have been updated by a group of experts [6], [7]. The work also accounted for the preliminary understanding of the Fukushima Daiichi accidents, which resulted in only slight reorientations of priorities because most involved phenomena were already addressed as being of highest priority. Related to the issue of MCCI the following two categories are of interest: a) phenomenon that could lead to early containment failure and b) phenomena that could lead to late containment failure, but only the second one is relevant to the Fukushima Daiichi accidents. There, phenomena related to late containment failure are important since they are linked to melt pool / debris configuration in the cavity (the pedestal) during the MCCI process, the concrete ablation, and the corium coolability issue, all of which affect the containment integrity. Encountered phenomena are heat transfer from the melt, crust formation, sparging gas and cracking of crusts. Reactor scale calculations performed within SARNET indicate that the major sources of uncertainties lie with the oxide-metal corium interactions (evolution of layer configuration). Furthermore, knowledge of the cooling mechanisms by top flooding of the ex-vessel corium pool needs to be increased, because this affects the containment integrity with a possible basemat erosion stop or not.

What so far was not yet considered or studied is the relocation of material from a BWR RPV bottom down along the multiple structures (see Figure A.1 in chapter A.1.1; control rod drive systems, support tubes, beams, grids or plates, instrumentation pipes) and its potential retention or freezing there. In case of BWR in addition other equipment needed for maintenance is placed in the reactor cavity allowing melt to accumulate and freeze there. In conclusion, there is a high probability that only a portion of the melt released from the RPV bottom may reach the cavity bottom with the chance to start the MCCI reaction or to spread further out of the cavity. Such behaviour is typical for BWRs especially and needs further consideration.

A.1.3.2 Safety Research Interest

The safety research interests discussed below consider open topics that are already known, as well as results of the ongoing BSAF project [8], [9]. A large number of experimental studies on ex-vessel phenomena (especially on melt spreading, FCI and MCCI) accompanied by model development have been performed during the past decades, typically by using simulant materials or prototypic materials, but with technical limitations related to the realistic simulation of decay heat input to the melt. Looking from a scientific/technical perspective, the Fukushima Daiichi accident may provide a variety of valuable data on the ex-vessel processes and associated phenomena especially with view on BWR specific aspects and may help to solve open questions especially related to the melt relocation from the RPV, melt flow along structures below RPV vessel as typically for BWR, and the distribution in the lower part of the containment, melt spreading phenomena, the ability of a debris bed to get or to remain cooled, the concrete ablation process, the erosion front behaviour, the impact of water on the coolability of the melt, the termination of the erosion process, and the final melt composition at termination.

In the case of theFukushima Daiichi accidents, MCCI very likely occurred in Unit 1, and probably as well in Unit 3 for some time, but did not lead to a large containment failure with significant melt release to the reactor building. Most probably, MCCI stopped after some hours of concrete erosion due to its coolability by the water or air injected into the containments. The damage found in Unit 1 leading to water leakage from the containment may be caused by MCCI, which however has not yet been verified. The extent of melt relocation into the pedestal and of the erosion process as well as the causes for its termination are not yet known; water being available in the containment especially after melt relocation from the RPV may have contributed by cooling the melt. As mentioned above, the amount of melt which was finally released to the pedestal floor is still unknown; and this would be the same question if melt was flowing out of the pedestal opening to the surrounding lowest containment floor. The latest results of plant inspections in Units 1 and 3 provided interesting results as obviously no large damage inside the lower containment is visible [9]. These findings put in question analysis results that predict a long duration of MCCI.

Regarding safety research interest of ex-vessel examinations, we can therefore conclude that prototypic full-scale data obtained from Fukushima Daiichi Units 1, 2, and 3 offer the unique opportunity to resolve many open issues with information related to melt relocation from BWR vessels, melt spreading, MCCI and melt cooling in BWRs. Available information suggests that post-accident examinations could provide significant insights into key late phase ex-vessel phenomena**,** which should be considered as **high priority examinations from a safety research point a view.**

A.1.3.3 Decommissioning Interest

A.1.3.3.1 NDF's strategic plan

NDF has established a Strategic Plan for the decommissioning of Fukushima Daiichi NPPs. Details has already been discussed in chapter A.1.1 with regard to fuel debris retrieval from the RPV. Similar issues are valid here. Table A.7 presents the information related to ex-vessel.

For decommissioning, the main interest is listed in the table above. Beyond that the question of potential local recriticality in the melt/debris during the decommissioning process should be considered. Considering the key challenge of the fuel debris retrieval from Fukushima Daiichi units 1-3, NDF considers that **the priority of ex-vessel issues in terms of decommissioning interest is high**, as long as it updates and validates the estimation of the final location and composition of fuel debris in the ex- vessels and assessment of safety and risk of fuel debris remaining in the PCV including potential recriticality issues.

Table A.8. Overview on key issues and related information to be gained from the decommissioning operation for the ex-vessel phase

Key issues	Information to be analysed/obtained
1. Fuel debris location and distribution	
1. Fuel debris relocated to the PCV (MCCI) and remained inside the pedestal	1. Mass, form, size, morphology, and composition of fuel debris (MCCI)
2. Fuel debris (MCCI) relocated to the periphery and outside of pedestal	1. Extent of relocation of fuel debris (MCCI) at the PCV bottom
2. Structural integrity of the PCV and R/B	
1. Degradation evaluation for pedestal exposed to high temp. environment	1. Extent of relocation of fuel debris (MCCI) at the PCV bottom
	2. Temperature history of the PCV
1. Feasibility and adoptability of fuel debris retrieval method	
1. Feasibility of cutting and removing methods of fuel debris	1. Fuel debris properties at each location (hardness, toughness etc.) - Core and lower RPV - PCV bottom - S/C, etc.
	2. Long-term stability during fuel debris retrieval operation (chemical activation by laser, property degradation etc.)
2. Design of storage canister of fuel debris	1. Long-term stability of fuel debris during storage
	2. Fuel debris properties (possible dehydration and drying)
	3. Possible chemical reaction of fuel debris (gas generation etc.)

A.1.3.3.2 BSAF project

In order to support the decommissioning operations, the CSNI decided to conduct a benchmark study of the accident progression for the Fukushima Daiichi NPS units 1-3 accident with some of NEA member countries having a long experience in developing and using severe accident computer codes and methods of analysis, constituting an international project named Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) on the basis of the proposal of the METI, as already mentioned in section A.1.1.3.2.

With regard to MCCI, it should be mentioned, that:

- for Fukushima Daiichi Unit 1, the RPV has failed in all the simulations; melt was released; and MCCI is predicted to be ongoing at the end of the simulations, 6 days after scram, which is questionable;
- for Fukushima Daiichi Unit 2, the majority of simulations do not predict RPV failure, and therefore no MCCI;

- for Fukushima Daiichi Unit 3, two scenarios are predicted, either with RPV failure, melt release and MCCI or without melt release from the RPV. In the first scenario, the flammable gas generation is considerably larger compared to the second case. It seems to be more plausible as the hydrogen generated in unit 3 also led to the combustion in the reactor building of Unit 4, what requires a larger amount of hydrogen to be generated.
- In general, large uncertainties remain with regard to MCCI, melt relocation down from RPV, and melt spreading and cooling.

On the basis of the analyses and findings, the report [8] proposes a preliminary list of data needs from decommissioning operations, ranked based on their importance and feasibility given the methods of decommissioning; it is already provided in chapter A.1.1.3.

A.1.3.4 Potential Examinations

As mentioned in chapter A.1.1.4 with regard to potential examinations for in-vessel phenomena, the experience gained at TMI-2 shows that data acquisition in the reactor vessel will be essential both for properly designing fuel removal operations and for scientific purposes. In Fukushima Daiichi fuel debris retrieval is as well linked to the part released from the RPV into the drywell of the containment. So experiences gained from plant examination in Chernobyl especially related to melt relocation and spreading and from material examinations linked to the various MCCI tests could be helpful, which should be collected and considered while planning the decommissioning.

First of all information needed is similar to some extend to those for the in-vessel case as presented in chapter A.1.1.4. From the decommissioning process information is expected to relate to the following aspects as summarised below:

- End state (mass, density, composition[3], distribution, morphology of various phases: crusts, debris, re-solidified molten phases) of relocated (along the structures below the RPV) core materials and of the core material collected inside and outside the pedestal. Characterisation of the distribution of melt composition throughout the debris by sampling means the determination of composition data at several locations, e. g. at the corium/concrete interface, at the corium/water interface and a variation of measurement location in vertical and lateral direction within the bulk of the corium and the crust) to see whether some stratification/segregation phenomena (e.g. refractory/non-refractory material or metal/oxide) occurred.
- Evidence of degradation of structures or interactions between fuel and structures below the RPV and between fuel and concrete in the pedestal.
- Physical characteristics affecting debris coolability (particles shape and porosity, cracks, gaps in crusts, research for the presence of boride phases, mechanical properties of debris);
- Evidence of effects due to sea water injection.
- Finally, to collect any information related to the erosion process/progress in axial and radial direction at the different locations of the containment floor/pedestal; this information can be provided through systematic bores (43 bores had been drilled in the concrete walls of Chernobyl fuel debris-containing rooms [38]) and/or successive removal of the melt/debris from the containment floor.

3. If possible weight fractions and oxidation states of main components in oxide and metallic phases, indications of FP distribution and association in oxide and metallic phases.

- Further information for the ex-vessel case which can typically be taken by visual inspection is summarised:
- The contours of melt distribution, which may be either frozen at components below the RPV or accumulated respectively onto the pedestal of the lower containment floor; repartition between particulate debris and bulk solidified melt; visual inspections would be the first step to provide related information (as an initial step already started in Unit 1 in April 2015),
- Determination of melt spreading / relocation into different areas of the pedestal (inner and outer part) including determination of a possible containment liner attack; visual inspections would be a first step to provide related information (as an initial step already started in Unit 1 in April 2015),
- Confirmation of melt relocation into condensation tubes connecting the drywell and the wetwell would be principally possible, but can be easily ruled out by visual inspection,
- Characterisation of the morphology of the melt/debris on the containment floor by visual inspection and sampling, e.g. discrimination between a fraction of the corium which is obviously fragmented and a fraction of melt that can be described as more or less continuous pool or cake and the geometrical configuration of both fractions (e.g. fragmented fraction on top of the frozen pool), presence of "eruptive cone", evidence on possible anchoring of crusts on the pedestal or sump walls; in a second stage, sieving of particulate debris would provide useful insights on fragmentation processes.
- Determination of potential deformations that could be attributed to a mild steam explosion (a violent explosion would have already been detected), by visual inspections and also sieving of particulate debris (fine debris – median size of a few tens of millimetres – are an indicator of steam explosion).

This number of issues above is grouped towards a graded approach, starting with examinations that are less challenging and more easily feasible. Melt samples should be collected from different places within the containment, allowing to determine the different processes and the composition of the melt/debris at various locations.

A.1.3.5 Ongoing R&D Activities

A.1.3.5.1 In Europe

Currently, an effort to collect all existing information related to MCCI is underway in an NEA SOAR report. This report, which is under final preparation, describes challenges to the containment integrity by MCCI with a description of the findings from experiments, and current computer tools and methods. Finally, knowledge gaps and needs for further research will be addressed. No details are ready for presentation yet.

Two French national research projects deal with Ex-vessel corium issues: MIT3BAR deals with the coolability of oxide-metal corium mixtures by top or bottom flooding, including experiments with simulant materials at MOCKA (Karlsruhe, Germany) and prototypic corium in MERELAVA facility at CEA Cadarache (France); ICE project on ex-vessel Fuel Coolant Interaction, including experiments in KROTOS, VITI and ATTIHLA facilities at CEA, France.

R&D activities that could complement the proposed inspections and sampling processes during decommissioning are related to two issues, the analyses performed within the NEA BSAF project [8] and complementarily the provision of relevant knowledge and material data gained e.g. through the experimental work performed in the past decades in the both NEA member countries and non- NEA member countries and the knowledge gained by the SARNET project on the relevant issues. Thus separate supporting project should be established.

Within the European SAFEST project, a round-robin exercise on the analysis of an oxide-metal MCCI sample from SICOPS facility (Germany) by laboratories in CEA Marcoule and Cadarache (France), UJV (Czech Republic) and ITU (European Commission). It might be enlarged to the OECD level for a further stage (to be negotiated).

Relevant MCCI related data with reactor typical materials including radionuclides etc. can only be gained by the decommissioning of the Fukushima Daiichi units; within all experiments performed so far the technical limitations and artefacts (decay power simulation, size, and duration) were one of the biggest deficits so far.

A.1.3.5.2 In the US

A systematic investigation of the costs and benefits of TMI-2 inspection information is not available. It is clear that visual inspection information from within the vessel and the containment offered important insights at a lower cost than insights gained from post-accident examinations of radioactive samples removed from TMI-2. Nevertheless, important insights were gained from examinations of vessel steel related to peak temperatures, and examinations nozzles removed from the vessel provided important insights related to the potential for melt to relocate through these lower head penetrations. Likewise, a systematic investigation of 'lessons learnt' from TMI-2 examinations is not available. Such a cost/benefit evaluation could provide important insights related to the desired number and type of sample measurements, advanced testing of sample extraction techniques, and the benefit of separate effects testing. In addition, such evaluations might identify information not obtained from TMI-2 that would be useful to obtain from Fukushima Daiichi. Although these systematic evaluations were not available to the expert panel participating in the US forensics efforts, knowledge of the TMI-2 experience was applied in identifying information needs. For example, information needs focused on visual information that could provide important insights at a lower cost [24]. In addition, the US DOE is currently planning to hold a workshop to transfer knowledge learned from TMI-2 examinations as part of an effort to assist Japan in their decontamination and decommissioning efforts.

A.1.3.5.3 In Japan

In order to investigate generation behaviour of molten corium and concreate interaction and studies to characterise mechanical and chemical properties, IRID/JAEA has been conducting fundamental studies such as the following: Estimation by thermodynamic calculations has been done by focusing phase change behaviour of MCCI products. It suggested that, typical components of the bulk of MCCI products in Fukushima Daiichi NPS were $(U,Zr)SiO_4$, UO_2 and $CaAl_2Si_2O_8$ in homogeneous case. And also, research of phase relationships in simulated MCCI product was made by small scale examination of arc melting (homogeneous melt) or light-concentrating heating (temperature gradient) of concrete/SUS/Zr/$(U,Zr)O_2$. Besides that, in collaboration with CEA, evaluation on the large scale simulated MCCI products made by previous VULCANO test has been done. A few campaigns were selected from previously formed MCCI test products held by CEA, based on the criteria that conditions of the campaign should closely reflect concrete components and the ratio of fuel and structural materials found after the accident at Fukushima Daiichi NPS. Several samples from the test products were taken and subjected to chemical composition and hardness tests. As a result, data that could be used for predicting the general characteristics of MCCI products were obtained. [25, 26]

NRA and JAEA have initiated studies to develop methodologies for evaluating ex-vessel debris coolability in a wet cavity strategy as a severe accident countermeasure, in which molten debris is ejected into a water pool formed in reactor cavity or pedestal area when the lower head of reactor vessel is failed. This project includes experimental research in co-operation with the Royal Institute of Technology in Sweden and the

Institute of Nuclear Technology and Energy Systems in Germany. The experimental database will be extended for debris jet break-up in the water pool, melt spreading on the wetted floor and debris bed coolability including re-melting, and applied into the improvement and validation of simulation codes such as JASMINE code developed at JAEA for molten fuel/coolant interactions. Experimental facilities to be included in this study are DEFOR-A, PULiMS and REMCOD (new one). Numerical models consist of multi-dimensional debris morphologies, interaction of molten core and solidified debris and two-phase flow heat transfer in the porous media not only in the short term but also in the long term. The re-melting phenomena focus on interactions between molten metal and solid oxide debris bed. As a safety evaluation method, not only deterministic approaches but also statistical approaches will be developed.

JAEA analysed chemical compositions of the concrete samples which were drilled from the floors and walls of the reactor buildings of Unit 1 and 2 for analysis of surface contamination. The analyses with ICP-AES and precipitation method showed the following concrete composition.

Table A.9. Composition of concrete for Fukushima Daiichi Units 1 and 2

Fukushima Daiichi Unit 1		Fukushima Daiichi Unit 2	
Element	**Mass fraction [%]**	**Element**	**Mass fraction [%]**
Al	7.0	Al	6.5
Ca	7.8	Ca	9.1
Fe	3.6	Fe	3.3
Si	25.0	Si	27.0

The results indicates that approximately 85% of the concrete consist of oxides of the four main elements and the remaining 15% consists of minor elements such as K, Mg, Ti, Mn and water [27].

Also, the status of corium falling from melted tube such as temperature and particle phase formation can be obtained and will be used to investigate the melt spreading phenomena on the PCV floor. Preliminary test for corium melt generation had been performed. About 5 kg of materials with the composition of UO_2: Zr: ZrO_2: Fe: B_4C=60 :10 :15 :14 :1(wt%) was heated by induction coil and it was confirmed that materials with above composition could be melted by induction heating. Tests in which a real size test section with in-core monitors housing or control rod housing will be performed using real corium in this fiscal year. In the test, materials with above composition up to 100 kg will be heated by induction coil in the melt crucible. The composition and amount of melt can vary. Melted materials will fall to the test section of the penetrating tube through melt delivery channel. Heat by induction coil will be added to simulate decay heat at the test section. Temperatures at various locations of the test section will be measured to grasp the heat conduction phenomena by real corium. Also video from different positions will be taken to visualise broken position and status of corium falling from the broken part. After the test, composition of the debris at various positions will be measured by ICP-AES, etc.

To gather all the information related to MCCI from the decommissioning process, the involved Japanese experts and engineers attach great importance to a careful preparation of the task and ask for support by the international experts' community. Combining decommissioning and safety research interests should be done to a large extent possible to allow maximum benefit to solve open questions and to improve models to be applied for other plant studies thereafter. It could be beneficiary to hold a workshop – as a first step – which should bring experts together from various countries and Japanese organisations to exchange knowledge gained with regard to MCCI and data acquisition and to support the preparation of the related decommissioning activities. Further, it allows defining and discussing main important areas of interest related to MCCI data to be gained through the Fukushima Daiichi decommissioning process. As a follow-up activity of the workshop a closer co-operation in preparation of the decommissioning activities at the Fukushima Daiichi site could result. A related research proposal is provided in chapter 4.1.

A.1.3.6 References

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A.1.4 *Containment Failure and Venting*

A.1.4.1 Background

The containment serves as the ultimate barrier against the release of radioactive material into the environment in case of a severe accident. Because of this role, compromising the containment could increase the risk of a large release in the unlikely event of an accident. According to the NPPs operating experience, various forms of degradation have been observed in the containment vessels e.g. in the United States. The examples include corrosion of the steel shell or liner, corrosion of reinforcing bars and pre-stressing tendons, loss of pre-stressing, and corrosion of bellows [1].

The severe accidents in Fukushima Daiichi Units 1-3 underlined the importance of the containment especially under beyond design conditions with significant releases of fission products and combustible gases from the reactor circuit into the containment. In all three units the pressure significantly exceeded the design pressure of the containment during the core degradation period and leakages (gases and fission products) from the containments occurred, before containment venting processes could be initiated in Units 1 and 3 to reduce the pressure. In Unit 2 such containment venting process was not successful, so that an uncontrolled depressurisation through leakages took place. Other challenges to the containment integrity are be caused by high temperatures and radiation loads.

Besides the significant fission product releases from all three units, hydrogen explosions were observed in the reactor buildings of Fukushima Daiichi Units 1, 3 and 4, and prevented in Unit 2. It is assumed that the hydrogen causing the explosion in Unit 4 originated from Unit 3. In the Units 1 and 3 the source of the hydrogen into the reactor building is most probably from containment leakages; while leakages from pipelines used for the containment venting have been discussed, too. The exact location of the main containment gas leak path as well as that of water leakages in the long term is still under investigation. Containment sections in which leakages are likely to occur are e.g.: drywell head flange, diverse hatches (equipment hatch, safety relief valve (SRV) hatch, control rod drive (CRD) hatch, suppression chamber (S/C) hatch), personnel airlock as well as various containment penetrations, and piping bellows. The behaviour of penetrations and used sealing materials under accident conditions is discussed in the chapter A.2.2.1 related to SSC performance and system survivability.

Integral large scale experiments related to containment failure are difficult to be performed, as various influences of severe accident phenomena cannot all be treated. Material data at high temperatures are sometimes unknown as well. For this some smaller scale experiments can probably be performed. Experiments regarding failures of primary containment vessels (PCVs) are reviewed recently [2]. In their review, a link to literature summarising experiments performed since the late 1970es can be found. It is mentioned that investigations have been made on primary containment vessel (PCV) failure modes and magnitudes of leakage under severe accident conditions, adopting full-scale models, scale models, and actual compounds used for seals and gaskets of different PCV boundaries. Clear indications which of the experiments are directly linked to the containment design of the Fukushima Daiichi units were not made. The history of US containment integrity tests can be seen at the SNL web page [6] and the results are summarised in [4]. An early study with regard to Japanese Mark I containments is published in [3]. In other countries containment integrity studies, typically as a part of a probabilistic risk assessment have been performed as well, which are not reflected here.

The latest information related to BWR with Mark I containment is provided by NRC as part of the State-of-the-Art Reactor Consequence Analyses [5] project. SOARCA analysed the potential consequences of severe accidents at the Surry Power Station near Surry, Va. (PWR) and the Peach Bottom Atomic Power Station near Delta, Pa (BWR Mark I). The project, which began in 2007, combined up-to-date information about the plants' layout and operations with local population data and emergency preparedness plans. This information was then analysed using state-of-the-art computer codes that incorporate decades of research into severe reactor accidents. Earlier US research and analyses results are described e.g. in different reports [4] and [1].

The SOARCA study for a BWR equipped with a Mark I type of containment [5] describes a potential failure of the drywell head flange. In [1] the leakage at the containment head flange is expected to occur first already at an average pressure of 1 MPa or 145.4 psig. Taking uncertainties into account leakages may start at lower values. This indicates that in the case of the Fukushima Daiichi accidents, leakages at the drywell head flange should be considered in any case.

Further, the EPRI Crosswalk report comparing MAAP and MELCOR results illustrated that this result was related to in-vessel debris coolability assumptions [1]. Preliminary indicators from Fukushima Daiichi suggest the MELCOR modelling is more representative, but much uncertainty remains and modelling is based in part on expert opinion [12].

To prevent containment over-pressure failure and to limit long term pressure increase in case of an accident, severe accident filtered containment venting systems (FCVS) are often installed in operating NPPs, back-fitted as part of accident mitigation. The containments in the Fukushima Daiichi power plant had a possibility for depressurisation either from the drywell or the gas space of the wetwell. The installed systems are called hardened vent systems (similar to US plants) and do not contain specific aerosol or iodine filters. If the venting line connected to the wetwell gas space (suppression pool) is used, this leads, in general, to a larger retention of fission products in the water by pool scrubbing. If the venting occurs from the drywell or in case of a significant leakage from the drywell to the adjacent buildings, the retention of the fission products in the primary containment is in general smaller, as pool scrubbing phenomena in the wetwell are not possible. In the Fukushima Daiichi Units 1 and 3, only the venting from the wetwell gas space was actuated: while in Unit 2, the foreseen venting failed and the depressurisation may have happened through extended containment leakage.

For many years, research activities have been related to containment venting as well. The CSNI released two related reports in May 1988 [7, 8], following a specialised workshop on this issue. Since then, knowledge and calculation tools to assess radioactive releases due to a severe accident as well as filtration technologies have progressed significantly. Further, some countries are now considering that (filtered) CVS can play a valuable role in managing early stages of an accident with fast containment pressurisation as well even though they were generally designed for managing slow containment pressure build-up. These evolutions in knowledge, technology and strategy led the Working Group on Analysis and Management of Accidents (WGAMA) of the CSNI to launch an action in 2012 to establish an updated Status Report on FCVSs. This action was completed in 2014 and the new SOAR report was published in July 2014 [9]. This new CSNI Status Report provides a comprehensive description of safety requirements associated with FCVS and of the status of FCVS implementation as provided by the various contributing countries. Currently assessed benefits and negative aspects related to FCVS use and the related potential improvements or countermeasures for existing systems are described. The information provided by the SOAR may also be used to guide the design, implementation and operation of future FCVS to reduce these risks. New FCVS may perhaps be designed to deal with more challenging conditions (management of early phases of an accident, cycling or long-term use in severe accident conditions). The robustness (including a design withstanding several external events), the safe use and the reliability of FCVSs for such conditions should be further assessed either to improve existing systems or to propose upgraded design requirements for future systems.

A.1.4.2 Safety Research Interest

The safety research interest for this area is high. Results from decommissioning can help to understand what phenomena caused the containment leakages and should answer the question where the main containment leakages are located and why the reactor buildings have been damaged by hydrogen explosions. These are issues which will become of importance in NEA BSAF project phase 2 (see chapter 2.3), as especially the radioactive releases from the containment through reactor building into the environment will be further examined. The information provided through decommissioning can support further code model improvement and definition of plant improvements as part of the accident management.

Most probably, the reason for the hydrogen explosions in reactor buildings of the Units 1 and 3 is gas leakage from the primary containment to the reactor building carrying hydrogen and/or carbon monoxide. The reason for the explosion in reactor building of Unit 4 is different. It was caused by the venting actions of Unit 3, and carryover of some gases backwards into Unit 4 through connected ventilation/venting pipes (e.g. a common stack is used). It is possible, but less probable, that venting could have contributed to the hydrogen explosions in Unit 3 as well by leakage of hydrogen from the vent line. In Unit 2 the containment pressure was very high and the venting was not successful, so hydrogen was presumably released into the reactor building due to containment leakages which caused the pressure to decrease. Nevertheless, an explosion in the reactor building was prevented most probably due to an open blow out panel in the reactor building in combination with steam build-up by evaporating water in the torus room. Such a scenario was discussed within the NEA BSAF project Phase 1 and will be further studied in the future.

For simulation of the accident progression and for assessment of the source term and the potential release of the activity to the environment, it is important to know the effectivity of the venting actions with regard to the retention of fission products in the wetwell water pool by pool scrubbing (see separate issue). It is generally accepted that venting of the containment through the suppression pool led to a considerable decrease in the release of the activity during the accidents in Fukushima. One of the major open questions concerning hydrogen and fission product release is the question to what extent the venting from the wetwell contributed to the releases compared to releases through containment leakages.

Available information suggests that post-accident examinations could provide significant insights, which should be considered as **high priority examinations from a safety research point a view.**

A.1.4.3 Decommissioning Interest

The **decommissioning interest is high** as there is an ongoing reliance on containment structures to prevent unmanaged release of radioactive materials during decommissioning, as there is particularly importance to establish the boundary conditions for fuel debris retrieval operations. The interest is linked to radiological aspects (protection of people and equipment during decommissioning) and the question of the integrity of the containment in general.

The following presents the information related to containment failure.

Table A.10. Overview on information in relation to containment failure, to be gained from the decommissioning operation

Key issues	Information to be analysed/obtained
Establish containment function	
1. Estimation of amount of FP remaining inside the PCV	1. Amount of FP generated during accident progression and leaked out of the PCV (gas phase and liquid phase)
2. Confinement of dust (cutting chip) during fuel debris retrieval operation	1. Fuel debris properties at each location (hardness, toughness etc.) - Core and lower RPV, PCV bottom, S/C, etc.
	2. Dust generation mechanism (amount, size, dose rate etc.) when cutting fuel debris
3. Capturing when immobilised FP adhered to the PCV and reactor internal is refloated to gas-phase	1. Condition where immobilised FP is refloated (temperature, pressure, structures surface condition etc.)
	2. Amount of possible FP nuclide to be refloated
	3. Possibility of capture/removal by ventilation filter
4. Capturing when immobilised FP is leached to submerged water	1. Condition where immobilised FP is leached (temperature, pH etc.)
	2. Nuclide and amount of FP leached again
	3. Possibility of capture/removal by water recirculating filter
5. Possibility and extent of additional release of FP from fuel debris (leaching from fuel debris etc.)	1. Condition of additional release of FP from fuel debris (leaching characteristics, temperature, pressure etc.)
	2. Amount of FP additionally released
6. Possibility ensuring containment boundary	1. Identification of leak and release pass of FPs by visual inspection as well as plant data analysis

A.1.4.4 Potential Examinations

The determination of main fission product transport paths through the reactor coolant system into the containment, as well as of the potential release from the containment into the adjacent buildings and any deposition there, is of major importance. The release path of fission products from the containment may be through a water pool (suppression chamber) or directly from the drywell either through leakages or the venting containing. The composition of the atmosphere released and the fission products released along with the gas may vary. The two different release paths (water and gas path) result in different amounts of fission products being deposited along the path, and in different concentrations of activity being released to the reactor building and/or to the environment. More detailed analyses are planed within the BSAF project phase 2.

The information available related to the various containment openings, hatches and penetrations should be collected and categorised first. This is an issue which will become of importance in BSAF project phase 2, as the radioactive releases into the environment will be further analysed by using computer codes.

Thereafter, the visual inspection of penetrations and seals, especially detailed inspection of containment head flange after its opening including a measurement of any plastic deformation of bolts used. This should be done to try to answer the question of the main containment leakage. Also the detection of other leakage paths from containment (water and gas leakages) is important both to support the decommissioning and to enhance safety research.

The difficulty in general will be to distinguish the possible damages of openings, hatches, seals and penetrations caused by the severe accident from those caused by the long term environmental impact thereafter before decommissioning.

A.1.4.5 Ongoing R&D Activities

Further containment design data are needed which would allow comparing the analyses results of US studies and others with the situation in Fukushima Daiichi. It would also be helpful to know if actual containment failure analyses exist for theFukushima Daiichi Unit 1-3 NPPs and what are the differences between the units / in what way the units are different.

R&D activities that could complement the proposed inspections and sampling processes during decommissioning are related to the analyses performed within the NEA BSAF project, phase 2.

JAEA [11] continues research activities for the improvement of an integral severe accident analysis code, THALES2/KICHE, mainly for source term evaluation. These activities are related to re-volatilisation of iodine species from a water pool of suppression chamber in the operation of containment venting and the influence of seawater injection on the formation of volatile iodine species in an aqueous phase. In addition, JAEA has initiated, under a support of NRA, a large-scale experiment on thermal-hydraulics in a containment vessel including gas phase temperature distribution associated with over-temperature damage of containment vessel structures and hydrogen mixing.

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A.1.5 *FP Behaviour and Source Terms*

Fission product (FP) behaviour and source term research covers a large range of release mechanisms over the entire reactor transient duration, and also the time after cold shutdown. It also covers release from various pathways and structures (e.g. spent fuel pools). The transient release processes encompass release from fuel failure, fuel oxidation during core degradation, release during melting, and transport within the reactor vessel, containment vessel (drywell and wetwell) via filtered venting, if any, to the atmosphere or suppression pool. It also encompasses transport to the environment through containment breaches.

A.1.5.1 Background

A thorough review of experiments examining fission product release behaviour under accident conditions was published in 2008 [1]. The review concluded that fission product release behaviour observed in both in-pile and out-of-pile tests were consistent with those determined by examination of the TMI-2 core examination with respect to the behaviour of the volatile (Xe, Kr, I, Cs, Te and Sb), semi-volatile, (Mo, Rh, Ba), low volatile (Ru, Ce, Np, Sr and Eu) and non-volatile (Zr, Nb, La and Nd) fission products except for the release of barium, where a reduced volatility was observed in the in-reactor experiments compared to the annealing tests due to thermochemical effects as a result of the presence of iron and zirconium oxides. The review also concluded that prevailing local atmospheric conditions (i.e. oxygen potential) particularly influence the release characteristics of the semi-volatile and low-volatile fission products. Comparison of integral versus separate effects experiments indicate that the non-coherent nature of melt progression tends to mask individual release mechanisms as identified in the out-of-pile experiments. For example, significant enhancement of release due to fuel liquefaction is observed in integral tests, but not in the separate-effects experiments.

For the most part, studies of the airborne releases of xenon, iodine and caesium isotopes during the Fukushima Daiichi reactor accident(s) illustrated that most of the airborne dose was caused by Cs, I and Te species, as expected based on observations from experimental programmes and understanding of the TMI-2 accident. In general, the observation of additional radionuclides, in air samples and in deposition on soil is also rather well explained. Schwantes et al. [2], note that measurements of radionuclides within environmental samples indicated that the majority of the radioactivity found in the surface soil at the site originated from venting of Units 1 and 3, and found that a trend was observed between the fraction of the total core inventory released for a number of fission product isotopes and their corresponding $\Delta G°_f$ for the primary oxide form of the element. They state that this suggests that releases were driven primarily by chemical volatility at a given temperature and reduction potential within the containment. Schwantes et al. [2] also report that seven out of the twelve most radioactive isotopes predicted to be in the environment after 300 days had thus far eluded measurement and reporting, including ^{85}Kr, ^{103}Ru, ^{91}Y, ^{127m}Te, ^{125}Sb, ^{151}Sm, and ^{129}Te. However, they also note that these seven isotopes represent less than 20% of the total predicted activity released from Fukushima Daiichi, Hence, they conclude that the majority of the isotopes with greatest environmental and human-health impact have been identified [1]. Another interesting result of this work is that ^{90}Sr was not predicted to be of major concern to the local environment. This was confirmed by studies performed by Steinhauser et al. [3], who found an intrinsic coexistence of ^{137}Cs and ^{90}Sr in the contaminations caused by the Fukushima Daiichi nuclear accident. Although the ratio of ^{90}Sr to ^{137}Cs varied significantly depending upon location, their extensive measurements confirmed that the approach used in food monitoring campaigns, which assume that the activity concentrations of β-emitting ^{90}Sr (which is relatively laborious to determine) is not higher than 10% of the level of γ-emitting ^{137}Cs (which can be measured quickly) in terms of deposition on soil and vegetation.

Airborne isotopes were observed on particulate filters at a distance of 210 km from Fukushima Daiichi site and analysed by the comprehensive nuclear test ban treaty organisation (CTBTO) laboratories, as reported in Le Petit et al. [4]. Among these isotopes are ^{103}Ru, ^{27m}Te, ^{125}Sb, and ^{129}Te, none of which were detected in soil samples. Note that publication of this data occurred after Schwantes, et. al. [2]. ^{133}Xe releases were estimated to be about 80 % of the reactor core inventories. This result strongly suggests that there was broad damage of reactor cores.

From analysis of the airborne isotopes, Le Petit et. al. [4] have drawn a number of conclusions.

- Volatile FPs, (^{134}Cs, ^{136}Cs, ^{86}Rb, ^{127m}Te, ^{129}Te, ^{129m}Te, and ^{132}Te) exhibited the same behaviour as ^{137}Cs and were released with ratios close to 100 % when normalised to ^{137}Cs [4], which is consistent with the severe nuclear power plant experiments conducted by CEA, IRSN and AECL programmes [1], [4].
- ^{25}Sb and ^{110m}Ag, considered as volatile FPs, seem to behave as semi-volatile FP due to their well-known strong retention within the reactor core structures. Their release fractions normalised to that of ^{137}Cs are, respectively, about 7 and 11 %; this is consistent with the value calculated for ^{99}Mo (13 %) as determined in VERCORS and Phébus FP experiments [1].

Using data from Schwantes et al. [2] and performing additional thermodynamic estimates, Abrecht and Schwantes [5] reached some conclusions regarding the likely state of the core in Units 1 and 3. They estimate that the upper bound temperature of the reactor cores during radionuclide release was within the range of 2015 to 2 060°C. They also note that, based on both inventory and volatility, ^{90}Sr, ^{144}Ce and ^{147}Pm are expected to account for a large portion of the medium-lived isotope radioactive background from the accident, and recommend doing more measurements of the soil in and around the accident site for these isotopes.

A.1.5.2 Safety Research Interest

Although it appears that the airborne releases of many of the fission products during Fukushima Daiichi are generally well understood, there are still some knowledge gaps. The interest is high as new findings supports public dose assessment, and may contribute in reducing uncertainty in predicting dose significant FPs. Some of the issues are highlighted below.

A.1.5.2.1 Ruthenium release

On the basis of Phébus FP and hot-cell FP release experiments, releases of ^{103}Ru and ^{106}Ru from the fuel and the core region of Fukushima Daiichi were expected to be quite small unless (or until) nearly all of the Zr in the core was oxidised. Indeed, ^{103}Ru was only observed in one air sample with a release fraction at approximately 0.016% of that of ^{137}Cs. There is a discrepancy, however, in observations from air samples and soil samples. Reports of the ^{106}Ru content in soil samples indicate a release fraction of approximately 5% that of ^{137}Cs, However, ^{103}Ru was not detected in these same samples, and given that ^{103}Ru should have had a more intense gamma signature, and be present in higher concentrations, based on its inventory, it is likely that observations of ^{106}Ru in soil may have been due to interference from ^{132}I Confirming that the fraction of Ru released from the core was indeed low, would be very useful in understanding its release and transport.

4. Note that the Te isotopes were observed at ratios approaching 200% with respect to ^{137}Cs. This is attributed to uncertainty in the core inventories of the Te species. The ratio of $^{131}I/^{137}Cs$ was observed to be 23/1. This reflects its higher volatility and is consistent with observations from Phébus and other FP release experiments.

It would be useful to look at the relative abundances of radionuclides in environmental samples (soil, air filters) from closer range than 210 km where the CBTO samples were taken, so that differences in environmental transport behaviour can be taken into account. Most of these isotopes except ^{134}Cs, ^{137}Cs, ^{125}Sb and ^{110m}Ag will have decayed below observability now; $^{106}Ru/Rh$ (expected at one third of the ^{103}Ru activity) may have become observable, but may have been deposited closer to site or within the reactor. Some Ru may be in the water in the wetwells, but this is not expected to be a major Ru location. Also, scrape samples from surfaces inside the primary containment and the reactor vessel would help to locate deposited Ru.

A.1.5.2.2 Antimony release

One of the nuclides detected in CTBO air samples was ^{125}Sb, where its fractional release was observed to be about 10% of that of ^{137}Cs. However, there was only one ^{125}Sb observation in soil samples at a close location (2011 July 18 sampling of soil from Playground sampling point, ~2% of ^{137}Cs). ^{125}Sb is difficult to discriminate from minor lines of short-lived iodine and tellurium isotopes (which have much higher inventories), and may not have been observed properly in the earlier soil samples. Decay of other radionuclides may now allow it to be detected and confirmed in archived or new soil samples. Water samples would probably help to locate it as well, as Sb is expected to be fairly water soluble. Some Sb may also have chemisorbed onto structures (e.g. reactor vessel and internal structures) at high temperatures, so scrape samples would help in determining its locations as well. Antimony also releases by leaching of irradiated fuel with water. Given the differences in Sr releases to water compared to air (see below), ^{125}Sb should also be checked for in the water samples.

A.1.5.2.3 Releases to cooling water

It has to be noted that there was continuous dissolution of radionuclides from the damaged fuel to the coolant since the advent of the Fukushima Daiichi accident. It is reported that the amount of dissolution of ^{137}Cs into water was 240 PBq after three years of the accident [6]. Comparing the amount of releases into the environment (ocean and atmospheric) of 4-90 PBq [7], this is a huge contribution The radionuclides dissolved from the fuel in the cooling water are accumulated in the primary containment vessel and turbine building in the form of contaminated water. Therefore, proper management of contaminated water is very important. However, it was reported that there could be leakage paths from the turbine building to the soil outside of the building and eventually to the sea.

Moreover, scientific knowledge on the mechanism of dissolution of radionuclides from the damaged nuclear fuel into the water is very limited. A paper published in SCIENCE in 2012 [8] points out that “accurate fundamental models for the prediction of release rates of radionuclides from fuel, especially in contact with water, after an accident remain limited and relatively little is known about fuel corrosion and radionuclide release under the extreme chemical, radiation, and thermal conditions during and subsequent to a nuclear accident. The long-lived fission products such as ^{137}Cs and ^{90}Sr and transuranium elements within damaged fuel remain a concern more than for decades”. Therefore, the release rate and species of radionuclides dissolved in the cooling water is yet still highly uncertain.

Releases by the water pathway have not been well studied. In particular, ^{89}Sr and ^{90}Sr were found in seawater samples at respective activity ratios of 10% and 1% of the ^{137}Cs activity, while soil samples indicated air-path activity ratios of about 1% and 0.1% or lower. In other words, Sr releases to water relative to ^{137}Cs were an order of magnitude higher than those to the atmosphere. Significant amounts of Sr were also found in water treated by the Cs removal system after a spill of contaminated water from the evaporative condensation apparatus at Fukushima Daiichi (2011 Dec 8). Information on the actual Sr nuclide levels in the water released from the plant, the turbine hall water, or in the torus of (for example) Unit 2 would be informative in determining whether Sr is significantly released from the fuel (possibly by leaching as well as release during the transient). The ratio of ^{140}Ba activity to ^{137}Cs activity in Unit 2 turbine hall water (20%) was also much greater than in the air-path samples (0.8%), indicating that Ba may have been leached from the fuel as well (sampling data from TEPCO). Actinides could also be dissolved in the water, and be released to the environment by this path.

It is noted that during March 2012, the concentration of ^{90}Sr rose up to 1MBq/m^3. It could be due to unintended discharge of contaminated water. The $^{90}Sr/^{137}Cs$ activity ratio, which is important for the sea food contamination, varied between about 0.005 and 500. There is no clear explanation for this behaviour yet [9]. In addition, M. Castrillejo et al. [10] analysed the monitoring data by TEPCO and confirmed that there is a recharge of ^{90}Sr into the sea, probably due to the leakage of contaminated water into the sea, and the ratio of $^{90}Sr/^{137}Cs$ tends to increases with time.

A.1.5.3 Decommissioning Interest

There is considerable overlap in research needs from a decommissioning point of view, and those from a safety research point of view. In addition, many activities that must be performed from a decommissioning perspective can directly inform safety research needs.

In decommissioning, there is a need to clarify the distribution of radioactive materials and fuel debris from the point of view of concern for radiation exposure.

Part of the determination of the status of fuel debris will involve chemical and radiochemical analysis of the fuel debris. Activities such as gamma-scanning of the fuel, and radiochemical and ICP MS analysis of fuel debris will contribute both to the safety research, addressing some of the proposed safety needs identified in the next section, while also contributing to activities such as designing transportation casks and storage facilities, identifying needs for processing and disposal of fuel debris and radioactive wastes.

Radioactive wastes generated during the Fukushima Daiichi NPP accident and the decommissioning are very different from those generated from normal decommissioning of NPPs in terms of concentration of radioactive materials, shape, constituent materials, volume, chemical composition and form, etc.

Table A.11. Overview on information in relation to FP behaviour and source term, to be gained from the decommissioning operation

Key issues	**Information to be analysed/obtained**
Establish containment function	
1. Estimation of amount of FP remaining inside the PCV	1. Amount of FP generated during accident progression and leaked out of the PCV (gas phase and liquid phase)
2. Confinement of dust (cutting chip) during fuel debris retrieval operation	1. Fuel debris properties at each location (hardness, toughness etc.) -- Core and lower RPV, PCV bottom, S/C, etc.
	2. Dust generation mechanism (amount, size, dose rate etc.) when cutting fuel debris
3. Capturing when immobilised FP adhered to the PCV and reactor internal is refloated to gas-phase	1. Condition where immobilised FP is refloated (temperature, pressure, structures surface condition etc.)
	2. Amount of possible FP nuclide to be refloated
	3. Possibility of capture/removal by ventilation filter
4. Capturing when immobilised FP is leached to submerged water	1. Condition where immobilised FP is leached (temperature, pH etc.)
	2. Nuclide and amount of FP leached again
	3. Possibility of capture/removal by water recirculating filter
5. Possibility and extent of additional release of FP from fuel debris (leaching from fuel debris etc.)	1. Condition of additional release of FP from fuel debris (leaching characteristics, temperature, pressure etc.)
	2. Amount of FP additionally released
6. Possibility ensuring containment boundary	1. Identification of leak and release pass of FPs by visual inspection as well as plant data analysis
Dose reduction and shielding	
1. Decontamination capability for FP adhered to the PCV and reactor internals	1. Chemical form of FP (likelihood of removal)
2. Shielding capability during fuel debris retrieval operation	1. Dose rate estimated by form and composition (element distribution etc.) of fuel debris at each location (including self-shielding effect by metal-containing materials) - Core and lower RPV, PCV bottom, S/C, etc.
	2. Dose rate estimated by FP at each location -Core and lower RPV, PCV bottom, S/C, etc.

For decommissioning needs, it is necessary to understand both the content and the inventory of the radioactive materials generated during and after the accident. Assessment of inventory of radioactive materials, development of methods for analyses of debris, estimation of characteristics of secondary radioactive wastes generated during water processing, and analysis of radioactive materials that are difficult to measure are all aspects that require attention. Radionuclide characterisation is crucial to understanding what could become airborne during decommissioning processes, and to evaluation of measures to reduce employee dose. All of these needs can be partially addressed with activities identified to satisfy research needs. In particular, data on aqueous wastes will be essential to developing regulation on how to process and dispose of the large accumulation of aqueous waste, as well as addressing a safety research need of fission product leaching from fuel.

Decommissioning interest in FP behaviour and source term research **is High** as long as it will be useful for planning, designing and managing how to establish containment function and dose reduction and shielding: FP behaviour is key to actual public and worker dose, and there is a need to evaluate characteristics of FP residues and associated dose. There is also a need to assess potential releases during debris retrieval.

A.1.5.4 Potential Examinations

The following examinations are proposed, ranked in order of highest feasibility (or least effort) to lowest feasibility (or greatest effort).

High Feasibility (if sufficient sample material remains)/Low Effort

- Re-measure stored samples of soil and turbine hall water for activity (may find ^{106}Ru/Rh and ^{125}Sb)
- Assemble a database of radionuclides detected in aqueous samples, including turbine hall and basement samples shortly after the accident, Fukushima Daiichi waste tanks, filtration media, and off-site releases to sea and air (TEPCO mentions that as many as 62 isotopes were detected in the Advanced Liquid Processing System (ALPS) filtration media, but did not give the isotopes or quantity).

Medium Feasibility/Medium Effort

- Sample water in Unit 1, 2 and 3 torus, and perform:
  - Gamma spectrometry, looking for ^{106}Ru/Rh, ^{125}Sb, and other potential fuel leachates such as ^{144}Ce/Pr and ^{152,154,155}Eu, and ^{134,137}Cs.
  - Beta analysis, mostly looking for ^{90}Sr
  - ICP-MS analysis, looking for FP elements and actinides

Low Feasibility/High Effort

- Gamma scanning of fuel

If suspended sections of fuel rod remain above the molten pool, gamma-scanning along their lengths after they are removed from the reactor and before destructive examination could prove very informative about the temperatures to which they were subjected. Measurement of oxide thicknesses and morphology (oxide and residual metal) on the cladding might also be useful in this regard (e.g. prior-beta and alpha-$ZrO_{0.38}$ regions in the residual metal, columnar oxide growth at lower temperatures vs. "chunky" oxide morphology at high temperatures). Fission products (e.g. Sb, Te, Ru) released from the pool may also have deposited on the clad surfaces.

Gamma spectroscopy at hot spots (through venting piping, potential sealing leaks, damaged PRV areas in Unit 1, wet well) and temperature transitions. The focus should be on long lived fission products (e.g. Ru, Sb). This data will be used to confirm deposition of species suspected to be retained in leak paths, and the location of the leak paths themselves.

A.1.5.4.1 Sampling of containment and vessel surfaces

If possible, several regions inside the containment and pressure vessel should be scrape-sampled and examined by gamma spectrometry, possibly followed by dissolution of the scrape material and solution analysis by ICP-MS or similar isotope-discriminating techniques. The ICP-MS data will need careful treatment, because we are looking for isotope distributions characteristic of fission products in a high background of natural-abundance materials. Elemental separations and beta counting to look for deposited ^{90}Sr might also be useful. Some recommended scrape sampling locations:

- Cladding of upper regions of fuel rods
- Interior of reactor pressure vessel at various elevations
- Steam separators, steam dryers or other internal structures in reactor pressure vessel
- Line from reactor to wet-well
- Interior of wet-well
- Interior of dry-well
- Line from dry-well to interior of building

It should be noted that decommissioning activities will also benefit from a careful mapping of deposited fission products, so many of these examinations may already be planned. In addition, these examinations would support research needs identified in the sub-sections on containment failure and venting, and pool scrubbing, allowing for the identification of release paths and a clearer understanding of the genesis of hydrogen accumulation, and containment failure, venting and leakage paths.

A.1.5.4.2 Corium Sampling

In addition to sampling the suspended remnants of fuel rods and the interior of the RPV and its structures, the corium should also be sampled in order to determine its overall chemical composition and to determine whether large fractions (say, >5%) of other fission products or actinides have been released from the molten pool. Deposition is a much more sensitive indicator of release, but overall inventories may be best determined from the molten pool.

The experience gained in the analysis of corium samples produced during experiments like VERCORS/VERDON or PHEBUS for the radionuclides, or COMETA (Czech Republic), SICOPS (Germany) and VULCANO (France) for the analyses of chemical species will be useful to provide valuable data for modelling and decommissioning purposes.

A round robin could be proposed on irradiated degraded fuel and prototypic ex-vessel corium to compare the best techniques developed internationally by the relevant analytical laboratories for corium.

A.1.5.5 Ongoing R&D Activities

There are several national and international programmes that are aimed at understanding fission product releases, and these can contribute to improved understanding of the radionuclide distribution post Fukushima. Details are given below.

A.1.5.5.1 FP release from fuel

Tests were performed between 2011 and 2014 in the International Source Term Program (ISTP) in the CEA VERDON facility to complete the existing database on FP release

(SASCHA, ORNL HI/VI, CRL, VEGA, HEVA/VERCORS, PHEBUS PF) for high burn-up UO_2 and mixed oxide fuel. Ruthenium release under oxidant conditions was also investigated. An additional test has been performed in 2015 studying oxidant conditions and boron effect on FP transport.

A.1.5.5.2 FP transport in the RCS

Tests were performed between 2008 and 2014 in the ISTP in the IRSN CHIP facility to develop chemical models (thermodynamic and kinetic description) of the I, Cs, Mo, B, H, O chemical system able to calculate gaseous iodine fractions at the RCS break. In 2015, tests are performed to investigate the effect of Ag-In-Cd CRD materials. VTT has been studying the surface chemistry of FP deposits in the circuit during 2007-2016. The main emphasis has been on the release of gaseous iodine from the deposits under various conditions. Several mixtures of CsI, Ag, AgI, Mo, MoO_3, Cd and B_2O_3 have been investigated.

Ruthenium transport, deposition and re-volatilisation in the RCS were investigated in between 2011 and 2015 in the NEA STEM project conducted by IRSN to assess transport of gaseous ruthenium fractions to the containment for oxidant accident scenarios. VTT has studied the transport, deposition and re-volatilisation of ruthenium in between 2003-2007. Further experiments were performed in collaboration between VTT and Chalmers in 2014-2015. The experiments have produced information on the effect of atmosphere, air radiolysis products and additional aerosols on the transport of Ru as gaseous and aerosol compounds.

In 2013, IRSN launched the French MIRE programme (funded by the French National Research Agency [ANR]) which is investigating I and Cs re-emission from RCS deposits.

A.1.5.5.3 FP behaviour in the containment

Tests were performed between 2006 and 2010 in the ISTP and between 2011 and 2015 in the NEA STEM programmes in the IRSN EPICUR facility to develop chemical models for gaseous iodine interaction with paints and organic iodides (Org-I) formation under radiation in the containment. These tests were complementary to tests performed in the NEA BIP1 and BIP2 programmes operated by CNL (formerly AECL) between 2008 and 2014. Tests investigating effects of paint ageing on gaseous iodine adsorption and Org-I formation will be performed as part of the NEA BIP3 and NEA STEM2 (2016-2019) projects.

As part of the NEA STEM programme, iodine oxides (I_xO_y) particles formation and decomposition and metallic iodides aerosols decomposition under radiation were investigated. These processes were shown to affect substantially the volatile iodine behaviour in the containment and their effect on source term will be further investigated as part of the BIP 3 and STEM2 projects. In collaboration between VTT and Chalmers, the desorption of gaseous iodine from I_xO_y and CsI particle deposits on various aged paint and metal surfaces was studied in 2011-2012. The release of radio-labelled iodine was followed when the deposits were exposed heat, humidity or gamma radiation. Iodine seemed to release even without radiation.

VTT has studied the radiolytic oxidation of gaseous inorganic and organic iodine by UV, beta and gamma radiation in 2009-2012. The formation iodine oxide particles could be measured in detail. The formed particles were sensitive to air humidity and decomposed to iodic acid and elemental iodine after being in contact with humid air.

Tests have been performed between 2006 and 2010 in the ISTP at IRSN to study gaseous ruthenium tetroxide (RuO_4) formation and stability under radiation. Models describing RuO4 behaviour in the containment have been developed.

Within the European SAFEST project, a round-robin exercise on the analysis of an oxide-metal MCCI sample from SICOPS facility (Germany) by laboratories in CEA

Marcoule and Cadarache (France), UJV (Czech Republic) and ITU (European Commission). It might be enlarged to the OECD level in a further stage (to be negotiated).

In Japan, the_NRA and JAEA continue to conduct analytical and experimental studies on chemical behaviour of FP during release, transportation and deposition mainly in BWR system, partly in collaboration with CEA. JAEA also conducts fundamental research by using relatively small devices focused on FP behaviour under steam-starvation atmosphere. An integral test that reproduce a realistic LWR-SA thermal-hydraulic condition will be performed by using the VERDON facility in CEA-Cadarache, with focused on both Ru behaviour under air ingression and boron-injection conditions [11]. The Ru behaviour is of great importance under air-ingress case, which generates the volatile Ru oxide. As many countries focused on this Ru behaviour, the integral test aims at obtaining those with high burn-up UO_2 fuel. Boron effects are especially important since they could cause a higher gas Iodine fraction that was observed in PHEBUS-FPT3 test. As mentioned previously, the integral test, VERDON-5 test, will be performed in this year under the framework of collaborative work by multiple institutes. Detailed test conditions agreed upon by each participant include a short period of re-irradiation in a test reactor to enable evaluation of the short-lived Iodine behaviour, the use of a newly developed Boron injection device, to evaluate the gas phase chemistry among FPs and Boron possible. Online gamma spectrometry capabilities have been installed for measuring gas Iodine release behaviour. A counterpart basic test using non-radioactive material will be carried out in the JAEA device under a reductive atmosphere, to compare results with the VERDON-5 test result.

JAEA is also conducting experimental and analytical studies on Cs chemisorption behaviour onto SS surfaces, in order to construct a mechanistic process model for Cs-chemisorption [12]. Although little is known for Cs-chemisorption phenomena, this phenomenon is important in views of both improved source term and Fukushima Daiichi decommissioning work. Some previous studies have reported the formation of insoluble Cs-Si-O compounds by the Cs-chemisorption onto SS surface at high temperature around 1000 K, which shed a light to the importance of retained Cs behaviour under changing conditions in Fukushima Daiichi, water immersion, change of temperature, and so on. Cs-chemisorbed samples are analysed by various experimental techniques such as XRD, SEM/EDS, X-ray photoelectron spectroscopy, and radiation synchrotron analysis. The experimental data are analysed by ab-initio calculation. The first fundamental test results using surrogate materials have shown the congruency of Cs and Si impurity in the Cs chemisorbed compounds, which agreed with the previous results. Possible micro metre order Cs intrusion into SS was also observed in some tests, which should be further investigated. Additional data acquisition is needed in order to elucidate the basic mechanisms. Also other conditions directly related to Fukushima Daiichi needs should be considered, for example chemisorption features both at lower temperature region and for other nuclides than Cs, solubility into water, and so on.

CRIEPI/ITU have been conducting leaching tests of irradiated fuel pellets and TMI-2 debris in aqueous solutions including borated water and seawater.

In order to understand the chemical and transport behaviours of Pu and U in the corroded debris which could be being cooled in several types of aqueous solution, IRID/JAEA has been performing experimental study. As the part of results, the UO_2 powder submerged in H_2O_2 aqueous solution changed to be monoclinic UO_4-$4H_2O$, whereas PuO_2 powder remained unchanged under the same condition [13].

Given that there is already existing radionuclide data from Fukushima Daiichi, and that further measurements will be required for decommissioning and remediation, the uncertainties in fission product behaviour identified above can be addressed most effectively by analysis of Fukushima Daiichi specific data. It is anticipated that the NEA BSAF Phase 2 project will provide further insight into gaps regarding our understanding of fission product releases during Fukushima Daiichi accident, and guide additional activities in this area.

Radiochemical analysis of rubble and trees collected from the site of Fukushima Daiichi NPP was conducted by JAEA. The main results are as follows [14].

- Tritium was uniformly distributed in the collected samples in low radioactivity concentrations (<2 Bq/g).
- ^{14}C and ^{60}Co were detected only in the rubble, and the concentrations of ^{14}C are not correlated with those of ^{137}Cs. In the rubble around unit 3, a weak correlation between ^{60}Co and ^{137}Cs was observed.
- ^{79}Se and ^{99}Tc were detected only in the trees in low radioactivity concentrations (<1 Bq/g).
- The radioactivity concentrations of ^{90}Sr were correlated with those of ^{137}Cs, and the $^{90}Sr/^{137}Cs$ ratio was at almost the same level as that in environmental soil samples collected after the accident. The observed results for ^{90}Sr also implied that the $^{90}Sr/^{137}Cs$ ratio was different for each unit of the NPS.
- Unfortunately, ^{36}Cl, ^{94}Nb, ^{129}I, ^{152}Eu, ^{154}Eu, and alpha-particle-emitting nuclides were below the detection limit of the conventional method.

JAEA is going to start a programme with the grants from the Japan Science and Technology Agency which aims at developing the methodology to evaluate the inventories of radionuclides in the solid wastes in the Fukushima Daiichi NPS site. Radionuclides in the soils, trees and particles collected around the Fukushima Daiichi NPP site are systematically analysed and the evaluation methodology is developed based on the dependencies of distance and directions from the site as well as the time dependencies on the radionuclides concentrations and species [15]. JAEA has been conducted analysis of soil samples collected in the environment out of the NPS site though the original locations are somewhat localised or away from the site. The soil samples collected by JAEA may be candidates of the proposed examinations indicated in 2.7.4.

A.1.5.6 References

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A.1.6 *Pool Scrubbing*

A.1.6.1 Background

Some BWR and PWR severe accident scenarios involve transport of fission products (aerosols and gas phase species) through pools of water where the fission products can be retained. This phenomenon, known as pool scrubbing, has the potential to reduce the source term significantly. Nuclear aerosol investigations and with it the investigation of pool scrubbing of aerosols began in the late 1960s and early 1970s. The progress made both in experimental studies and in code development is attested by the publication of three NEA CSNI-sponsored State-of-the-Art Reports (SOAR) on nuclear aerosols since 1979. In the latest SOAR published end of 2009 [10], the theoretical, experimental and modelling studies summarise the status and current understanding of a wide range of nuclear aerosol topics. The significance of these results is that they allow for the identification of existing gaps in code capability and experimental data that prevent accurate predictions of the source term. A general conclusion from this document is that there still exist a number of items for which additional work is required. However, the status of aerosol codes and experimental databases has improved substantially since the publication of the SOARs.

Pool scrubbing was extensively investigated in the 1980s' and 1990s' in several large research programmes, (Cunnane et al., 1985 [1]; Guentay, 1990 [3]; Hashimoto et al., 1988 and 1991 [4], [5]; Hillary et al., 1966 [6]; Marcos Crespo et al., 1993 [8]; McCormack et al., 1989 [9]; Ramsdale et al., 1992 [12]; Dehbi et al., 2001 [2]). These investigations provided valuable insights into the effects of variables like particle type and size, gas flow rate, submergence and carrier gas composition, on the decontamination factor. Most of these investigations were integral in nature and were carried out under bubbly flow and globule regime. The most common pool scrubbing codes (SPARC, BUSCA, SUPRA) are based on the data from investigations from this time (Ramsdale et al., 1995 [12]; Owczarski et al., 1985 [11]; Wassel et al., 1991 [14]). As a result, calculations by both stand-alone and integral code models indicate satisfactory agreement with simple experiments for integral retention, but the models cannot capture the aerosol retention with more complicated geometries and high gas flow rates (jet injection regime, churn-turbulent flow) in the water pools. This is due to the fact that the models have been developed for the aerosol and soluble gas scrubbing taking place when gas bubbles rise through stagnant pools of water.

In these models, the pool is divided into three regions: the injection region, the bubble rise region, and the surface / bubble break-up region. In many of the pool scrubbing sequences, however, the flow is far more complex than modelled in the pool scrubbing codes. Therefore, comprehensive characterisation of the two-phase flow hydrodynamics during pool scrubbing sequences would be a necessary prerequisite for improved modelling of pool scrubbing phenomenon. Particular attention should be given to characterising the two-phase flow hydrodynamics under high gas injection conditions, high gas temperatures and the use of multi-hole spargers as well as the effects of submerged structures and contaminants/impurities/surfactants on the hydrodynamics and fission product retention. The decreased retention within the pool due to elevated pool temperatures close or at saturation condition is another issue being of special interest with regard to Fukushima Daiichi accidents.

A.1.6.2 Safety Research Interest

Safety research interest is **High**: Fission product (FP) transport through the reactor coolant system and into the containment (especially through the suppression pool), as well as potential FP release from the containment, are of major importance when assessing the release of activity to the environment. Based on theFukushima Daiichi accident progression analysis, several specific pool scrubbing related research needs can be identified.

1. Injection geometry. In the Fukushima Daiichi NPPspower plant, pool scrubbing of aerosols and gas phase iodine species is assumed to have taken place in the water space of the suppression pool. There were three paths of the gases carrying fission products to be released to the suppression pool: i) through the reactor safety valve spargers, ii) from the drywell through condensation pipes, and iii) from the gas discharge of the RCIC and high-pressure coolant injection (HPCI) systems. The geometry of the injection in the three cases is very different with the safety valve spargers consisting of a large number of very small holes through which the gas is released at a very low position within the suppression pool, the multiple number of condensation pipes having one large diameter opening and less submergence by water, and the RCIC and HPCI discharge spargers having different geometries in different units. Therefore, different pool scrubbing efficiency is also expected for the different gas injections into the suppression pool resulting from the different injection geometry, submergence, and gas flow rate into the pool. It should be noted that the release path iii) was of limited importance in Fukushima Daiichi because the core degradation took mainly place after the RCIC and HPCI injection was stopped or degraded.

2. Containment venting and other transients. Pool scrubbing phenomena in BWR suppression pools is linked to containment venting processes, if such are done from the suppression pool gas space. The primary containments in the Fukushima Daiichi NPPs power plant had a possibility for depressurisation either from the drywell or the gas space of the suppression pool. The systems installed are called hardened vent systems (similar to US plants) and do not contain specific aerosol or iodine filters. If the venting line connected to the suppression pool gas space is used, as it was the case for units 1 and 3, this leads to a depressurisation of the suppression pool first. If the pressure difference to the drywell gets large enough, the steam/gas atmosphere of the drywell starts to flow through condensation pipes into the suppression pool, and thereby pool scrubbing works for aerosol retention. The conditions in the suppression pool are highly transient during the venting operations with potentially a rapid decrease of the pressure in the suppression pool. Pool scrubbing efficiency and possible resuspension from the pool has not been well-characterised under such conditions.

3. Saturated / boiling conditions. During the accidents, the water in the suppression pool heated up due to frequent releases of steam into the suppression pool resulting in the parts of the pool being under saturated conditions. The saturated / boiling conditions in the pool have an effect on the efficiency of the pool to retain the fission products as the steam condensation in the water pools is known to increase aerosol retention. Iodine chemistry is also significantly affected by the temperature. Boiling conditions may also have been generated as a result of the pressure decrease in the containment due to containment venting. The possibility of fission product release from the suppression pool water under boiling conditions has not been sufficiently investigated. In addition, boiling together with high gas injection rates may generate droplets on the pool surface, and some activity may be transported away from the pool in these droplets.

4. The water pH and the chemistry effects. It is expected that during the progression of a severe accident, many impurities enter the suppression pool and may alter the pH and chemical composition of the water in the suppression pool. For example, acidic gases are generated by molten-core concrete interaction and cable fires, boron may be added to the coolant, etc. This can significantly change the pH of the pool affecting the pool scrubbing efficiency for iodine and enabling release of iodine from the pool. The suppression pool walls may also contribute to the pH change in the water. Chemical reactions between iodine and other compounds may also decrease the efficiency of pool scrubbing, and promote release of volatile species from the suppression pool.

5. The use of sea water or "dirty water" for cooling the reactor cores. This specific concern is related to Fukushima. Such water contains a wide variety of impurities which may affect the potential release of iodine and some other fission products from the water in the long-term. This topic has been investigated to limited extent in earlier research projects. Some new projects have been initiated after Fukushima, but only incomplete information is available so far on the effects of impurities in the water pool. This topic is addressed in section A.2.1 of this document.

6. Characterisation of suppression pool hydrodynamics in the realistic flow regime. Characterisation of the hydrodynamic behaviour of the two-phase flow in the suppression pool, i.e. the bubble size, gas and liquid phase velocity, as well as the interfacial area, in the flow regimes which are realistic for suppression pools is needed in order to analyse the extend of pool scrubbing taking place during the accidents. The effect of different injection geometries (see point 1), flow rates, and pool temperature on these parameters is important. This is a necessary prerequisite for understanding the fission product retention in the suppression pool.

A.1.6.3 Decommissioning Interest

Decommissioning interest in pool scrubbing **is low**: The information about pool scrubbing supports estimates of FP distribution phenomena and behaviours inside the plant and the release to the environment during the accident period, however, it is not expected that significant information on the latest condition of radioactive materials can be gained from the detailed examination regarding pool scrubbing itself, while research on deposited fission products and source term remains importantplant investigations (see A.1.5).

A.1.6.4 Potential Examinations

The most critical information related to releases from the containment into the reactor building by leakages, and pool scrubbing is to determine the dominant release paths in the units 1-3 in Fukushima. For this, any evidence of very high activity in the suppression pool, in the vent lines, close to the vent lines, and other possible leakage locations such as the PCV head flange would be of importance. In addition, any visual observation of possible leakages, high temperature areas, corroded components, status of sealing of containment head, etc. would be useful in determining the possible leakage paths. This information is of major importance for assessing the source term from the plants, and getting it from other sources is not possible.

However, it is noted that corrosion and degradation of the structures proceeds with time; and once visual observation of the structures will become possible; the usefulness of the information may be limited.

A.1.6.5 Ongoing R&D Activities

After 2011, several investigations both in Japan and internationally have been initiated to address more in detail the retention of aerosols and gas phase iodine compounds in water pools related both to suppression pools in BWRs, and wet scrubbers in filtered containment venting systems (FCVS). Special focus in the recent investigations is placed on several phenomena which were not investigated in detail earlier. Some of the examples are:

- EU-PASSAM project (2013-2016) to address, e.g. long-term behaviour of iodine in the water pools, the retention of organic iodides in water, the effect of pool hydrodynamics on aerosol retention, the effect of additives and surfactants on the hydrodynamics and aerosol retention, and the two-phase flow hydrodynamics in the presence of submerged structures (EU)

- French national programme MIRE launched in 2013 to investigate FP filtration on solid filtering media and in liquid scrubbing filters in containment venting systems. The filtering efficiencies for existing venting systems for gaseous iodine species, particularly org-I and IxOy, and for gaseous RuO4 are being looked at more thoroughly. Also, investigations of innovative filtering media (zeolites, metal-organic frameworks) are conducted (France)
- Investigation and model improvement of the scrubbing model that describes possible degradation of the aerosol removal in the pool region under fairly rapid depressurisation conditions presumed in severe accident. The experimental research includes both large scale (equivalent to S/P) and small scale experiments. The large scale test cases were dedicated for identifying influences of operating parameters including depressurisation rate on the decontamination factors as a function of aerosol diameter with small scale distortions. Instrumentations of the large scale test are not necessarily oriented to the high resolution. On the other hand, the small scale test cases were dedicated for identifying underlying mechanisms such as detailed two phase flow evolution and aerosol movements inside each bubble with relative high resolution instrumentations. (Japan)
- Swiss national project for investigation of iodine retention in the FCVS, with the focus on improving retention of organic iodides in liquid pools, and ensuring long-term retention of iodine in wet scrubbers, new phase of research started in 2012 (Switzerland)
- Planned THAI3 (to start 2016) to address resuspension of aerosols from saturated suppression pool (NEA)
- Planned pool scrubbing experiments at Research Center Jülich, Germany (Germany)
- Development and characterisation of a Korean FCVS is under progress at Korea Atomic Energy Research Institute in collaboration with industry participants (Korea). A scaled test facility at full height and full pressure with reduced flow area is established for the qualification proposed FCVS.

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A.1.7 *Hydrogen Distribution and Combustion*

A.1.7.1 Background

In the case of a severe accident in an NPP, hydrogen (H_2) produced by the metal-water reaction in the core and after RPV failure and melt release by molten core concrete interaction (MCCI) in the cavity may be released into the containment in which typically large amounts of oxygen (O_2) are present due to steam and the initial air inventory together with steam released during the accident. As the accident progresses, a flammable gas mixture can be produced. Upon ignition, slow flame propagation can be generated, and can accelerate or be quenched depending on the prevailing atmospheric conditions (geometry configuration, gas composition, turbulence level, etc.). Note: Carbon monoxide from MCCI is a second flammable gas that could be produced during the course of severe accidents together with other non-condensable gases.

Hydrogen explosions were observed in Fukushima Daiichi units 1, 3 and 4 outside of the N_2 inerted containments in the reactor building causing massive damages to it, distributing radioactivity into the environment, and exposing the spent fuel pools in the damaged buildings. It is believed that the hydrogen causing the explosion in unit 4 originated from unit 3, and was transported to unit 4 through the Standby Gas Treatment System (SGTS) and the common pipe work downstream towards the stack. The hydrogen causing the explosion in units 1 and 3 originated from the hydrogen released into the containments of the respective units.

The analyses of the accident progression carried out in the NEA Benchmark Study of the Accidents at the Fukushima Daiichi Nuclear Power Plant (BSAF) project show that a sufficient amount of hydrogen can be expected to have been generated in units 1 and 3 during core degradation to cause hydrogen explosions. Available transport paths to the reactor buildings have been postulated, and some simulations can predict combustible concentrations of hydrogen in the reactor buildings at the time of the explosions. The hydrogen generation and transport are considered in the BSAF project Phase 2, in which also the effect of hydrogen combustion to the source term will be addressed.

For operating reactors, the status on the implementation of hydrogen mitigation means for light and heavy water reactors including systems already installed and contemplated have most recently been described in a Status Report on “Hydrogen Management and Related Computer Codes” written by an NEA group of experts [1]. Information on the issues described below in a short manner can be found there in a more detailed way. Especially information on hydrogen mitigation and implemented measures in a large number of countries and different plant types is provided in this status report. The report shows the mitigation measures are implemented in many NPP containments while very seldom in buildings surrounding the containment, as e. g. the reactor building. It shows further that the principal phenomena are in general well understood and that codes exist enabling detailed analyses of the expected phenomena of H_2 combustion and recombination. Open issues are related to analyses of H_2 fast deflagrations and detonations. One reason for this might be that the primary intention is to prevent such phenomena by dedicated measures. In any case research activities to either of the topics mentioned below is ongoing.

A.1.7.2 Safety Research Interest

Safety research interest is medium: Gaps in knowledge on hydrogen combustion in the containment and other areas of the power plant are a key to the proper assessment of hydrogen risk. In addition, possible hydrogen explosions may have a significant effect on the containment and structure integrity and the fission product release from the plant.

Hydrogen combustion in the containment and other areas of the power plant has a potential to threaten the integrity of the containment and to damage safety systems. This is not the case for BWR, as most of the containments are nitrogen inerted. Despite extensive research in the area of hydrogen distribution, combustion and mitigation during the past years (as described in the following sections), there are still open issues which have a medium safety significance. Some examples of detailed open issues related to combustion phenomena are:

- Scaling of H_2 stratification/erosion phenomena to plant dimensions
- Effects of gas mixture turbulence levels on flame propagation speed
- Effect of steam content on flame acceleration, flame deceleration and flame quenching
- Flame propagation in multi-compartment environment
- Various criteria for flame inception, propagation, acceleration and transition to detonation, “Static” vs “Dynamic” criteria.

Some questions remain to hydrogen countermeasures, especially to the Passive Autocatalytic Recombiner (PAR) behaviour:

- PAR performance in counter-flow environment typical of small containments or installation in narrow compartments (PAR start-up delay or lower efficiency)

A.1.7.3 Decommissioning Interest

During decommissioning, drilling and cutting of structures may involve flames. Currently, at Fukushima Daiichi, nitrogen injections are used to inert the PCV gas spaces in order to minimise the risk of potential hydrogen combustions or explosions. The information about concentration of hydrogen, however, will continue to be is obtained from dedicated monitoring systems.

On the other hand, research on hydrogen distribution and combustion mechanism and phenomenology as well as visual inspections of the structures of the Fukushima Daiichi NPPs reactor buildings may provide useful information on the degree of damage caused by the explosions, but there is limited information on the effects and possible location of hydrogen explosions during the accident, and eventually current status of hydrogen residue.

The decommissioning interest in hydrogen distribution and combustion **is low** as information on such mechanisms and phenomenology as well as damage to the reactor building, components, and structures may be of limited use due to other affecting factors. There may be residual hydrogen in the piping and this need to be taken into account.

A.1.7.4 Potential Examinations

Determination of possible transport paths of hydrogen in the Fukushima Daiichi plant can be done based on plant specific information, e. g. construction details, together with visual observation of the damage to the structures caused by the hydrogen explosions. This is foreseen to be a part of the ongoing BSAF project Phase 2. By visual observation of the damage to the structures in the reactor buildings in combination with the analysis of the accident progression, information can be deduced on the combustion / detonation phenomena and the level in the building where it might have started, as well as on the amount of hydrogen which was transported to the reactor buildings. Using such information, also possible hydrogen stratification in the reactor buildings could possibly be evaluated.

Some of the structures and components in the reactor buildings in Fukushima Daiichi do no longer exist, and some are still inaccessible due to high dose rate levels. Therefore, a careful documentation of the damage in different parts of the power plant, e.g. above the primary containment vessel, in the area of the spent fuel pool, and in the different volumes/rooms in the reactor building, etc. is of vital importance. Any visual observation of possible leakages, high temperature areas, corroded components, etc. will help to determine the hydrogen release and transport.

A.1.7.5 Ongoing R&D Activities

Hydrogen research activities can be grossly lumped in four areas: generation, distribution, combustion, and mitigation, the last three of which are actively investigated by the international research community and are very much interconnected.

A.1.7.5.1 Hydrogen Distribution

Distribution deals with the potential formation of flammable H_2 clouds, combustion deals with H_2 burns which can take the form of slow to fast deflagrations and up to detonation, while mitigation looks at ways to reduce H_2 risk by e.g. using hardware (PAR's, igniters or inertisation of rooms).

Two large international programmes on H_2 distribution have been concluded recently. The NEA THAI-I and THAI-II [2] programmes provided 3D data on the formation and erosion of light gas (He, H_2) clouds under various conditions (condensation) in the single compartment THAI vessel. The ERCOSAM-SAMARA project [3], funded by EU, Switzerland, and Russia, include various experiments addressing accident scenarios scaled down from existing plant calculations to different thermal-hydraulics facilities (TOSQAN, MISTRA, PANDA, SPOT). The tests investigate hydrogen concentration build-up and stratification during a postulated accident and the effect of the activation of Severe Accident Management systems (SAMs), e.g. sprays, coolers and simulated PARs. Both test programmes have been accompanied by intense analytical activities to improve and validate various computational methods. Code benchmark activities on the basis of conceptual near full scale HYMIX facility will eventually provide a further opportunity to evaluate the applicability of the various methods to the study of scaling issues. Both of these programmes are in the process of being extended.

JAEA has started the ROSA-SA programme in 2013, which focuses on the containment thermal hydraulics related to hydrogen risk, over-temperature damage of the containment vessel and the FP transport [4]. A large-scale test facility called Containment InteGral Measurement Apparatus (CIGMA) was constructed in March 2015 and the first test was conducted in October 2015, entrusted by NRA. The CIGMA facility has a cylindrical geometry of 11m in height and 2.5m in diameter, which is characterised by the high design temperature of 573 K for the pressure boundary and 973 K for the gas injection, and the design pressure of 1.5 MPa. It is expected to enlarge validation-range for prediction models and validate several mitigation procedures.

A.1.7.5.2 Hydrogen Combustion

In the last decade, the international nuclear research community has been very active in the H_2 combustion area, conducting a host of experiments and accompanying simulations in order to identify and understand the important governing phenomena. On the simulation front, collaborative efforts have been taking place in the NEA framework, which culminated in the International Standard Problem ISP-49 [5]. The stated goal of the ISP-49 was to determine the level of adequacy of the current modelling and simulation tools with regards to H_2 combustion, as well as to identify knowledge gaps. The ISP-49 dealt specifically with the simulations of two types of H_2 combustion processes in two facilities, THAI and ENACCEF, which display very different geometrical and initial combustible mixture characteristics.

The German THAI facility is a 60 m^3 unobstructed cylindrical volume, in which flammable mixtures were prepared prior to ignition in the bottom or top of the vessel. A slow combustion regime resulted. The THAI tests are directly relevant to slow combustion processes in unobstructed areas of the reactor such as the dome. The

ENACCEF [6] facility is a partially obstructed acceleration tube (0.15 m diameter, 3.7 m long) in which a non-uniform H_2 concentration was set up prior to ignition. The set-up is directly relevant to obstructed areas such as series of NPP rooms connected by door openings. The test displayed a fast combustion regime.

In addition to these programmes, the HYCOM tests [7], conducted in the large scale RUT facility, aimed at studying slow and slow-to-fast turbulent flame propagation in multi-compartment geometries typical of NPPs. The RUT facility is a large, variable area duct (180 m^3 in volume, 35 m long) subdivided into a number of compartments separated by obstructions.

From the various simulations of combustion tests, it can be concluded that Lumped Parameter (LP) and Computational Fluid Dynamics (CFD) codes are reasonably capable of predicting slow deflagration in a homogeneously mixed open-space. In obstructed spaces with H_2 concentration gradients, codes were able to reproduce the flame trajectory and qualitatively the short pressure peak. However, more quantitative agreement was only possible in the post-blind test computations, after code users were able to adjust their combustion models after analysing the data. In addition, when steam is present, codes systematically over-predict the flame speed. It can be concluded that in the current state of knowledge, LP and CFD codes need to incorporate more advanced models than the current Eddy Break-Up/Eddy Dissipation models to provide high quality blind predictions under NPP conditions.

A.1.7.5.3 Hydrogen Mitigation

Recent activities in H_2 mitigation are exemplified by the Hydrogen Mitigation Experiments for Reactor Safety (HYMERES) (2013-2016) [8] project which aims at improving the understanding of hydrogen risk phenomenology in containment to validate and improve modelling in support of plant safety assessment. In comparison to previous H_2 projects, HYMERES is characterised by three new features: 1) Realistic flow conditions 2) Interaction of safety components (spray, coolers, PAR's); 3) System behaviour for selected reactor types (e.g. various BWR, PWR or pressurised heavy-water reactors). Knowledge expected to be gained from the HYMERES project will contribute to the improvement of the Severe Accident Management (SAM) measures for mitigating hydrogen risks.

Tests are being performed in two large-scale facilities with complementary features, PANDA and MISTRA. PANDA, operated by PSI, is a large-scale, multi-compartment facility simulating a GE SBWR reactor, and has a volume of 520 m^3. Full vertical heights are preserved, at a volume scaling of 1:25. MISTRA is a large experimental facility belonging to the CEA with a volume of 100 m^3. The facility was modified to accommodate two compartments. The PANDA and MISTRA instrumentation is comprehensive to provide CFD-grade data.

Hydrogen mitigation using PAR's was studied extensively in NEA THAI-I and THAI-II [2]. The highly instrumented THAI facility allowed detailed local studies of PAR operations under severe accidents (excess O_2, O_2 starvation, aerosol poisoning, deflagration within PAR's). PAR's of different vendors were tested, and their performance assessed.

Further research on hydrogen distribution, combustion and mitigation has been proposed in follow-up projects: THAI-3, SAMHYCO. In these projects, some of the open issues listed above will be addressed. In Korea, a test facility named SPARC (Spray-Aerosol-Recombiner-Combustion, 80 m^3, 15 bar) [9] for the investigation of hydrogen distribution and the performance of PAR under severe accident environment has been constructed. A shakedown test is under progress. This facility is going to be used for the validation of SAMG actions for hydrogen mitigation, containment pressure control, and fission product removal.

A.1.7.6 References

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A.2 SCC Performance and Condition

A.2.1 *Salt and Concrete Debris Effects (Primary System and Spent Fuel Pool)*

A.2.1.1 Background

A.2.1.1.1 Seawater effect

The reactors at the Fukushima Daiichi site employed seawater as an emergency coolant. Possible safety important consequences can be ascribed to the use of seawater as an emergency coolant. Seawater is chemically complex. Solute concentrations in seawater are shown in Table A.12 [1]. These concentrations are remarkably similar in the various oceans of the world. Some variations are possible at special sites such as those near river outlets to the sea, but otherwise the compositions shown in the table are representative. Much of the attentions to issues of seawater have focused on the sodium chloride concentration of seawater. The sodium chloride concentration is high and deserves attention. There are other constituents of seawater that are of interest as well such as the carbonate concentration which are often found in river water.

Table A.12. Compositions of "raw" water sources including seawater

Solute	Solute Concentration (mole/kg-H_2O)			
	Rain	River	Lake	Seawater
Na^+	0.000083	0.000274	0.000973	0.48586
Mg^{2+}	0.000008	0.000169	0.000314	0.05520
Ca^{2+}	0.000008	0.000374	0.000181	0.01065
K^+	0.000005	0.000059	0.000153	0.01058
Sr^{2+}	-	-	-	0.00009
Cl^-	0.000093	0.000220	0.000497	0.56572
SO_4^{2-}	0.000033	0.000117	0.000103	0.02927
Br^-	-	-	-	0.00087
F^-	-	-	-	0.00007
HCO_3^-	0.000007	0.000958	0.001506	0.00241
$B(OH)_3$	-	-	-	0.00044
Ionic Strength	0.00021	0.00208	0.00276	0.723

Some of the effects of seawater are: (a) Metal corrosion, (b) Solids precipitation, (c) Effects on radiolysis and iodine chemistry and (d) Fuel corrosion and leaching

(a) Metal Corrosion – Aerobic and Anaerobic Corrosion of Mild Steel

The aerobic corrosion of mild steel proceeds by the summary reaction:

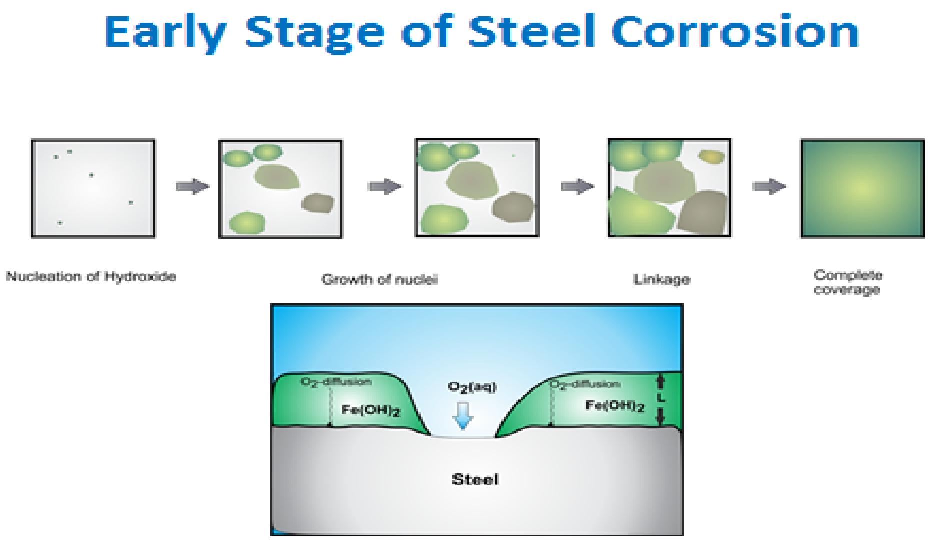
$$Fe(s) + ½ O_2(aq) + H_2O \rightarrow Fe^{2+} + 2\,OH^-$$

In the absence of any passivation, this reaction can proceed at the rate of oxidant mass transport to the bare metal surface. Once the local concentration of ferrous iron, Fe^{2+}, exceeds the saturation limit for ferrous hydroxide, $Fe(OH)_2$, a passivating film of this solid forces the reaction to proceed following parabolic kinetics: $Fe^{2+} + 2\ OH^{-} \leftrightarrow Fe(OH)_2(s)$

The presence of high concentrations of chloride can cause film formation by enhancing the solubility of both ferrous and ferric iron: $Fe(OH)_2(s) + Cl^{-} \leftrightarrow FeCl^{+} + 2\ OH^{-}$

Seawater is somewhat more complicated because the presence of sulphate and carbonate can lead to the precipitation of so-called "Green rusts" on the corroding surfaces (Figure A.3).

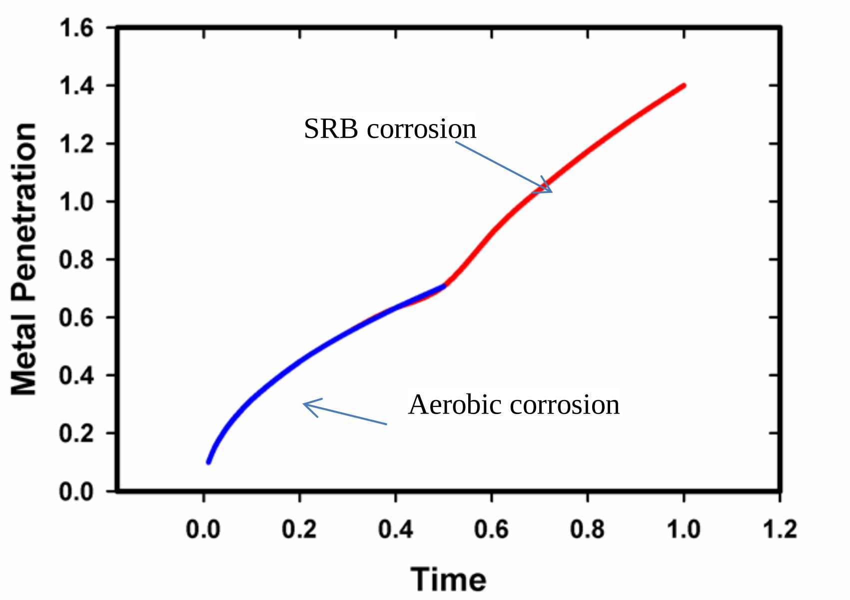
Figure A.3. Corrosion of steel



Aerobic corrosion can be inhibited by suppressing the oxygen concentration of the electrolyte say by inoculation with hydrazine. Steel can, however, corrode anaerobically: $Fe(s) + H_2O \rightarrow Fe^{2+} + 2\ OH^{-}$

This is typically slow kinetically. What is observed with mild steel in ocean water is that aerobic corrosion progresses following parabolic kinetics until the passivating layer of hydroxide (or Green rust) is thick enough that conditions near the metal interface become anaerobic. There is then a sudden burst of corrosion (Figure A.4) [2]. This second burst is due to the action of "sulphate reducing bacteria." Sulphate reducing bacteria are anaerobic species that survive by metabolising sulphate ion. Sulphate is present in seawater. It can also come from the thermal or radiolytic degradation of Hypalon cable insulation or the degradation of concrete within the containment. Whether or not sulphate reducing bacteria (SRB) or other bacteria will pose a corrosion issue in the reactor environment is not known. The issue of biota brought into the reactor by the use of raw water will be discussed further at the end of this section.

Figure A.4. Rate of seawater corrosion of mild steel



(b) Metal Corrosion – Corrosion of Aluminium

Salt water is usually regarded as very corrosive towards aluminium. The problem is that the passivating oxide film protecting aluminium can be ruptured by the chloride ion. The rupture is accentuated by the presence of catalyst such as mercury. Aluminium corrosion may be an issue especially in spent fuel pools where it has been used for fuel storage racks.

(c) Solids Precipitation

With a few notable exceptions (silver, copper, lead and thallium) chlorides of metal ios are water soluble. Thus chloride from seawater is not expected to lead to solids precipitation when used as an emergency coolant. The same cannot be said for other constituents of seawater. Seawater is essentially saturated with respect to both calcite ($CaCO_3$) and gypsum ($CaSO_4$). Addition of calcium from the corrosion of concrete – especially the dissolution of calcium hydroxide from concrete will lead to the precipitation of calcite on surfaces exposed to seawater. This is likely to be especially problematic in the reactor vessel. Addition of sulphate from the degradation of Hypalon insulation could also lead to the precipitation of gypsum. These precipitates can incorporate fission products such as barium, strontium and rare earths. That is the incrustations formed by these precipitates could be radioactive and need to be removed to facilitate reactor clean-up. Other impacts of these precipitates on cooling have not yet been evaluated, nor has the effect of precipitation of calcite and gypsum on coolant flow rates been evaluated. Observations of precipitated solids within containment waters is of modest concern save for the formation of deposits on inlet screens for coolant flow. Of more concern is the formation of deposits within the coolant flow pathways for any residual fuel assemblies still present within the reactor vessels.

(d) Effects on Radiolysis

Chloride will affect the radiolysis of water and can lead to the formation of hypochlorous acid (HOCl) and even chlorine (Cl_2): It can also have effects on the distribution of water radiolysis products, and subsequently on iodine behaviour.

(e) Leaching and Dissolution of Fuel

Zirconium alloy cladding of fuel is relatively inert towards even very hot chloride solution. Ruptured cladding can expose fuel to the seawater. Fuel dissolution in an electrolyte progresses by formation of the hexavalent species:

$UO_2(s) + \frac{1}{2} O_2(aq) + H_2O \rightarrow UO_2^{2+} + 2\,OH^-$

This dissolution is enhanced greatly by the presence of carbonate:

$UO_2^{2+} + HCO_3^- \rightarrow UO_2CO_3^{\circ} + H^+$

The reaction of aqueous oxygen with fuel is kinetically slow. When uranium dioxide dissolution is desired, it is usual to add multivalent cations to accelerate dissolution:

$2\,FeOH^{2+} + UO_2(s) \rightarrow 2\,FeOH^+ + UO_2^{2+}$

$2\,FeOH^+ + \frac{1}{2} O_2(aq) + H_2O \rightarrow 2\,FeOH^{2+} + 2\,OH^-$

To some extent, this oxidation-reduction couple drove fuel degradation and fission product release observed during the Paks accident [3]. Seawater will assist this couple by enhancing the solubility of both ferrous and ferric ion in solution. Some data on corrosion of uranium dioxide in chloride solutions are available from the waste disposal community though these data are for lower radiation dose rates than will be expected in the aftermath of reactor accidents.

(f) Interaction with molten core

The high temperature reaction between sea salt deposit and $(U,Zr)O_2$ simulated corium debris (sim-debris) was examined at JAEA [4] in the temperature range from 1 088 to 1 668 K. A dense layer of calcium and sodium uranate formed on the surface of a sim-debris pellet at 1 275 K under airflow, with the thickness of over 50 µm. When the oxygen partial pressure is low, calcium is likely to dissolve into the cubic sim-debris phase to form solid solution $(Ca,U,Zr)O_{2+x}$. The diffusion depth was 5–6 µm from the surface, subjected to 1 275 K for 12 h. The crystalline MgO remains affixed on the surface as the main residue of salt components. A part of it can also dissolve into the sim-debris. Similar effects may be observed in Fukushima Daiichi corium debris.

(g) Biotic Issue

We know that biotic species were able to thrive in the coolant of the damaged reactor at Three Mile Island Unit 2 (TMI-2). The TMI-2 accident did not involve use of raw water. Nevertheless, once the reactor coolant system was opened and hydraulic fluid spilled into the system, biotic species were able to establish a foothold in the system. What biota may have been brought into the reactors by the use of raw seawater at Fukushima Daiichi is not known. Certainly, the presence of sulphate reducing bacteria and the effects of these bacteria on mild steel corrosion as mentioned earlier cannot be discounted. Other bacteria could affect corrosion [5]. We do know that the biota at TMI-2 survived despite the radiolytic generation of hydrogen peroxide in the coolant water. On the other hand, the biota did not survive the inoculation of the coolant water with concentrated hydrogen peroxide. We do not know if biota will survive in the chloride-contaminated raw water where there is radiolytic generation of hypochlorous and chlorous.

A.2.1.1.2 Concrete Debris Effects

For the purposes of this discussion, concrete is considered a mixture of a porous cementitious phase, sand and coarse aggregate. The cementitious phase is the most readily leached and dissolved by containment water. An essential product of the concrete curing process is calcium hydroxide [$Ca(OH)_2$] which will dissolve in containment water. Sand and gravel used in concrete for structural purposes are highly variable in their compositions. In many cases (but certainly not all), the sand is relatively pure silicon dioxide [SiO_2]. Silicon dioxide is slowly dissolved in basic water to produce various anions related to silicic acid such as $H_3SiO_4^-$, $H_2SiO_4^{2-}$, $Si_4O_8(OH)_4^{4-}$, $Si_4O_6(OH)_6^{2-}$, and $Si_2O_3(OH)_4^{2-}$.

Though these aqueous species are never present at high concentration, they are noteworthy because they do tend to occlude and degrade ion exchange resins that may be used to decontaminate containment waters in the post-accident recovery efforts.

Coarse aggregate used in structural concrete can be broadly categorised as either siliceous or calcareous. Siliceous aggregates include granite, granodirite, illite, and basalt; at Fukushima Daiichi, the siliceous aggregates were basaltic. For the purposes of these discussions, it is adequate to consider these materials to be mixtures of silicon dioxide, $(Na,K)AlSi_3O_8$, and $CaAl_2Si_2O_8$. Calcareous aggregates are usually limestone [$CaCO_3$] though occasionally dolomite [$CaMg(CO_3)_2$] is found.

Corrosion of the siliceous aggregates is typically slow and gives rise to the same silicic acid anion concerns mentioned in connection with corrosion of sand. Corrosion of calcareous aggregates in seawater is slow because seawater is usually saturated in both calcium carbonate and magnesium carbonate. The saturation can be broken in the post-accident environment because of continuing carbon dioxide dissolution in the water:

$CO_2(gas) \leftrightarrow CO_2(aq)$

$CO_2(aq) + H_2O \leftrightarrow H_2CO_3$

$H_2CO_3 + OH^- \leftrightarrow HCO_3^- + H_2O$

$HCO_3^- + OH- \leftrightarrow CO_3^{2-} + H_2O$

$CaOH^+ + HCO_3^- \leftrightarrow CaCO_3(s)\downarrow + H_2O$

This process will tend to move calcium carbonate solid from the concrete to regions of high carbon dioxide concentrations such as surfaces and water inlets.

Calcium saturation of the containment waters can also be broken by sulphate and the formation of gypsum deposits: $CaOH^+ + HSO_4^- \leftrightarrow CaSO_4.nH_2O$

Continuing additions of sulphate to the containment waters can be expected by the hydrous radiolysis of Hypalon cable insulation.

Perhaps the most dramatic reduction of calcium ion concentration can come from the precipitation of calcium phosphate by interaction with buffers used to maintain basic conditions in sump water. A variety of calcium phosphate can precipitate such as $Ca_3(PO_4)_2$, $CaHPO_4$, $CaH_2PO_4.H_2O$ and evolve to the thermodynamically favoured hydroxyapatite phase – $Ca_{10}(PO_4)_6(OH)_2$. It is also possible for borate to precipitate calcium as variety of borate species such as CaB_2O_4 and CaB_4O_7. Anything that reduces the calcium ion concentration of the containment water or the carbonate concentration will enhance the corrosion of the coarse aggregate in concrete that uses calcareous aggregate.

The leaching of concrete is not a fast process, but it may be of some concern in the many years it takes to recover a reactor following an accident. The process kinetics involves transport of water through the pores of cementitious phase to a water soluble site, dissolution of material from the walls of the cement pore which will also change the permeability of the concrete, and transport of the solution out of the concrete. Rate control can arise in connection with any one of the steps in the overall process as well as from the solubility of the concrete materials. For a readable review, see T. Ekström [6]; an illustration of data obtained from experiments is shown in Figure A.5.

For structural considerations, leaching of concrete becomes of serious concern if the steel reinforcing bars are exposed to the corrosive action of water. If the containment water is kept basic, abiotic corrosion of the reinforcing bars will not be an especially serious concern – certainly no more of a concern than is abiotic corrosion of mild steel liners in the containment. Biotic corrosion especially by sulphate reducing bacteria may still take place.

Seepage of corrosion fluids which may be contaminated with radionuclides through a concrete structure may be of concern. The rate of seepage depends on the permeability of the concrete. This permeability depends on the water-to-cement ratio used in the formulation of the concrete and in the rate of concrete curing. What is of interest is the permeability of the cementitious material. Some typical values are shown in the table below:

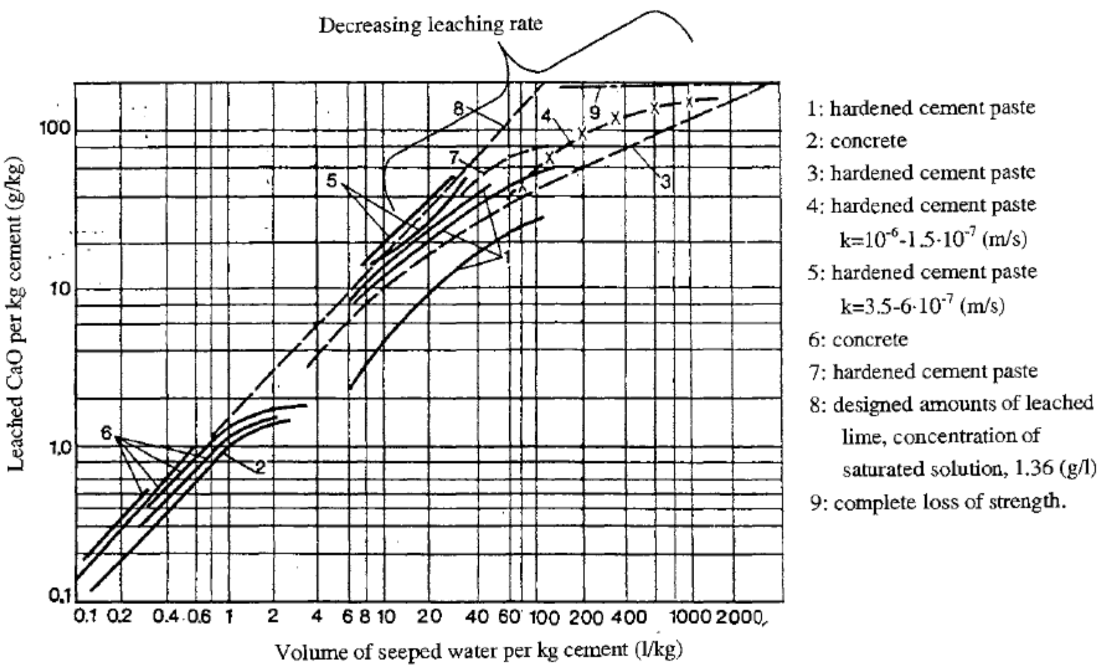
Table A.13. Values for the permeability of the cementitious material

Water-to-cement Ratio	Porosity (-)	Cement phase permeability(m/s)
0.23	0.25-0.33	10^{-16}-10^{-12}
0.47	0.42-0.53	10^{-15}-10^{-9}
0.71	0.55-0.62	10^{-13}-10^{-8}
1.00	0.65	10^{-12}-10^{-6}

More dramatic flows of contaminated liquids from the concrete can occur if the concrete has cracked due either to thermal or pressure stresses in the concrete during the accident. Liquid can flow, typically, through these cracks. This liquid flow will expose the reinforcing bar to any corrosive actions of the liquid as well as allow the escape of radioactive materials.

During decommissioning of MCCI debris, cracks may be produced in the partly ablated concrete structures. Moreover, the removal of corium debris may affect the mechanical strength of the remaining concrete structures.

Figure A.5. Illustration of the kinds of data available for the leaching of concrete by water. These data do not reflect the particular chemistry of water that will be present within reactor containments following a reactor accident.



A.2.1.2 Safety Research Interest

The safety research interest in this area is low. There are many possible effects of seawater on the reactor and the reactor containment. Observations and findings during the decommissioning of the damaged Fukushima Daiichi reactor could help to narrow the possibilities to a tractable set of important effects that should be included in the development of accident management strategies for nuclear power plants in which seawater (or raw water may be used in case of emergency cooling.

A.2.1.3 Decommissioning Interest

The decommissioning interest in this area is medium. During the stabilisation stage of the damaged reactors there has been a substantial evolution of the coolant from the raw seawater used initially. Coolant is no longer chloride rich and it has been inoculated with hydrazine to suppress the oxygen concentration. Consequently, inspections and examinations must be concentrated on phenomena and effects that will have survived the evolutions of the coolant chemistry.

In addition, long term corrosion mechanism as well as effects of seawater on SSCs and fuel debris may be important to decommissioning.

Seawater may be is contaminated with aggressive bacteria. As discussed before, during the TMI-2 clean-up effort, biota was able to survive within the TMI-2 vessel. The recovery operation was stopped because the visibility in the TMI-2 vessel water was reduced to a centimetre. The recovery effort resumed after the use of strong hydrogen peroxide to eliminate the biota. The question arise is that in Fukushima, whether hypochlorous also prevent bacterial corrosion or not. The biota from seawater is not the same as from river water.

The inspections and examinations should also focus on effects that need understanding to further the clean-up activities.

A.2.1.4 Potential Examinations

Table 14 shows the suggested observations and examinations to support the understanding of the effects of seawater and to support the remediation activities.

Examinations are prioritised primarily on remediation support in the order that remediation activities are planned. Lower priority is given to accident management support.

High priority examinations are those that support impending remediation activities. These examinations also likely support accident management.

Medium priority items are those that support longer term remediation activities. These examinations also likely support accident management activities.

Low priority items are those that support accident management activities but do not directly relate to short term remediation activities.

The specific reasons for the prioritisation are provided under the “support for remediation” and “support for accident management” sections under each item.

Table A.14. Suggested observations and examinations to gain understanding the effects of seawater

1. Details of water treatment and current water treatment	
Support for remediation	Essential for equipment and personnel protection in the remediation effort
Support for accident management	Basis for modelling the current aqueous chemistry status and for interpreting other observations and findings
Scope	To the extent possible describe water treatment including current chloride concentrations, hydrazine additions, borate additions and any buffer additions
Priority	high
Difficulty	low – much of this information can be obtained from accident management records and from sampling of water storage tanks
2. Evidence of past or ongoing corrosion	
Support for remediation	Determine the need for more or less intensive corrosion suppression efforts to preserve the integrity of the drywell liner, torus volume and the reactor vessel
Support for accident management	Validates corrosion modelling in computer codes to support accident management
Scope	Identify corrosion of aluminium and galvanised materials by the presence of flocculent precipitates in sumps and other water bodies Identify copper corrosion by copper content of precipitates Identify mild steel corrosion by the presence of iron hydroxide or green rust corrosion products on surfaces exposed to coolant
Priority	medium
Difficulty	medium; Identification of corrosion products can be done by visually or by remote sampling. Composition data will require analysis in a laboratory equipped to handle radioactive materials.
3. Evidence of biotic corrosion of mild steel	
Support for remediation	Determine the need for corrosion suppression actions to prevent biotic corrosion and to preserve the integrity of the drywell, torus and reactor vessel
Support for accident management	Determine the need to include biotic corrosion in accident management models and planning
Scope	Inspect mild steel surfaces for evidence of biological colonies which may be present in patches Test mild steel corrosion products for evidence of biological activity near the metal-corrosion product interface. Note that any samples of the corrosion products should be protected from degradation by air prior to testing.
Priority	low
Difficulty	medium: Inspection for biological activity can be done visually. Absence of evidence of surface colonies does not preclude biotic corrosion which may be taking place within the corrosion products. Testing for biological activity in the corrosion products requires an experienced laboratory capable of handling samples that can be radioactive

Table A.15. Suggested observations and examinations to gain understanding the effects of seawater *(cont'd)*

4. Evidence of fuel leaching and corrosion	
Support for remediation	Fuel leaching could provide a continuing source of radionuclides to the coolant that could interfere with remediation activities despite coolant clean-up efforts
Support for accident management	Validation of fuel leaching models
Scope	Determine uranium concentration in coolant adjacent to fuel exposed to coolant Sample surfaces of fuel exposed to coolant. Sample will need to be protected from degradation by air once extracted. Sample surfaces should be examined in a laboratory for presence of uranium peroxides and the like.
Priority	medium
Difficulty	high; Protection of samples from degradation in air is essential. Examination of samples will have to be done in a laboratory equipped to deal with radioactive samples.
5. Formation of carbonate and sulphate precipitates	
Support for remediation	Encrustations of precipitates may affect cooling efficiency and pose a continuing hazard due to radioactivity during remediation process
Support for accident management	Determine the need to model saturation phenomena and the incorporation of radionuclides in precipitates
Scope	Visually inspect surfaces especially near the surface-atmosphere interface for carbonate or sulphate precipitates Sample the precipitates and measure their radionuclide contents
Priority	medium
Difficulty	medium
7. Concrete Degradation	
Support for remediation	Loss of concrete integrity and the ability of contaminated fluids to escape from the plant would disrupt recovery and remediation activities
Support for accident management	Concrete degradation aside from the direct action of core debris is not usually an issue in reactor accident management
Scope	Examine concrete for evidence of cracking caused by thermal stresses and pressure stresses during the accident. Examine concrete surfaces for precipitates and for the loss of the 'finish' on concrete. If precipitates are present obtain samples
Priority	medium
Difficulty	low: Most of the examinations can be done visually. Extraction of samples of precipitates from surfaces of concrete will be more difficult but it is likely any precipitates are flocculants and can be extracted by suction.

A.2.1.5 Ongoing R&D Activities

The NRA is undertaking some research on seawater effects study on long term effects of corrosion of various metals (stainless steel, carbon steel, aluminium alloys) used in reactor pressure vessel, containment vessel and spent fuel pool; Results from these researches are expected in 2016. The main outcomes so far are as follows;

- Localised corrosion occurrence conditions for Type 304 stainless steel used in the lining of the spent fuel pool 1F were revealed.
- Uniform corrosion conditions and self-passivating conditions for carbon steel used in the primary containment vessel was suggested.
- A corrosion progress prediction model for the carbon steel pipe was suggested.

IRID, (Hitachi GE, Toshiba and JAEA) and CRIEPI conduct experimental studies to evaluate effects of seawater on the thermal-hydraulic behaviour in fuel bundles and debris beds. IRID, (Hitachi GE, Toshiba and JAEA) has been conducting corrosion tests and investigated actual fuel in an environment that simulates real storage conditions. Through these tests, the structural integrity of fuel assembly components during long-term storage stored in the pool has been being assessed.

Assuming there has been a corrosive effect caused by the injection of seawater, the integrity of equipment and structures in various plant conditions has been being evaluated in preparation for fuel debris retrieval. IRID/(Hitachi GE, Toshiba, Mitsubishi Heavy Industry and JAEA) is aiming to establish a life extension scheme, which includes countermeasures for corrosion control, etc., as well as predicting the long-term integrity of structures based on seismic evaluation of equipment and the structures and on corrosion test results for each material used.

IRID/JAEA performed the Tests for (U,Zr,O) solid solution prepared as simulated debris by reaction with seawater on high temperature. Major results of them were as follows:

- Magnesium oxide crystal was deposited on the surface of fuel debris by decomposition of seawater salt under high temperature.
- The results under high oxygen partial pressure: Dense layer of $(Ca,Na)UO_{4-x}$ or $CaNaU_2O_{7-x}$ was formed on the surface of U simulated debris.

Therefore, uranium can be eluted from uranite salt, while it is immersed in water. Phase of fuel debris changed to orthorhombic crystal by oxidation. U_3O_8 was formed under high U/Zr ratio. 3) The results under low oxygen partial pressure: Ca (and Mg) was diffused to the surface of cubic crystalline U fuel debris. Small fragments (micro metre order) of (Ca-U-O) compound were dispersed to MgO deposited layer, by generating and decomposing uranite while elevated temperature process [4].

JAEA, in collaboration with Tokyo Univ. and Osaka Prefecture Univ., conducts experimental and analytical studies of radiolysis and steel corrosion in diluted artificial seawater under gamma-ray irradiation. Hydrogen peroxide is known that it is produced by radiolysis of water under irradiation and then it accelerates steel corrosion. JAEA studied the radiolysis of water containing species derived from seawater such as Cl and Br ions, and the effect on steel corrosion. From the results of radiolysis study, it was predicted that the species like Cl and Br ions, especially Br ion, promotes production of hydrogen peroxide under irradiation [7]. And as the results of steel corrosion tests under gamma-ray irradiation, it was confirmed that steel corrosion accelerated in the solution including Br ion than Cl ion [8]. Additional information gather during decommissioning will provide invaluable data to validate these understandings and to better accident management planning.

A.2.1.6 References

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A.2.2 *Mission Time and System Survivability*

A.2.2.1 Cable and Sealing

A.2.2.1.1 Background

The survivability of containment leak-tightness following a severe reactor accident is an important issue during the first days of the accident and also for several months and years after immediate accident management measures for stabilisation of core melt and containment pressure and temperature. The radioactivity causes a major concern during severe accidents. The exposure of various materials and components to radiation, elevated temperatures and chemicals may have a long-term effect on organic materials in the containment, including penetration seals and cables.

The rate of dissolution of uranium into the containment water after a severe accident is important to know. The alfa-active radionuclides will increase the alfa-dose in water and also enhance the hydrogen and oxygen production by radiolysis. The slow process of dissolution of uranium into groundwater in the bedrock has been the interest of safety case building within the nuclear waste management discipline for years under subcooled conditions [1],[2]. The rate of dissolution increases with temperature and in the presence of oxygen. The chemicals in the water pools may be radioactive contaminants on one hand but may also be chemically reactive with organic materials in in the sealants, cables and coatings. The solubility of actinides in general into the water pools under severe accident conditions is thus of interest. Also, the dissolution of species from steel and other structural materials to coolant pools should be investigated, since dissolved fission products and chemical additives from severe accident management measures (such as pool pH control) can affect cables and coatings.

One important aspect is the leak-tightness of various organic materials under the conditions following a core-melt accident. The containment temperatures are likely to reach 100 °C or more. Fission products are settled on the containment structures or migrated into the water pool by scrubbing effects or along with the steam condensate flow. The key contributors to radiation dose are iodine and Cs-137, Cs-136 and Cs-134 during the first weeks following the accident and Cs after several months or years from the accident.

Hessheimer and Dameron [3] have summarised the results of the extensive containment integrity research programme performed by the NRC. The research project addressed the performance of the containment penetration seals. Aged and un-aged gaskets of ethylene propylene (EPDM), silicone and neoprene were tested at SNL and the Idaho National Engineering Laboratory (INEL). The gaskets were tested up to pressures of (0.98 to 1.10 MPa (143 to 160 psig)) and temperatures up to 370°C (700°F). The test indicated that failure, i.e. leakage, was independent of ageing and that the silicone seals failed at lower temperatures in steam than in air or nitrogen, while EPDM was not affected by the pressurisation medium.

Electrical penetration assemblies are used to provide a leak-tight feed-through into the containment building for power, control and instrumentation cables. Electrical penetrations, in addition to serving as a closure, in most cases will have to function 'electrically' during a severe accident and this function must also be evaluated. Three different commercial penetration designs were tested at SNL at pressures of 0.51 to 1.07 MPa (75 to 155 psia) and temperature from 180 to 370°C (360°F to 700°F) for up to 10 days. None of the penetrations assemblies tested failed. The authors of the test report, however, cautioned against assuming that for all electrical penetration.

A full size personnel airlock, originally fabricated for the cancelled Callaway Unit 2, was tested by the CBI Research Corporation under contract to SNL. Double EPDM gaskets provided the seal between each door and the bulkhead. The airlock was designed for an external pressure of 0.41 MPa (60 psig) and a maximum temperature of 170°C (340°F). It was tested to a pressure of 2.07 MPa (300 psig) and a temperature of 204°C (400°F) without failure. The inner door leaked when subjected to a pressure of 0.07 MPa (10 psig) at 343°C (650°F), although the outer door did not leak. Again, the authors of the test report [4] cautioned against extrapolating the results of this test to all airlocks, since designs can vary significantly.

Lee [5] has investigated the radiation effects to nine different elastomer samples including EPDM, Nitrile (NBR), silicone, neoprene Viton and different urethanes. The samples were irradiated with gamma rays. He observed that sulphur cured urethane has the best radiation resistance since it maintained elongation properties i.e. stiffness and damping and had tensile strength after 1000 Mrads and little change in hardness. But the compression set of sulphur cured urethane is poor. EPDM is reported to have the best all-around properties up to gamma dose of 500 Mrads. The hardness of EPDM increases but the compression set is good up to 500 Mrads. With higher doses, the EPDM samples became brittle and disintegrated. Other material became brittle already at 300 Mrads. Silicon and Viton became hard already at 50 Mrads and brittle at 300 Mrads. Urethane had the highest outgassing rate.

Placek et al [6] investigated the degradation process of EPDM. They observed that the dominating degradation process is the oxidation. Also, the ethylene content affects ageing characteristics. According to Placek et.al., the key property for evaluating ageing is the compression set. Their studies focused on simultaneous thermal and radiation in normal and DBA post-LOCA conditions. The service temperature of the sealant was defined to be 80°C and a dose rate of 1 Gy/h was applied. Typical seal compression during service is 25 % describing a situation where the duty thickness of the seal is reduced by 25 % in respect to thickness without stress. The test samples of the EPDM sealant were aged at 110°C with the irradiation of 1 Gy/h or sequentially with irradiation forts at 22°C and the thermal ageing afterwards. The LOCA conditions were incorporated into the testing by exposing the previously irradiated samples (to 90 kGy cumulative dose) with the additional dose rate of 2 Gy/h and steam atmosphere of 150°C. The spray solution of water, hydrazine and boric acid was also applied during the steam exposure phase. Ultimately, the samples were submerged into the spray solution after 9 hour exposure to steam atmosphere for time of 3 days. The tests revealed that the initial thickness of the EPDM seal had an effect to compression set results. A thicker seal maintained their sealing properties better, i.e. thicker sealant need lower compression to maintain leak-tightness. Also the thicker seals relaxed better after compression. They also report that the simultaneous thermal and radiation ageing is more severe than sequential radiation and thermal ageing.

Le Ley [7] has investigated ageing of EPDM under thermal loads alone and in case of initial exposed to thermal loads following irradiation with a gamma source. The EPDM cable jackets that were aged in normal duty and operating temperature of 45°C did not show any dramatic decay even after 32 years of thermal exposure. A test series was carried out with first temperature exposure to the samples at 120°C for 20 days and then the exposure to gamma radiation of 80 Gy/h was applied to two sample sets till the cumulative dose of 10 kGy and to 20 kGy, respectively was reached. Further, two more test sets were conducted with initial thermal ageing of 20 days in 20°C following a first irradiation till dose of 10 kGy and 20 kGy with dose rate of 80 Gy/h and then a second exposure to a cumulative dose rate of 400kGy with a lower dose rate of 5.6Gy/h. His conclusion of the thermal and radiation ageing was that it caused hardening of EPDM. With an irradiation dose of 400 kGy the compression force increased by 130%.

Bellows are used primarily in steel containment penetrations. One type of commercial bellows is used in BWRs and PWRs in process piping with diameter ranging from 152.4 mm to 1524 mm. A larger bellows is used in vent lines in BWR Mark I with the size ranging from 1651 mm to 3175 mm. Sandia has performed a series of 13 tests for nine different bellows geometries studying the survivability in extreme temperature, pressure and stress conditions [8]. In the tests with only pressure loading the leak tightness was maintained even with the bellows fully extended to the cylinder. In the test set with compression load at room temperature, the tested bellows maintained leak-tightness even at full compression. A leak developed only if a lateral displacement was applied to a fully compressed bellows. When the temperature of the bellows was increased to 218°C, the full compression and applied lateral displacement did not cause a leak. The leak was obtained when applying two rounds of compression and extension. Lambert and Parks observed that bellows with shallower convolutions were more load resistant than the ones with twice as deep convolutions. The deeper convolutions also showed a tendency to roll over the end spools which may tear a leak. They concluded from the experimental studies that a bellows will likely develop a leak if being fully compressed followed by outward movement of the containment wall for example by hoop strain cause by inflating pressure increase or by vertical extension of containment by over pressure. Lambert and Parks conclude also that their tests were conducted with new bellows components and these tests did not address the effects of aged bellows with potential corrosion. Also, these tests did not address the effects of torsional loading.

Bartonicek et al [9] have investigated the ageing effects of electric cables. They suggest that oxidation is the main ageing process for polymers. Oxidation results mainly in embrittlement of polymeric materials. They have performed measurement of the effects for different cable jacket materials including EPR/EVA (ethylene vinyl acetate), polyethene (PE), and polydiene (PD and cross-linked polyethylene (XPE). Several different commercial specimens were tested. The main contributors to oxidation are the temperature load and radiation load in a nuclear power plant. Bartonicek et al. report that, based on their experiments, the radiation ageing followed by the thermal ageing is more severe than the thermal ageing followed by the radiation ageing to most of the polymer materials except for PE. For PE, the thermal ageing followed by irradiation reduced the elasticity of material sample significantly more than the loading case with irradiation performed before the thermal loading. They further noticed that the EVA test samples that were aged under radiation load only showed a trend that lower dose rate of 0.1 kGy/h resulted in more than 3 times deeper oxidation layer in the sample surfaces than with higher dose rate of 1.2 kGy/h during similar exposure time. The authors conclude that for each jacket material, the sequences of ageing effects need to be first investigated for determination of most limiting conditions for qualification of a cable material.

A.2.2.1.2 Safety Research Interest

The safety research interest in this area is high.

- **Elastomer seals**

The effects of ageing on elastomers are observed as extrusion, chemical transformations, mechanical wear, tensile cracking, load relaxation and compression set. The seals and gaskets have been reasonably well studied [10]. The replacement of the damaged seals is usually easy. Challenges are to be expected on the replacement pf submerged seals, especially under severe accident conditions and radioactive contamination in the water pools, such as encountered in post-accident Fukushima Daiichi NPPs.

It is concluded in [10] that further information gaps exist related to dose rate effects and in synergies between the thermal and radiation ageing and mechanical stresses. The ageing effects of these stressors are not simply additive.

Farmer et al [11] identified organic seal degradation under BDBE conditions as a knowledge gap. They report that even though the seal material performance is reasonable well known under DBA conditions, there is much less data on the survivability of the seals exposed to elevated temperature, pressure, steam atmosphere and radiation effects. In particular, data is needed on the performance of the already aged sealants.

The status of the penetrations seals (manholes, electrical penetrations) and cables should be investigated by visual inspection and the observations reported during the decommissioning effort, but should not cause undue adverse impact. Any samples from the seal materials or electrical cables from selected locations should be collected for further characterisation and material testing. This information would be useful in evaluation of the long-term leak-tightness of the Fukushima Daiichi containment. Furthermore, this information would be useful for evaluation of the SAM features of the restarted Japanese NPPs as well as of the operating NPPs worldwide.

Research is needed for evaluation of lifetime criteria for the elastomer seals. The knowledge of organic seals in long-term submerged conditions following a severe accident needs further investigations. The experience gained from post-accident management and decommissioning for Fukushima Daiichi will provide important information about the seal performance in challenging, real severe accident conditions.

- **Cables**

Thermal embrittlement of insulation is observed to be the most significant ageing mechanism for low-voltage cables under DBA conditions. Very limited amount of data is available of cable survival in extreme BDBE situations. The main interests of international research programmes have been the lifetime prediction of cables, the definition of representative accelerated ageing tests and the development of reliable condition monitoring methods. Despite the extensive amount of research carried out internationally on organic material survivability there are no definite criteria for end of life in terms of functionality of a component and survivability of a component during DBA and BDBE situations. Severe accident management measures installed in some plants require robust and resistant instrumentation including cables. Such devices are for example hydrogen igniters with cabling and monitoring systems of various plant parameters (Temperatures, pressures, water levels and radiation levels). Also SAM requires reliable measurements and signals to the operators in the Main Control Room (MCR). Such measurements and signals are for example the core exit temperature often used for criterion to change from emergency operating procedures to SAMGs. The effects of ageing of the cables for these instruments are not yet well known. The effects of containment spray operation to the aged cables during BDBE conditions may not be sufficiently understood yet.

The development of lifetime criteria for cables serving severe accident management systems need further research as well as better knowledge on survivability of aged cables for SAM instrumentation. Also further research may be needed to better understand the effects of cable sheath deterioration caused by fire, mechanical loads or chemical attacks on performance of other severe accident management operations (igniters, PARs, closed loop coolant circulation, fission product chemistry).

The experience gained from post-accident management and decommissioning of Fukushima Daiichi will provide important information about the condition and survival of cables in challenging, real severe accident conditions. The experience gained from post-accident management and decommissioning of Fukushima Daiichi will provide important information about the seal performance in challenging, real severe accident conditions.

A.2.2.1.3 Decommissioning Interest.

As the Fukushima Daiichi accident is being the most severe accident that have occurred in western designed BWRs, the radiation levels during the accident would be of great interest. The development of dose rates and accumulated doses during the accident would be valuable when considering radiation resistance of polymeric components and their acceptance criteria during severe accident. Available radiation data should be first evaluated whether it can be considered as valid and comprehensive so it could be used as a basis for acceptance criteria.

The decommissioning interest in this area is low: From the decommissioning point of view, ensuring the containment function of the PCV boundary is crucial for planning and implementing of the fuel debris retrieval methodology. In this case, the potential of leakage from seals and eventually the containment of such leakage should be considered. However, research on seal material degradation mechanisms under SA condition and ageing after the accident is not critical for the planning of the establishment of a multi-barrier containment.

A.2.2.1.4 Potential Examinations

The visual inspections of the elastomer seals and cable jackets at various places within the containment and the reactor building would be useful in obtaining reference information of the survivability of the polymers. Also, samples could be taken and for standard elastomers, property testing if possible. The visual inspection and characterisation of the post-accident state of the bellows in the containment penetrations would be useful.

As the cooling of the damaged reactor remains key priority during post-accident management, the reliability of the cooling pumps becomes more important. Testing of the o-rings used in the cooling pumps would provide data on the performance of o-rings used in cooling pumps during severe accident. Tightness of the o-ring is the most important function of the component which would mean that a suitable test method for the o-rings would be compression set. Since radiation is known to oxidise polymer components, DSC (Differential scanning calorimetry) can be used to evaluate remaining amount of antioxidants within the studied o-rings and thus evaluate the remaining lifetime of the o-rings. Both test methods require a reference sample to compare that has not been exposed to severe accident conditions.

A.2.2.1.5 Ongoing R&D Activities

Various national research projects are ongoing in different countries on the topics of seal materials and cable survivability during severe accidents. As an example a Nordic research projects is presented in the next.

A joint Nordic project on Condition Monitoring, thermal and Radiation Degradation of polymers inside NPP containments (COMRADE) has been developed as part of the Finnish nuclear safety programme SAFIR2018, following an initiative from the Nordic NPPs through Energiforsk. The project development has been made by SP Technical Research Institute of Sweden and at VTT Technical Research Centre of Finland Ltd.

COMRADE is developed based on input from a feasibility study from Energiforsk AB [12], and the ongoing study by the Finnish radiation authority (STUK). The project was motivated by the gaps in knowledge for setting functional based acceptance criteria of polymer components at the nuclear power plants and a need to gain a better understanding on how a polymeric component reacts to different levels of low dose radiation and synergistic effects between thermo-oxidative and irradiation degradation. The work is carried out in three work packages:

- WP 1 focusing on method development of condition monitoring and implementation at NPPs
- WP 2 is a pre-study to map how the closed down plant Barsebäck can be used to verify the method developed in WP1
- WP3 focusing on polymer ageing mechanism and effects inside the NPP containment

The aim of WP1 is to identify the acceptance criteria for the function of the polymeric component. This includes developing robust test methods that can be used by the power plants for condition monitoring through a material property, the correlation of the material property to the function of the component, performing experimental tests to validate the method, development of a theoretical model that can be used to calculate acceptance criteria for components with different geometries and deployment of the results into the daily operations at the NPPs.

The aim of WP2 is to study materials from Barsebäck NPP that have undergone ageing during operation for many years. This includes a pre-study to identify the polymeric components that can be available to study, analysis of the degradation of the selected materials and a workshop to present and discuss the results.

In WP3 the effects of radiation and heat on polymer degradation are evaluated. This includes a study in a form of a literature survey and experimental work. The effect of oxidation depth to mechanical properties of polymer components under radiation is also evaluated as well as dose rate effect and the methods that can be used in extrapolating the dose rate effects from experimental data to lower dose rates.

The goal of the project is to provide information to the nuclear power plants and the radiation safety authorities on functional based acceptance criteria, to see how Barsebäck can be used to verify models for acceptance criterion, to gain better understanding in synergistic effects between heat and irradiation and see how different levels of low dose irradiation will affect the polymeric component. The work will be done in co-operation between VTT, SP Technical Research Institute of Sweden, the Swedish and Finnish Nuclear Industry through Energiforsk and a manufacturer of nuclear grade elastomers.

A.2.2.1.6 References

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A.2.2.2 Instrumentation

A.2.2.2.1 Background

During the Fukushima Daiichi accident, numerous instrumentation measurements were unavailable as a result of loss of AC and DC power. Further, harsh BDBE conditions, including high temperature, high pressure, high radiation, water spray, and flooding, impacted the performance of instrumentation systems, which include sensing elements, reference legs, leads, cables, electronics, transmitters, signal processors, etc. After power was restored, inaccurate and inconsistent values and/or trends across instruments measuring the same parameters were observed. As a result, operators struggled to discern whether significant changes were taking place and to integrate data from different sources to verify instrumentation readings. In this way the accident at Fukushima Daiichi illustrated the importance of reliable instruments (or other means of assessment of reactor status) in BDBE conditions.

Within the US, efforts were undertaken by the DOE, industry groups, and the NRC to address this issue. US DOE sponsored LWR instrumentation survivability evaluations [1], [2], [3], which focused on determining the critical parameters needed for severe accident management and mitigation, the instrumentation required to provide data for these parameters, characterising the environment that instrumentation monitoring this data would have to survive, and identifying gaps in existing instrumentation capability.

These initial studies evaluated BWR and PWR pilot plants. Because of the availability of severe accident analysis information from simulations performed in support of the US NRC-sponsored State of the Art Consequence Assessment (SOARCA) programme [4], [5], [6] the pilot plants used for this evaluation were the Peach Bottom Atomic Power Station BWR and the Surry Power Station PWR. For risk-important sequences identified in the SOARCA studies, critical instrumentation needs were identified based on plant-specific accident management procedures and discussions with plant operators. MELCOR simulation results obtained from the SOARCA studies were then used to quantify the environment to which critical instrumentation systems may be subjected to during these events. Then, MELCOR results and instrumentation qualification ranges were compared to assess instrumentation survivability. These pilot plant evaluations were completed in 2015.

A Severe Accident - Instrumentation & Monitoring Systems (SA-Keisou) programme was implemented in Japan to develop instrumentation and monitoring systems that could prevent the escalation of an event similar to the accident that occurred at Fukushima Daiichi [7]. SA Keisou emphasised the need to monitor important variables such as reactor water level, reactor pressure, and hydrogen concentration, that operators can use to prevent an event from escalating into a severe accident, mitigate the consequences of a severe accident, achieve a safe state for the plant, and confirm the plant continues to be in a safe state over the long term. The SA-Keisou programme addressed BWR and PWR instrumentation needs and included representatives from electric power companies, vendors, and instrumentation manufacturers. The SA Keisou programme was completed in April 2015.

During 2015, the US DOE sponsored several efforts to gain consensus among US experts related to gaps in our knowledge about severe accident progression [8] and information needs from forensics examinations of the damaged reactors at Fukushima Daiichi [9]. The US expert panel, which included representatives from national laboratories, universities, and industry, identified knowledge about instrumentation performance as an important gap in our knowledge about severe accidents.

Similar to what occurred after the accident at Three Mile Island Unit 2 (TMI-2) [10], the US expert panel observed that the affected Fukushima Daiichi units offer the international community a unique means to obtain prototypic severe accident data from multiple full-scale BWRs. To maximise the benefit from information obtained from the affected reactors at Fukushima Daiichi during decommissioning, US experts developed a list of prioritised time-sequenced examination information needs and supporting R&D activities that could be completed with minimal disruption of the decommissioning plans for Fukushima Daiichi. In particular, to address the knowledge gap related to instrumentation performance, the US expert panel recommended that components be examined at all the units at Fukushima Daiichi to provide important data for recently-initiated efforts to develop and validate new severe accident models. In addition, such examination data could aid in improved plant staff training, revised severe accident management guidance and potential plant equipment enhancements; i.e. mitigation strategies.

A.2.2.2.2 Safety Research Interest

The **safety research interest is high**. TMI-2 and the events at Fukushima Daiichi demonstrate the importance of having reliable post-accident instrumentation that provides operators critical information for diagnosing the condition of the plant and the impact of mitigating actions during BDBEs. The performance of critical instrumentation under BDBE conditions, such as occurred at Fukushima Daiichi, is not well understood in part because environmental qualification testing does not include testing to identify margin to failure. Further, the environmental conditions of various instrument locations during various BDBE scenarios are not well known. Better understanding of the condition of key instruments at Fukushima Daiichi could inform guidance for compensatory strategies, such as relying on alternate instruments or indicators when the performance of a key instrument is suspect and improve severe accident management capabilities. Examination and a representative sampling of instrumentation and cabling from the affected reactors at Fukushima Daiichi could provide important insights related to instrumentation survivability and characterisation of instrumentation degradation and failure modes so instrumentation degradation and failure can be better understood, predicted, and managed. It could also provide insights on the potential effects that are thought to have affected measurements taken during the Fukushima accident. This knowledge could thus improve the accident data for accident progression analysis and code assessment.

A.2.2.2.3 Decommissioning Interest

There is **low interest** for research of instrumentation as far as decommissioning strategy is considered, since instruments are replaced or not continued to be used.

A.2.2.2.4 Potential Examinations

An US DOE expert panel identified instrumentation as an area related to BDBEs in which knowledge gaps are known to exist. This panel concluded that research efforts currently underway by industry, NRC, DOE, and the international community could address the gaps and concluded that the DOE need take no additional action beyond monitoring these existing activities. Still, this panel identified consensus information needs for examinations at the affected reactors at Daiichi that could be helpful to these various efforts [11]. The SEG on SAREF has reviewed these results, and suggests the following examinations[1]:

- Visual/photographic examination, selected sampling and selected operability assessments of in-vessel and ex-vessel sensors, support structures and associated cables.

1. These instruments are considered by the SEG of SAREF as distinct from the electrical systems because of their size, location and function and as such the SEG of SAREF considers examination and selected sampling to be worthwhile.

Some of these components may be readily accessible, but others may be difficult to evaluate. In some cases, examinations may require near-proximity examinations, examination of support structures, cables, etc. and sampling/removal to examination facilities. Operators should take a graded approach, determining what components to visually examine, and what components, if any, to sample and subject to operability assessments based on the ease of examinations, the instrument category, location and noticed trends or other issues of interest. Information available related to the various instruments cables and support structures should be collected and categorised first to help guide these efforts. Some examinations could likely be done in the near term, but some are likely to be longer-terms activities.

A.2.2.2.5 Ongoing R&D Activities

The US NRC is taking steps to address severe accident instrumentation survivability. These efforts stem from post Fukushima Daiichi guidance in Section 4.2 of the Near Term Task Force (NTTF) report. A post-Fukushima action item (Identifier SECY-12-0025, Enclosure 2) was established to address this instrumentation survivability concern and to evaluate the regulatory basis for requiring reactor and containment instrumentation to be enhanced to withstand severe accident conditions. NRC staff is reviewing information from previous and ongoing severe accident management research efforts and is monitoring results of the DOE pilot plant evaluations and international research activities. NRC is participating in IAEA activities examining severe accident instrument needs and survivability. The US Nuclear Industry has proposed existing instrumentation for use in the transition from design basis event actions to beyond design basis actions, and development of alternative means for determining plant conditions or alternate courses of action when instrumentation becomes unreliable in the severe accident management strategies. The NRC staff is evaluating the industry's position and provided its initial assessment to the Commission for its evaluation.

In response to NRC Orders EA-12-049 [12] and EA-12-051 [13], the US Nuclear Energy Institute developed guidance for mitigation of certain beyond design basis accidents similar to the Fukushima Daiichi accident (See [14], [15], and [16]). This guidance is known as FLEX and includes both additional equipment to assure continued core, containment and spent fuel cooling during an extended loss of AC power as well as guidelines for the appropriate use of this equipment [17]. [2] The industry response also included development of guidance for assuring that reliable instrumentation indications were available for key instrumentation that would be used in decision-making by the licensed plant operators under these beyond design basis conditions. This guidance includes:

- Implement spent fuel pool wide-range level instrumentation,
- Provide freeze protection for critical instrumentation,
- Strategies to circulate and cool air in containment compartments to prevent any adverse impact on critical instrumentation,
- Strategies to circulate air in key rooms in the auxiliary building to prevent any adverse impact on power supplies and/or critical instrumentation,
- A strategy to deploy portable generators and cables to directly re-establish power to the power supplies in select cabinets thereby re-powering the instrumentation loops, and

2. New FLEX Support Guidelines (FSGs) are a subset of the emergency operating procedures for use in certain Beyond Design Basis (BDB) conditions to provide alternate strategies for core, containment, and spent fuel cooling. FSG-7, "Loss of Vital Instrumentation or Control Power,"51 which provides actions to establish alternate monitoring and control capabilities, has the objective to ensure that operators have access to accurate data for critical parameters.

- A strategy to utilise handheld instruments to tap into the instrument loops locally to monitor essential parameters.

While these recent enhancements are directed towards the initial (e.g. "pre-core" damage) phases of an event, they also provide enhanced instrument availability and an alternate means of obtaining key parameter values if the event progresses to a severe accident.

Both the US PWR Owners Group (PWROG) and the BWR Owners Group are developing enhanced post-Fukushima Daiichi generic SAMGs with Technical Support Guidelines (TSGs) on instrumentation behaviour [18], [19]. In these enhanced SAMGs, instrumentation indications are used to determine challenges to plant fission product boundaries, to identify and prioritise needed actions, and to determine whether implemented actions are successful. Correct interpretation of signals from instrumentation is fundamental to the successful diagnosis, control, and mitigation of a severe accident. Since severe accidents are beyond the design basis of the plant, conditions may be more extreme than ranges for which the instrumentation was designed or calibrated. Several key factors that are being considered in these owner group evaluations include:

- Instrumentation typically relied upon for a DBA may not be available (e.g. power supplies, isolation valves, etc.) during a severe accident,
- The instrumentation range may not be adequate during a severe accident,
- Use of instrumentation may challenge fission product boundaries (e.g. hydrogen analyser), and
- The magnitude of the environment (pressure, temperature, radiation, etc.), as well as the time at which elevated conditions are present, in comparison to the equipment qualification basis may lead to erroneous readings.

The PWROG recommends that instrumentation indications be validated by an independent means if possible. The enhanced PWROG SAMGs include TSGs with guidance for determining the validity of the information being provided by the plant instrumentation. The instrumentation provides the SAMG user with additional information that can be used to determine the validity of the instrumentation indications. This guidance is knowledge-based and relies on comparing instrumentation indications with other key information including: alternate instrumentation for the same parameter, assessment of other related or linked parameters (such as pressure and temperature), other indications not directly provided by instrumentation, calculational aids, and expectations for trending of plant parameters based on the accident progression. Guidance is to be provided for all key parameters needed for effective severe accident management using the new, enhanced PWROG SAMGs. BWR Owners Group activities to develop TSGs are currently focused on obtaining insights from detailed evaluations of available TEPCO instrumentation data from Fukushima Daiichi Units 1, 2, and 3 and include an assessment of how differences between indicated and actual values may have influenced actions taken at Fukushima Daiichi accident. Results are being used to develop principles for validating instrument indications received during an accident.

As part of their post-Fukushima Daiichi activities, EPRI formed a Technical Advisory Group (TAG) to address Instrumentation and Control (I&C) for beyond design basis and severe accidents [20], [21]. The purpose of the TAG, which consists of representatives from the Institute for Nuclear Power Operations (INPO), EPRI, PWR and BWR Owners Groups, NRC, and DOE, is to promote collaboration and co-ordination in:

- Addressing the lessons learnt from the events in Japan about the required durability and capabilities of I&C systems during severe accident events.

- Identifying the required parameters and ability of reactor and containment I&C systems to withstand severe accident conditions.
- Performing research to determine if the availability of I&C can be improved so that plant data are not lost during beyond DBAs.

Hence, the TAG's role is primarily to facilitate exchange of information and research results.

There have also been recent international activities in this area.

The IAEA established an Action Plan on Nuclear Safety in response to the Fukushima Daiichi events. One of the action items of this plan was to provide guidance to Member States on "Post-accident and severe accident monitoring systems." [22] was prepared in response to this action item to reflect current knowledge, experience and best practices in this area and is based on the results of a series of meetings. This reference provides a common international technical basis to consider when establishing new criteria for accident monitoring instrumentation to support operation under design basis and severe accident conditions in new and existing nuclear power plants. It considers monitoring instrumentation and the associated instrumentation support systems for accident prevention and mitigation. The monitoring systems support on-site staff in making decisions for the management of DBAs and severe accidents.

France has launched two national projects devoted to severe accident instrumentation: DECA-PF on the diagnosis of the status of a nuclear reactor core from the measurement of fission products and Distributed Sensing for Corium Monitoring and Safety (DISCOMS) on the monitoring of corium by distributed optical fibre sensors and self-powered neutron/gamma detectors [23].

NRA is conducting research on survivability of aged cables under SA. The cable types employed in this study are those which are used as safety-related low voltage cables subject to environmental qualification requirement. The cables are aged by simultaneous thermal and radiation conditions to simulate the ageing given under normal operating condition through their service life. Following this, they are exposed to assumed SA environmental condition. This research will be completed in March 2017, and will provide recommendation for long term integrity evaluation test of cable considering SA condition.

Another research is on Assessment of Electrical Cable Condition Monitoring Techniques. It is to identify cable condition monitoring (CM) techniques which can track the ageing trend of safety-related low voltage cables. Electrical CM techniques currently in scope are Broad band Impedance Spectroscopy (BIS), Line Resonance Analysis (LIRA) and Pulsed Electro-acoustic Method (PEA). NRA evaluates capabilities of these CM methods for verifying the actual applicability and develops a long-term integrity evaluation method.

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A.2.2.3 Reactor Core Isolation Cooling Systems (RCIC)

A.2.2.3.1 Background

There is much that isn't known about the end-state of Units 1, 2, and 3 at the Fukushima Daiichi Nuclear Power Station. Some of this uncertainty can be attributed to lack of information related to Reactor Core Isolation Cooling (RCIC) system operation. The RCIC system is the key safety system for many BWRs. It is used to remove decay heat from the reactor under a wide-range of conditions ranging from operational pressures down to lower pressures approaching cold shutdown conditions. The RCIC system uses steam produced by water boiling from the reactor core decay heat to drive a steam turbine which in turn powers a pump to inject water back into the core and maintain the needed water inventory for long-term core cooling. Steam flow is drawn off directly from the boiling water in the core upstream of the safety relieve valves (SRVs) and the main steam isolation valves (MSIVs), powering the turbine-pump system, injecting water from the condensate storage tank (CST) or from the BWR wetwell.

Based on what is currently known about the events at Fukushima Daiichi [1], it is believed that RCIC system operation was critical in delaying core damage for days (almost three days for Fukushima Daiichi Unit 2) even though the turbine-pump system ran without DC power for valve control and with high water temperatures from the BWR wetwell. The RCIC system apparently operated in a self-regulating mode supplying water to the core and maintaining core-cooling until it eventually failed at about 72 hours. This suggests that there is significant margin in RCIC system performance that has been neither quantified nor qualified and is not taken into account in safety analysis. In this way the accident at Fukushima Daiichi illustrated the importance of reliable safety systems, such as RCIC, in BDBE conditions.

This issue is also relevant to Turbine-Driven Auxiliary Feedwater (TDAFW) systems, which have a similar configuration and perform a similar function in many Pressurised Water Reactors (PWRs). Like the RCIC system, the TDAFW system uses steam produced by water boiling from the reactor core decay heat to drive a steam turbine which in turn powers a pump to inject water into PWR steam generators to maintain the needed water inventory for long-term core cooling. Further, the TDAFW pump also provides a means to reduce pressure in the RCS, thereby reducing any inventory losses and prolonging the time to core damage, particularly for station blackout (SBO) or Extended Loss of AC Power (ELAP) events. The importance of the TDAFW pump has increased in recent years with the installation (or planned installation) of low leakage Reactor Coolant Pump (RCP) seals in most PWRs. If core damage occurs due to Reactor Coolant System (RCS) inventory loses, the TDAFW pump also has a high importance in preventing fission product releases from the plant in that it keeps the steam generator tubes submerged and protects them from high temperature creep rupture failures. For extreme external events, the TDAFW system can also extend the time at which containment venting might be required. As such, increased knowledge of the performance of RCIC systems is applicable to TDAFW systems and PWR emergency response systems.

The panel sponsored by the US DOE to gain consensus among US experts on gaps in our knowledge about severe accident progression [2] and information needs from forensics examinations of the damaged reactors at Fukushima Daiichi [3] identified knowledge about RCIC system performance as an important gap in our knowledge about severe accidents. It was recommended that new severe accident models be developed with a testing programme to validate such models under Beyond Design Basis Event (BDBE) conditions. The ultimate objective of this effort was to quantify the increased margin associated with the performance of RCIC and TDAFW systems.

Similar to what occurred after the accident at TMI-2 [4], the US expert panel observed that the affected Fukushima Daiichi units offer the international community a unique means to obtain prototypic severe accident data from multiple full-scale BWRs. To maximise the benefit from information obtained from the affected reactors at Fukushima

Daiichi during decontamination and decommissioning, US experts developed a list of prioritised time-sequenced examination information needs and supporting research and development (R&D) activities that could be completed with minimal disruption of TEPCO decontamination and decommissioning plans for Fukushima Daiichi. In particular, to address the knowledge gap related to RCIC system performance, the US expert panel recommended that components be examined within units at Fukushima Daiichi and at Fukushima Daiini to provide important data for recently-initiated efforts to develop and validate new severe accident models. In addition, such examination data could aid in improved plant staff training, revised severe accident management guidance, and potential plant equipment enhancements; i.e. mitigation strategies.

A.2.2.3.2 Safety Research Interest

The **safety research interest is high**. Probabilistic Risk Assessments indicate that the risk-dominant BDBE accident sequences with ELAP would involve RCIC system operation for BWRs and TDAFW system operation for PWRs. Thus, extended performance of RCIC and TDAFW systems under BDBE conditions is important to overall plant safety in terms of reducing both the likelihood and the consequences of core damage events involving ELAP. There is significant margin in RCIC and TDAFW systems that has neither been quantified nor qualified. Technically, this is a highly important lesson learnt from the Fukushima Daiichi accident that should be explored and quantified for the benefit of operating fleets.

Furthermore, quantifying the effects of emergency response equipment performance under BDBE conditions involving ELAP could provide estimates of the safety margin added with the safety-enhancing strategy proposed by the US nuclear industry strategy – known as "diverse and flexible mitigation capability" or "FLEX" for accident mitigation. These FLEX measures are currently being implemented in the US for PWR and BWR designs and are under consideration by other countries. Data from Fukushima Daiini suggest this is a longer-term (>15-16 hours) equipment performance issue. Equipment implemented at this plant successfully worked for extended time periods, mitigating the accident progression and allowing the units to successfully reach cold shutdown [5]. In the US, industry plans [6] for implementing BDBE mitigating strategies to address EA-12-049 relies on the use of portable systems to provide core cooling (BWR) and secondary side makeup (PWR). A better understanding of the performance of RCIC and TDAFW systems could form the technical basis to inform emergency mitigation strategies (i.e. available time) for use of the portable equipment. In particular, any information related to extending the time/conditions under which these systems will continue to operate will provide additional margin to potentially time critical actions related to both core damage prevention and mitigation. This is recognised as a very important area for further research by US industry as well as international organisations.

The principal objective of R&D in this area would be to reduce knowledge gaps on emergency response equipment performance under BDBE conditions for both BWRs and PWRs; specifically, RCIC and TDAFW systems. In effect, there is a need to determine the actual operating envelope of these components under BDBE conditions in order to expand the time margin before transition to other systems is needed. In addition, the evaluations should focus on quantifying performance under a range of conditions and defining operating regimes where these pumps will no longer be able to supply core (for RCIC) or steam generator (for TDAFW) cooling. The evaluations would further focus on identifying any potential down sides to extending operation, such as development of RCIC leak paths that could drain down the BWR suppression pool. As the RCIC systems in Unit 2 and Unit 3 operated extensively under BDBE conditions, examination of these systems could provide empirical evidence of the operational capabilities of these systems in similar and possibly other BDBE conditions.

A.2.2.3.3 Decommissioning Interest

There is **low interest** for RCIC research as far as decommissioning strategy is considered, since RCIC is no longer required.

A.2.2.3.4 Potential Examinations

A US DOE expert panel identified RCIC performance as an area related to BDBE in which knowledge gaps are known to exist. The DOE is conducting analysis and considering a full scale test programme to address these gaps. The DOE panel identified consensus information needs that could be helpful to this effort. The SAREF group has reviewed these results and suggests the following examinations:

- Photos/videos [7] of condition of RCIC valve and pump before drain down and after disassembly at affected units at Fukushima Daiichi (Unit 1 & 2).

Other examinations that might be considered as a part of this effort if these RCIC systems are disassembled include:

- Check for evidence of damage to the RCIC turbine caused by extended operation in the self-limiting mode
- Determine RCIC failure mode(s) for RCIC on Unit 2

The RCIC Systems at Fukushima Daiini were also subjected to accident conditions and information on RCIC status at Daiini could potentially be more easily obtained at a lower cost and with less radiation exposure to personnel. As such, the SAREF group suggests operators consider examination of the Fukushima Daiini RCIC units through:

- Complete forensic disassembly of the RCIC turbine and pump with measurements of clearances and visual conditions via pictures. This could include:
  - Inspection of RCIC pump impeller/wearing rings for cavitation related damage
  - Oil sampling from RCIC turbines operated under event conditions or sample results if sampling and analysis is already completed
  - Pump mechanical seal condition

If these detailed examinations of the RCIC systems at Fukushima Daiini are implemented, operators should document RCIC operational history during the event to include; length of time RCIC operated during post event response, trips/restarts of RCIC, actions taken to preserve RCIC operation, local operation of RCIC performed during event response, and temperature history of RCIC suction during post-ELAP operation

Operators should take a graded approach determining what examinations to implement based on ease of access and issues of interest.

A.2.2.3.5 Ongoing R&D Activities

The US DOE is currently supporting efforts that include researchers at Texas A&M University (TAMU) and at Sandia National Laboratory (SNL) to develop a thermo-mechanical analytical model of the steam-driven RCIC system operation with mechanistic accounting of liquid water carryover and pump performance degradation. This model is targeted for implementation in system level severe accident codes, such as MELCOR [8] and MAAP [9] to increase the fidelity to which these tools can analyse beyond-design basis events (e.g. BDBEs involving ELAP). Effects of operator actions will also be included. Initially, the Fukushima Daiichi Unit 2 accident reconstruction will be used as the basis for benchmarking this model. A second key objective of this task is to use insights developed from RCIC model application as a technical basis for developing a RCIC testing programme that would obtain data on RCIC operation under ELAP conditions.

The US DOE is also considering a possible collaborative RCIC test programme to help determine the RCIC (or TDAFW) operational envelope for BDBE operational conditions. This full-scale test data would enable operational system improvements to support extended RCIC operation in a BDBE, even with a loss of DC control. Numerous stakeholders have been involved with discussions of this collaborative effort including the US DOE, the Electric Power Research Institute (EPRI), the BWR Owners Group, the PWR Owners Group (PWROG) and Japan. However, as of this point, no decision on a test programme has been made.

The US NRC has supported a scaled, experimental facility with the major components of the RCIC system at TAMU. The experiments focused on the temperature distributions in the suppression pool, which impact RCIC pump inlet temperatures and the available net positive suction head, as well as the pressure transient in the containment. Modifications to the RCIC system were suggested to help address the competing concerns of containment pressurisation and pump cavitation.

In Japan, METI/IAE is conducting a research programme to promote a better understanding of the operating characteristics of RCIC in severe accidents and constructing analytical models. Specifically the thermal hydraulic behaviour in the RCIC piping of Unit 2 is being evaluated with computer codes. In addition, another research plan to determine the operating characteristics of RCIC in severe accidents and to develop analytical models is under development. Japan and the US DOE are co-ordinating their respective RCIC programmes.

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A.2.2.4 Relief Valves and Piping

A.2.2.4.1 Background

Primary system Safety Relief Valves (SRVs) are essential for controlling RPV pressure in accident management procedures for BWRs. After reactor trip, high capacity SRVs (or Pilot Operated Relief Valves in PWRs) are used to reduce primary system pressure and remove decay heat. In addition, SRVs are used by BWR automatic depressurisation systems. These valves provide overpressure protection for the reactor vessel and associated piping systems. Current uncertainties with respect to valve operation involve reliability under extended cycling at high temperature conditions associated with gases flowing through the valve as well as the high temperature and pressure conditions inside containment during protracted BDBE scenarios such as those experienced at Fukushima Daiichi.

In the SOARCA Peach Bottom study [1], the effect of this high temperature effluent on the material strength of the steam line piping was estimated; it was shown that if Reactor Pressure Vessel (RPV) effluent temperatures exceed 1 200 K, MSL rupture is very likely at the ~1100 psi SRV cycling pressures. The SOARCA study also considered the thermal degradation of the SRV valve itself through thermal distortion of valve stem clearances and the potential for material strain that could lead to SRV seizure in some intermediate position between 100% open to 100% closed. Sensitivity studies suggest that if the SRV were to seize open in any position less than a 50% open fraction, then the RPV pressure could not be relieved fast enough to avoid MSL rupture.

During the Fukushima Daiichi Unit 3 accident, the cycling of the SRV was observed to stop as shown in RPV pressure strip chart data [2]. Thereafter, the RPV was dramatically depressurised. This rate of RPV depressurisation is consistent with a possible break in the Main Steam Line (MSL) due to high temperature loss of strength, which could have been caused by high temperature gases flowing from the RPV to the suppression pool. This represents a major bifurcation point in plausible accident progression. Depending on the nature of SRV failure, the MSL could retain its function with the consequent transfer of RPV inventory to the suppression pool and retention of much of the fission products in the suppression pool water or the MSL could fail with consequent transfer of RPV inventory directly to the drywell with no suppression pool scrubbing. In this way the accident at Fukushima Daiichi illustrates the importance of reliable safety components, such as SRVs, in BDBE conditions.

The panel sponsored by the US DOE to gain consensus among US experts on gaps in our knowledge about severe accident progression [3] and information needs from forensics examinations of the damaged reactors at Fukushima Daiichi [4] identified knowledge about RCIC system performance as an important gap in our knowledge about severe accidents.

Similar to what occurred after the accident at TMI-2 [5], the US expert panel observed that the affected Fukushima Daiichi units offer the international community a unique means to obtain prototypic severe accident data from multiple full-scale BWRs. To maximise the benefit from information obtained from the affected reactors at Fukushima Daiichi during decommissioning, US experts developed a list of prioritised time-sequenced examination information needs and supporting R&D activities that could be completed with minimal disruption of the decommissioning plans for Fukushima Daiichi. In particular, to address the knowledge gap related to SRV performance, the US expert panel recommended that components be examined at all the units at Fukushima Daiichi to provide important data for recently-initiated efforts to develop and validate new severe accident models. In addition, such examination data could aid in improved plant staff training, revised severe accident management guidance and potential plant equipment enhancements; i.e. mitigation strategies.

A.2.2.4.2 Safety Research Interest

The **safety research interest is high**. Under beyond design basis conditions (i.e. severe accident conditions) there are no SRV performance data other than what may eventually be revealed from the Fukushima Daiichi accident decommissioning activities. Data from the Fukushima Daiichi accident indicate that protracted SRV cycling took place in all three reactors and under the added duress of extreme temperatures cause by core degradation processes. Reducing uncertainties in relief valve and associated piping performance under BDBE conditions would provide a better understanding of accident progression. New models could be developed and validated with appropriate test data to characterise relief valve and associated piping performance under BDBE conditions. Post-accident examination information could also improve our understanding of the accident progression, support development of more effective accident management and response procedures, and identify potential plant system enhancements.

The principal objective of R&D in this area would be to reduce knowledge gaps on emergency response equipment performance under BDBE conditions. In effect, there is value in determining the actual operating envelope of these components under BDBE conditions in order to better understand and predict accident progression. As noted above, SRVs are essential components for controlling RPV pressure during BWR severe accidents. Thus, data on the reliability of these components under extended BDBE conditions are important for reducing modelling uncertainties related to severe accident progression, as well as enhancing accident management planning. Similarly, data on the reliability of performance of certain associated piping under BDBE conditions could reduce severe accident progression modelling uncertainty and enhance accident management planning.

The principal knowledge gap for these components relates to their reliability (i.e. failure rate as well as failure mode) under adverse conditions associated with seismic, flooding, high radiation, high temperature and pressure experienced at Fukushima Daiichi. Information obtained from Fukushima Daiichi examinations could provide critical information about the performance of these components when subjected to these conditions.

A.2.2.4.3 Decommissioning Interest

There is **low interest** for Relief Valves and piping research as far as decommissioning strategy is considered.

A.2.2.4.4 Potential Examinations

A US expert panel identified SRV performance as an area related to BDBE in which knowledge gaps are known to exist. DOE may consider future SRV analysis and testing to address these gaps. This expert panel identified consensus information needs that could be helpful to future consideration of this issue. The SAREF group has reviewed these results, and suggests the following examinations:

- Photos/videos of main steam lines and Automatic Depressurisation System lines to end of SRV tailpipes, including instrument lines
- Visual inspections of SRVs, including standpipes (interior valve mechanisms)
- Photos/videos of TIP tubes and SRV/intermediate range monitor tubes outside the RPV
- Photos/videos, probe inspections, and sample exams of Main Steam Lines (MSLs); Interior examinations of MSLs at external locations

A.2.2.4.5 Ongoing R&D Activities

Testing to reduce knowledge gaps related to SRV performance under BDBE conditions may be possible in a facility similar to the type that would be used to test RCIC as noted above. However, no decision has yet been made in this regard.

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A.2.2.5 Environmental Degradation of Metallic Components

The aim of this chapter is to identify potential research areas of interest using metallic materials such as RPV, connected pipelines or containment components that can be harvested from Fukushima Daiichi NPPs during the dismantling decommissioning process.

A.2.2.5.1 Background

Environmental degradation mechanisms of metallic systems, components and structures (SCCs) are a big concern for ageing management of operating nuclear power plants. This aspect becomes more and more important as many plants are reaching 40 years of operation and they are getting ready for Long Term Operation beyond the 40 years. It should be remembered that Fukushima Daiichi unit 1 (1F1) was almost 40 years old at the time of the accident.

Effective life management requires sufficient knowledge of the life expectancy of each important SSC. This can be achieved through a systematic approach that takes into account aspects such as: i) better knowledge of initial conditions, ii) better knowledge of operating conditions and iii) better knowledge of degradation mechanisms.

Concerning degradation mechanisms, the objective is to anticipate degradation issues of SCC that may evolve during the life of the NPP. The nuclear community has already identified several degradation mechanisms, such as corrosion fatigue, embrittlement, and stress corrosion cracking (on its different ways and conditions). For certain operating conditions or operating time, also degradation mechanisms such as creep, thermal fatigue or void swelling should be taken into consideration.

The areas of research concerning degradation mechanisms of metallic materials are already identified in many roadmaps worldwide, and there is nothing new to be added (see references hereinafter). Any of these roadmaps could serve as a reference to address the work needed, as all of them take into account the degradation mechanisms that take part in an NPP in operation and the materials to which these degradation mechanisms apply.

The opportunity here is that there may be additional information that can be obtained from Fukushima Daiichi which is unique due to the conditions these materials experienced, conditions that can happen only in accident conditions such as those of Fukushima Daiichi. This is the added value that may be obtained. With this in mind, and recognising the difficulties of separating out short-term (accident related) and long-term (degradation related) effects, it is proposed that for any materials harvested from metallic components, consideration be given to evaluating their microstructures and material properties. The information gained may be valuable for predicting the behaviour of aged metallic components under accident conditions.

A.2.2.5.2 Safety Research Interest

The accident at Fukushima Daiichi NPPs has exposed SCCs to conditions not achievable in normal operation or in labs; therefore **the safety research interest is medium**.

One way of monitoring SCCs degradation is to correlate the evolution of microstructure and material damage with applied loadings and conditions and this is particularly useful in the case of infrequent transients.

In this sense, some of the materials coming from the decommissioning process of especially the Fukushima Daiichi reactor and connected pipelines as well as containment components could be of interest for the international community, as inFukushima Daiichi the materials were exposed to conditions that cannot easily be reproduced. The study of the characteristics of the materials under these accidental conditions could help in the understanding of the effects of the degradation mechanisms expected in operating

NPPs to some extend and to gain some insights into the response of aged structural materials under accident conditions.

A.2.2.5.3 Decommissioning Interest

This area of research focused on environmental degradation mechanism has little or no **interest concerning decommissioning** of Fukushima Daiichi, while maintaining structural integrity of SSCs is important (see chapter A.3.1).

A.2.2.5.4 Potential Examinations

The information to be gained from Fukushima Daiichi would be in the way of making direct observations and harvesting material of those SCCs subject to the most adverse accident conditions to be tested in labs. This will serve to make a determination of the environmental degradation of these SCCs and the effect of the degradation on their subsequent behaviour. The following steps are proposed:

- Identify the SCCs more likely to be investigated. To do so a good understanding of the accident progression is needed. Materials from the reactor vessel, the reactor coolant system, the torus, etc. could be among the eligible ones, pending a more thorough interdisciplinary analysis.
- Estimating the conditions to which these materials were submitted. These conditions include time at temperature, irradiation levels (fluence), pressure, external loads, chemistry, etc. This is a very challenging task, but an estimate of the expected degradation should be possible.
- With the materials identified according to a) under the conditions of b), the response of the materials to the predicted accident conditions (temperature and loading transients) can be assessed.

A.2.2.5.5 Ongoing R&D Activities

To deal with the knowledge gaps of degradation mechanisms of metallic materials, the international community is performing surveillance on a regular basis and also research projects are undergoing. The results of these R&D projects are incorporated in the regulation and consequently in the life management policies on each country.

These R&D projects are being performed using mock-ups that try to reproduce the materials and the operating conditions of the SCCs under investigation. But this aspect is not always easily achievable, because there is no archive material and it is difficult to reproduce the same fabrication process, for instance, so a lot of uncertainties are always present. That's why, in recent times, the international community is focusing on getting materials coming from already decommissioned NPPs, as the best way to determine their characteristics and behaviour under the period of time and conditions they have been in operation.

In this sense, international projects like ZIRP (Zorita Internals Research Project), that is using materials (SS 304) harvested from the internals of the reactor of this Spanish NPP with an operation time of 26.5 EFPY, and a fluence level in these internals of 50 dpa or more, are of paramount importance for a better knowledge about the behaviour of the characteristics of this material under these operating conditions. The results of this R&D project will be used, no doubt, for a better approach on the life management programmes of those countries and institutions that participate on it.

There are at present other NPPs that have been decommissioned and that could be a very important source of materials to be harvested and tested to determine the characteristics of the materials under the operating conditions of such NPPs.

There is an IAEA initiative, in form of a Coordinated Research Project, with the aim of promoting research projects on materials coming from decommissioned NPPs. This Coordinated Research Project is under a definition process, promoted by IAEA in conjunction with Japanese utilities and some other countries or institutions have already joined this initiative.

The opportunity here is that there may be additional information that can be obtained from Fukushima Daiichi which is unique due to the conditions these materials experienced, conditions that can happen only in accident conditions such as those of Fukushima Daiichi.

A.2.2.5.6 References

These article was prepared taken into account ideas, expressions, and approaches from the following documents:

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[3] Light Water Reactors Sustainability Program. Integrated Program Plan. U.S Department of Energy. Office of Nuclear Energy. January 2012

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[5] DOE-NE Light Water Reactor Sustainability Program and EPRI Long-Term Operations Program – Joint Research and Development Plan. April 2014

[6] Strategic Plan for Light Water Reactor Research and Development. Idaho National Laboratory/Nuclear Power Industry. INL/EXT-07-13543. November 2007

[7] Safety Aspects of Long Term Operation of Water Moderated Reactors. IAEA-EBP-SALTO. July 2007

[8] Environmentally-Assisted Degradation of Structural Materials in Water Cooled Nuclear Reactors. Advanced Nuclear Technology International. 2006

A.3 Recovery Phase

A.3.1 *Long-term AM and Recovery*

A.3.1.1 Background

Once a severe accident occurred in an NPP, site management and recovery activities are required for long periods of time. In the Fukushima Daiichi NPP accident, Units 1, 2 and 3 experienced severe core damages, and the reactor buildings of Units 1, 3 and 4 were destroyed by hydrogen explosions. As a result, many structures, systems and components (SSCs) were damaged and the cooling capability was lost. In addition, those damages resulted in the long-term processing and storage of continuously discharged contaminated water and execution of recovery activities under a severely contaminated environment with highly radioactive materials.

By referring to the status of the Fukushima Daiichi NPP accident, topics on long-term accident management (AM) and recovery are summarised below with the relevant information necessary from the viewpoints of ensuring countermeasures against contaminated water, understanding of plant status, and maintaining structural integrity of the plant.

A.3.1.1.1 Countermeasures for contaminated water

After a severe accident, damaged nuclear fuel assemblies have to be continuously cooled until they are removed from the reactors. Processing of a large amount of contaminated water would be conducted for a long term in cases that damaged nuclear fuel assemblies cannot be retrieved for a short period, the plant is severely damaged, and contaminated water is mixed with groundwater, as seen in the Fukushima Daiichi NPP accident. The following information is required for management and processing of the contaminated water.

- Leakage and migration path of cooling water, mixing of water leaked from reactor building and groundwater, and leakage path of contaminated water into the environment
- Management and processing of contaminated water generated before and during fuel debris retrieval
- Maintaining structural integrity of Cs adsorption vessel during storage after the contaminated water processing (countermeasures against heat generation, hydrogen generation and corrosion)
- Detailed analysis of contaminated water within PCV and other locations to specify the sources of contamination and to evaluate the status of reactor core.

No experience exists on taking countermeasures against massive amounts of contaminated water after reactor accidents accompanied by destruction of facilities.

In Japan, research on long-term safety management of secondary radioactive wastes generated during processing of contaminated water in terms of heat generation, hydrogen generation, and corrosion in adsorption vessels and others are in progress, using virgin vessels, non-radioactive materials and computer codes.

A.3.1.1.2 Understanding of plant status and maintaining of structural integrity

For the management and recovery of the plant after the accident, it is necessary to access the inside of the buildings. The following information is required for that:

- Evaluation of contamination and decontamination inside the reactor building, PCV, SSCs, etc.

- Evaluation of radiation dose from the fuel debris and radioactive materials which were released from the fuel assemblies and transported inside the vessels and buildings, and evaluation of effectiveness of radiation shielding structure (especially, when filling the water into the PCV is evaluated impossible and fuel debris is cooled by air without shielding effect of water). This kind of information becomes more important in considering the decommissioning methods.
- In connection with the above, obtaining the information regarding status of fuel debris cooling and water level inside RPV and PCV, and the information regarding distribution of radioactive materials inside the buildings

For the evaluation of the structural integrity of SSCs, the following items are important and knowledge to be obtained will be useful for decommissioning of nuclear facilities, including the Fukushima Daiichi NPPs.

- Analytical study and knowledge concerning structural integrity of RPV, PCV and cooling system (influences of high temperature and high pressure under severe accident condition, fuel debris, hydrogen explosion and corrosion environment due to radiation, boron, concrete components and seawater)

Through the course of the Fukushima Daiichi NPP accident, many kinds of materials were produced by the destruction of buildings and fell down into the spent fuel pool (SFP); furthermore, seawater was injected into SFP for emergency cooling. Therefore, careful attention should be paid for the long-term storage of spent fuel and the following information is hence required.

- Influence of changes in water quality due to seawater injection into SFP on the structural integrity of spent fuel
- Structural integrity during dry storage of spent fuel taken from SFP which experienced seawater injection and into which debris fell.

A.3.1.2 Safety Research Interest

The safety research interest for this area is medium. There exist especially some interest in how to manage contaminated water and how to assess structural integrity.

A.3.1.2.1 Countermeasures for contaminated water

Not only release of radioactive materials into the atmosphere but also leakage of contaminated water lead to release of a large amount of radioactive materials into the environment, and therefore, long-term stable management of contaminated water is important. Analytical data of contaminated water are useful to estimate the extent of damage in fuel assemblies, MCCI and FP release from fuel debris, to develop analytical methods concerning those phenomena and to increase safety level in the existing rector facilities.

A.3.1.2.2 Understanding of plant status and maintaining of structural integrity

Data related to the status of contamination in the SSCs (including distribution of radioactive materials, components of adhesives and radiation dose data) can be used not only for the safety evaluation and regulatory judgements in decommissioning processes but also for the verification and improvement of source term analysis methods. Knowledge and evaluation methods of coolability, depending on the cooling method of fuel debris, and distribution of radiation dose from fuel debris and radioactive materials inside the reactor core can be used as a basis for the safety evaluation and regulatory judgements in the decommissioning processes.

There is little knowledge on the long-term structural integrity of RPV and PCV in the heavily damaged reactor facility. This kind of knowledge obtained in the Fukushima

Daiichi NPP can be utilised for the evaluation of long-term safety level of the plant and as a basis of regulatory judgements during and after the decommissioning processes.

In case the decommissioning process is continued for decades like in the Fukushima Daiichi NPPs where seawater was injected for emergency cooling, knowledge on the long-term influence of corrosion can be used for the establishment of an evaluation model.

As another topic, painting materials on the inner wall of PCV have been exposed to a high temperature environment during the accident and will be subjected to high level radiation fields generated by the migrated FP for a long period of time, resulting in loss of the original functions. Such data on quality changes of paint materials are also rare and valuable.

There is no existing knowledge on the long-term structural integrity of spent fuel in contact with seawater and debris dropped from the broken structures of the facilities. Therefore, corresponding data could be used for related safety evaluations and regulatory judgements.

Some interest exists in how to manage contaminated water and maintain the structural integrity from the safety point of view, and data from the related investigations are available to analyse the accident progression.

A.3.1.3 Decommissioning Interest

This area is ranked high from the perspective of decommissioning interest. Results from the research activities mentioned above will contribute to implementing efficient decommissioning activities, ensuring the safety of the involved workers, reducing the generation of contaminated water, and achieving stable and efficient cooling of the fuel debris. Each of the following items is expected to contribute to the progress of decommissioning at the Fukushima Daiichi NPPs.

A.3.1.3.1 Countermeasures for contaminated water

Information on flow paths of reactor coolant, contamination source, contact situation of fuel debris with coolant, leaching of radioactive materials from the fuel assemblies can be obtained by analysing the radionuclides in the contaminated water. E.g. if the contamination level of the water after touching the fuel debris is low, the contamination level could possibly be reduced by draining the highly contaminated water stagnated in torus room once. Information on how the water leaked from the reactor building mixes with the ground water and how the contaminated water is leaked into the environment could contribute to the leakage prevention activities.

A.3.1.3.2 Understanding of plant status and maintaining of structural integrity

During decommissioning of the Fukushima Daiichi NPPs, cutting and removal of the piping systems in the reactor building and the structures within the PCV should be performed while paying careful attention to the possibility of hydrogen explosion.

In case where the PCV is flooded with water, maintaining robustness of the vessel and evaluation of structural integrity are key factors. Also, a high concentration of radiolysis products is expected in the contaminated water exposed to high radiation dose, and if parts of penetrations are in direct contact with contaminated water containing high concentration of radicals and organic materials for a long period of time, attention should be paid to the possibility of deterioration of the sealing parts and the consequent leakage.

Knowledge of the structural integrity of fuel assemblies exposed to seawater and dropped debris in SFP is useful to determine long-term storage condition in future and to establish evaluation items for safety assessment and regulatory standards.

Reducing the quantities of contaminated water and maintaining the structural integrity are needed to ensure continued safety of damaged reactors and to minimise radioactive releases. Behaviour of contaminated water is of particular interest.

A.3.1.4 Potential Examinations

At the moment, it is important to know and understand the status of structures and components, dose and temperature distributions, etc. to establish the basic strategy for decommissioning. It is also important to identify the coolant leak paths from the reactor pressure vessel (RPV) to the CV and then to the turbine building (TB) in each unit and also to better understand its mixing behaviour with highly contaminated water accumulated inside CVs and TBs. However, it is still very difficult to conduct even visual examinations because of the highly restricted accessibility, although the Japanese organisations such as TEPCO, NDF and IRID have attempted in the past and in the future will continue to attempt examinations inside RBs and CVs using robotics and other means.

Therefore, it would be a useful activity to collect, compile and analyse basic information on the status inside the RBs and CVs to be obtained by the Japanese relevant organisations and to track and update the future long-term project proposals under the framework of SAREF of CSNI. The activity also should include checking and testing the feasibility of transportation, examinations at hot laboratories and others of the various samples to be taken.

Such basic information is also indispensable to better understand and reconstruct the progression of the severe accident that actually took place.

A.3.1.5 Ongoing R&D Activities

Although samples and data obtained on-site are important, the amount of available data may be severely limited. To supplement the collected data, tests with mock-up devices, fresh fuel assemblies or simulated materials and research with analytical simulation codes could be effective.

High radiation dose inside the site, securing resources for the activities not included in the decommissioning (stabilisation) processes, increasing radiation exposure for workers and influences on the decommissioning processes are of concern. Investigations related to dose rate evaluation are also conducted.

IRID/(Hitachi GE, Toshiba and JAEA) have evaluated the integrity (corrosion resistance, etc.) of fuel assemblies removed from the spent fuel pool and in the common pool for a long-term storage. As a part of this evaluation, assemblies were retrieved from the spent fuel pool in Fukushima Daiichi NPS Unit 4, and surface observations, observations of the inner threaded portion surface in crevice areas and measurement of the thickness of the cladding oxide layer were performed. Results of surface observations showed that while a white deposit had formed on spent fuel, there were no issues with external appearance. Furthermore, corrosion was found to have not occurred on the inner surface of the threaded section, and measurements of the thickness of the cladding oxide layer showed that when compared with oxide layer thickness in existing fuel stored in the common pool, no increase in thickness had taken place.

IRID/(Hitachi GE, Toshiba, Mitsubishi Heavy Industry and JAEA) is aiming to establish a life extension scheme, which includes countermeasures for corrosion control, etc., as well as predicting the long-term integrity of structures based on seismic evaluation of equipment and the structures and on corrosion test results for each material used.

JAEA is developing the evaluation method of the most possible dose rate distribution in PCV by using information obtained from irradiation calculations of fuel assemblies and structural materials, severe accident analyses, local dose rate measurement in PCV, and so on. As a preliminary analysis, the dose rate distribution was calculated for a simple two dimensional cylindrical model of PCV by use of the particle transport Monte Carlo calculation code PHITS. As a result, the radiation sources sensitive to the dose rate of each location in the PCV were successfully estimated.

The radioactively contaminated water has been accumulated at the Fukushima Daiichi NPS. The radioactivity concentrations determined for the water were normalised with the fuel composition and divided by that for ^{137}Cs in order to investigate the behaviour of some elements with respect of water contamination. Transport of Sr was initially suppressed and gradually increased to a magnitude comparable to Cs for almost one year. Transport of ^{3}H, I and Se was greater than that of Cs. Transport of Pu, Am, Cm and Eu was considerably smaller. With respect to activation products, the extent of Co transport was more significant than that of Ni [1].

The concentration of ^{137}Cs in radioactive wastes such as used Caesium adsorption vessels and sludge generated from the Caesium adsorption devices, and the decontamination device, which have been being operated or been suspended as a part of the contaminated water treatment system in Fukushima Daiichi Nuclear Power Stations, was calculated by using analysis data of the contaminated water. The total decontamination amount of ^{137}Cs from 28 June, 2011 to 12 August, 2014 was estimated.

A.3.1.6 Reference

[1] J. Kato, Y. Meguro, “Inventory Estimation of 137Cs in Radioactive Wastes Generated from Contaminated Water Treatment System in Fukushima Daiichi Nuclear Power Station”, The second International Conference on Maintenance Science and Technology (ICMST2014), T1-4, 2–5 November, 2014, Kobe, Japan (2014).

A.3.2 *Fuel Debris and Waste Management*

In the Fukushima Daiichi NPP accident, the reactor core possibly melted down in Units 1, 2 and 3, resulting in fuel debris being relocated into the PCV located in the lower part of the RPV as well as the PCV after RPV failure and melt release. Many SSCs are contaminated due to the release of radioactive materials during the accident. Under such a circumstance, a large amount of contaminated waste was and will be generated through various decommissioning activities and adequate management of them is essential.

A.3.2.1 Background

In Units 1 through 3 of the Fukushima Daiichi NPP, it is considered that a large amount of fuel debris exists on the bottom of RPV and probably in the lower part of the PCV below the RPV. To identify the progression of the accident and distribution of fuel debris, a number of the accident progression analyses are being conducted not only by Japanese organisations including TEPCO and NRA but also by many foreign organisations and also under the international frameworks such as the NEA BSAF project taking newly obtained knowledge into consideration.

However, the actual status of fuel debris is still uncertain and it is impossible to deny the possibility that fuel debris may cause re-criticality in the case that the moderation condition unexpectedly becomes optimum due to the change in shape during the retrieval operation. Information on position, shape and chemical composition of fuel debris is required for implementing decommissioning activities as well as for the verification of detailed accident progression.

For the management of fuel debris, the US and other countries have experience and data from the TMI-2 accident recovery effort, and these are very useful for the decommissioning activities in the Fukushima Daiichi NPPs. Therefore, a review is underway in Japan with the support of the US. Also in Japan, tests with simulated debris and TMI-2 debris are being conducted to study characteristics of fuel debris, for example focusing on mechanical properties which are useful for retrieval tool development. Tests on the chemical composition of fuel debris and miscibility among materials, thermodynamic analysis and criticality experiments with debris-type test section are underway or under planning. Furthermore, studies on container, transportation and short-term and longer-term storage of damaged fuel assemblies and fuel debris and evaluation analysis of debris coolability, etc. are underway. The US DOE is currently planning to hold a workshop to transfer knowledge learned from TMI-2 examinations as part of an effort to assist Japan in their R&D efforts.

Radioactive wastes generated during the Fukushima Daiichi NPP accident and the decommissioning processes are totally different from those generated from normal decommissioning of NPPs in terms of concentration of radioactive materials, shape, constituent materials, volume, chemical composition and form, etc.

A.3.2.2 Safety Research Interest

The safety research interest for this area is low. However, some importance is attached to the safety of handling fuel debris and radioactive waste as identified below.

A.3.2.2.1 Management of fuel debris

Analysis of the accident progression, estimation of distribution and composition of fuel debris and assessment of the possibility of re-criticality are important for the improvement of accident analysis method, establishment and evaluation of countermeasures against accidents and safety assessment for retrieval and processing of fuel debris.

It is important for the safety assessment and the regulatory judgement on activities relating to retrieval, storage, processing of fuel debris. In addition, establishment of regulation for processing and disposal of fuel debris after retrieval is also essential.

A.3.2.2.2 Management of radioactive waste

Characterisation of radioactive wastes generated by the accident can contribute to the general understanding of characteristics of miscellaneous radioactive wastes generated by accidents, accumulation of analysis data and improvement of accuracy in characteristics evaluation, and it is also useful to develop inventory estimation method.

Investigation of concepts regarding processing and disposal of radioactive wastes generated by accidents can contribute to establishing concepts for long-term storage, processing and disposal of radioactive wastes generated by accidents and solution to confirm safety.

Interest in managing damaged fuel or components is not so high, though information specified to the Fukushima Daiichi NPPs is of interest.

A.3.2.3 Decommissioning Interest

This topic is considered of high importance from a decommissioning perspective.

A.3.2.3.1 Management of fuel debris

Estimation of the status of fuel debris and analysis of sampled materials as well as knowledge acquisition and technical development relating to retrieval through storage of fuel debris are very useful for the selecting, optimising and implementing of method, technical development and optimisation of the Fukushima Daiichi NPP decommissioning processes, as well as for ensuring safety of workers.

A.3.2.3.2 Management of radioactive waste

At present, almost all of the radioactive wastes generated in the Fukushima Daiichi NPPs are under storage on-site. Establishment of appropriate disposal methods for these wastes is very useful to optimise decommissioning processes and ensure the safety of workers in the Fukushima Daiichi NPP.

Retrieval, handling and management of fuel debris from the Fukushima Daiichi NPs are of great concerns to be solved and development of options is needed for that. Therefore, the research interest from the de-commissioning point of view is *high*.

A.3.2.4 Potential Examinations

It is indispensable to undertake appropriate preparations in order to enable fuel debris retrieval work under safe and stable condition with steadily applicable technology. With this view, it is important to implement fuel debris characterisation (form, composition and physical characteristics of fuel debris) through experimental analysis using simulated fuel debris. It is also indispensable to develop radiation dose and exposure evaluation methods during fuel debris retrieval operations such as fuel debris sampling, retrieval and management.

It is a potential international activity of NEA to share the latest knowledge on the fuel debris characterisation, and the radiation dose and exposure evaluation methods during fuel debris retrieval operations, and to discuss future projects on fuel debris analysis techniques. The outcome of this project will be useful for sharing and accumulating knowledge and expertise of fuel debris retrieval work and for further deepening understanding SA progression for each member countries.

A.3.2.5 Ongoing R&D Activities

IRID/(JAEA and Toshiba, Hitachi GE and Mitsubishi Heavy Industries) has been conducting research on characterisation of fuel debris in order to select safety and steady method of defueling and to prepare useful tools and containers for fuel debris, using simulated debris, simulated MCCI products and TMI-2 debris. For example, characteristics data of oxide fuel debris has been being expanded by evaluation of the mechanical properties, which were hardness, elastic modulus, and fracture toughness, on various samples of $(U,Zr)O_2$ made by small scale examinations. Also, in order to evaluate the properties of surrogate material of fuel debris compared with that of actual fuel debris, preparation, observation and measurement of Vickers hardness were implemented using actual fuel debris of Three Mile Island Unit 2 (TMI-2) accident. From the view point of packaging, transportation and storage of fuel debris, the characteristics of water containing/drying, which is one of the dominant factor of hydrogen gas generation, has been being investigated.

IRID also has been developing the debris analysis technology in order to prepare it, for timely analysis when debris sample will be obtained. For elements analysis of fuel debris known as sparingly soluble material, an alkali fusion method has been being examined. In that connection, IRID has been conducting feasibility study on transporting debris samples from Fukushima Daiichi site to hot laboratories such as JAEA's one in Japan.

IRID/(Hitachi GE, Toshiba, Mitsubishi Heavy Industries and JAEA) is in charge of the R&D program for the criticality safety issues of Fukushima Daiichi debris. IRID has been conducting research on soluble and insoluble neutron absorbers for control of criticality during water filling in the reactor or debris retrieval work. Regarding soluble neutron absorbers, corrosion testing using boric acid was performed. As for insoluble neutron absorbers, irradiation test was performed to evaluate the radiation-resistance of the materials. And also IRID has been conducting research on a new detection method of re-criticality and sub-criticality during water filling in the reactor or debris retrieval work and a method to stop the re-criticality such as using neutron absorber or water dumping. One of the important re-criticality detection systems is "Gas sampling fission product gamma radiation detector system" and an improved system will be applied as a demonstration test on the Fukushima Daiichi site near future.

JAEA has its own R&D project entrusted by NRA for criticality safety study for Fukushima Daiichi, including the criticality experiments using Static Experiment Critical Facility (STACY). From the viewpoint of safety regulation, criticality control of the fuel debris in the Fukushima Daiichi NPPs (1FNPS) is a risk-informed control to mitigate consequences of criticality events, instead of a deterministic control to prevent such events. The NRA of Japan has set up a research and development programme to tackle this challenge. A significant difference between situations in the three units in Fukushima Daiichi NPPs and the TMI-2 is that cooling water for fuel debris in the Fukushima Daiichi reactors cannot be poisoned continuously. Hence, the programme consists of three activities: Computation of basic criticality characteristics of fuel debris, development of a risk assessment method using information on fuel debris and reactor conditions, and critical experiments to validate the computation using the modified STACY. The activities of the first year in FY2014 produced computation results on criticality characteristics of MCCI products, concept design of the criticality risk assessment, and basic design of the modified STACY and its auxiliary facilities necessary to conduct criticality experiments using the pseudo fuel debris samples.

Recently, the method of removal of fuel debris retrieval in the air has been considered as a way that does not use the water as a coolant for debris. Since this method does not need to seal the PCV, contaminated water isn't generated. However, the method has problems for evaluation of fuel debris air cooling performance, for example, effect of fuel debris distribution, debris shape, debris property, and so on. Thus, evaluation method by considering various phenomena regarding fuel debris air cooling performance will be needed. To evaluate the fuel debris air cooling performance considering effects of various phenomena, a numerical simulation based on the computational fluid dynamics (CFD) is useful. Therefore, JAEA has started a research project to evaluate the fuel debris air cooling performance by using the CFD technique.

In this research project, a numerical simulation method has been developed and experimental database will be constructed based on thermal-hydraulic experimental results [1].

A.3.2.6 References

[1] K. Tonoike, et al., "Study on Criticality Control of Fuel Debris by Japan Atomic Energy Agency to Support Nuclear Regulation Authority of Japan," Proc. of International Conference on Nuclear Criticality Safety (ICNC2015), Charlotte, NC, United States, September 13-17, 2015, p.20 (2015).

A.4 General/Already Addressed

A.4.1 *External Effects, Multi-Unit Risk and Loss of UHS*

TheFukushima Daiichi accident of Japan in 2011 has discovered various gaps related to the current PSA (Probabilistic Safety Assessment) approach usage for plant risk assessment. This has resulted in the need for some issues to be reconsidered and/or implemented in the PSA application and state of practice. Examples include the assessment of extreme external events (including the combined external hazards), site-wide risks (including multiple units and spent fuel pools) and extended accident scenarios (including loss of ultimate heat sink), implying, for instance, consideration for extended mission times and the development of full scope PSA.

A.4.1.1 Background

A.4.1.1.1 Multi-Unit risk issue

The multi-unit risk issue is a very important one, especially in some countries as Canada, France, Korea and Japan where there can be from 4 to 6 units per site. The Fukushima Daiichi Nuclear Power Station is a six-unit facility. Hydrogen explosions occurred in multiple units (Units 1, 3 and 4) and the operating units (Units 1, 2 and 3) affected Unit 4, which was defueled at the time of the accident: it appears that the hydrogen in Unit 4 reactor building came from Unit 3 through an unexpected path. The hydrogen explosion of Unit 1 affected the recovery activities for Unit 2. In addition core damage occurred in Units 1, 2 and 3.

Concomitant reactor accidents at a site have been ignored in most of the current PSAs, because they were performed with the assumption that the event leading to core damage can only occur in one reactor at a time. Following the Fukushima Daiichi accident, however, the issue of site risk is spreading over all the multi-unit sites, composed of two or more operating reactors.

After the Fukushima Daiichi accident, there are concerns on the risk of SFP (Spent Fuel Pool). So, the terminology “site risk” is used when the risk source such as SFP, dry storage cask, etc. are considered in addition to the multi-unit risk.

A.4.1.1.2 External events

Some standards (e.g. IAEA) set forth requirements for external event PSAs used to support risk-informed decision-makings for commercial NPPs, and prescribe a method for applying these requirements for specific applications. External events covered within these standards include both natural and man-made external events, despite the fact that, actually, in many cases some extreme external hazards had been screened-out due to the low probability of occurrence estimated by deterministic and probabilistic hazard analysis in and nearby the plant sites. Up to now only seismic risk has been considered as an extreme external event, in addition to flooding PSA in a few cases.

However, the tsunami caused by the massive earthquake facilitated the station blackout condition and subsequent core damage and gross containment failure in the Fukushima Daiichi site. Hence the analysis points out the need of the tsunami and seismic PSA development and more in general PSA enlargement to take account of all external hazards. This is consistent with the requisite of full scope PSA development for NPPs that includes all initiating events and all hazards (external and internal) and all plant states.

Indeed, since the Fukushima Daiichi accident in which a catastrophic earthquake was followed by a tsunami greater than anticipated for the design basis flooding analysis, extreme external events have emerged as significant risk contributors to NPPs. This accident shows that extreme external events have the potential to simultaneously affect redundant and diverse safety systems, and thereby induce common cause failure or common cause initiators.

A.4.1.1.3 Loss of ultimate heat sink

During extended accident scenarios, several passive systems enable a prolonged grace period to the operator during which the reactor is maintained in a safe state without any intervention. This, in essence, implies availability of a large heat sink within the reactor building, and its highly reliable uninterruptible thermal communication with the reactor core to facilitate continued removal of core heat for prolonged durations without any involvement of active systems or operator interventions (e.g. natural convection, radiation, and conduction cooling). This feature too, is highly relevant for some extreme external events when, on account of possible devastation outside the protected reactor building, it is quite likely that all the external sources of cooling water, electricity, and instrumentation air and ventilation system become non-available. In such scenarios, it is also conceivable that the operators may not be in a position to act in an efficient or effective manner.

However, the passive residual heat removal systems are only available to function if there is a place to reject the heat, known as the ultimate heat sink. The ultimate heat sink may consist of tanks, which are designed to have enough water inventories to allow reactor cooling for a minimum of 72 hours after shutdown. Since the heat sink is not unlimited, tank refilling is required so that additional surrogate measures are to be adopted as operations with mobile equipment as well as accident management strategies and emergency operating procedures implemented.

A.4.1.2 Safety Research Interest

The overall safety research interest is medium.

A.4.1.2.1 Multi-Unit risk issue

Research is to be performed with the main goal of development of site risk assessment methodology and models, including the extremely complex multi-unit accidents and development of site-risk profile, based on all power modes, all hazards, including the extreme risk factors.

There are a variety of initiating events such as certain loss of off-site power events, loss of service water events, and seismic events that lead to concurrent event sequences on two or more reactor units on a site. The probability of multiple concurrent reactor accidents is significantly influenced by the use of shared and dependent systems, as well as common cause failures in redundant systems at the multi-unit sites. There are several key inter-unit dependencies at an NPP which are likely to be found to influence the development of an integral risk statement: some important examples are the electric power systems and the service water supply systems.

It is expected that multi-unit accident sequences make a significant contribution to multi-unit risk in comparison with the linear combination of single reactor accidents at each unit and therefore cannot be dismissed. This issue may become even more relevant with some proposed Small Module Reactor (SMR) designs in which a large number of small power reactors will be planned to operate on the same site.

A.4.1.2.2 External events

External events can occur as single initiating events or as multiple external events that are a combination of two or more external coincident external occurring more or less simultaneously. In addition, they have the potential to result in an internal initiating event.

This emphasises as relevant another issue related to unanticipated scenarios concerning correlated hazards. They include for example, combined external hazards, such as the earthquake and tsunami in Fukushima, and events hazards causing internal events, such as seismic induced fire and flood.

Therefore, the simplifying assumptions of independence have to be avoided and implemented with appropriate models suitable to describe the correlation mechanisms,

Events that are part of a multiple external event are usually dependent in some way, meaning that the probability of the included events occurring simultaneously is higher than the product of the random probability of the included events.

This conclusion allows the implementation of the initiating event quantification, by properly capturing the interaction between the single frequencies characteristics of the various events: the frequency assessment of correlated hazards should take into account all the available information (i.e. site-specific, regional, worldwide), as well as the model uncertainties.

A.4.1.2.3 Loss of ultimate heat sink

Loss of ultimate heat sink draws attention to some relevant items, related to the sequence modelling.

Fukushima Daiichi accident progress over time and long duration scenario justifies the consideration for the duration of mission times for safety systems and components longer than 24 hours in a realistic way and, more generally for prolonged mission times to a very large extent (up to one month for instance), as it has been demonstrated that 24 hours recovery concept as for internal initiating events is not good for some external events.

These considerations referred to Level 1 PSA apply as well to Level 2 PSA, which demands consideration for longer mission times, especially with regard to the residual heat removal issue, the assessment of the containment performance and the analysis of time-related aspects of the accident that influence the severe accident progression and containment loading.

Therefore, to address these aspects, dynamic PSA approaches could be investigated to extend the conventional PSA to longer periods of time, taking into account the various phases of the accident progression, in an effort to capture the integrated response of the systems/components during an accident scenario.

Performance of passive safety systems, with main focus on the ones devoted to heat removal will represent a new challenge owing to the amount of uncertainties, e.g. the condensation and boiling heat transfer coefficients or the heat transfer coefficients under the presence of non-condensable gases. Consequently, difficulties arise to achieve a qualified reliability figure, since the scarcity of data and the little experience.

Due to the specificities of passive systems that utilise natural circulation (small driving force, large uncertainties in their performance, lack of data…), there is a strong need for the development and demonstration of consistent methodologies and approaches for evaluating their reliability. This is a crucial issue to be resolved for their extensive use in future nuclear power plants.

A.4.1.3 Decommissioning Interest

There is **low interest** for this area of research in nuclear safety as far as decommissioning is considered.

A.4.1.4 Potential Examinations

For the seismic or Tsunami PSA, the operating state and/or failure mode of important safety systems during the Fukushima Daiichi accident would be very useful. However, the Fukushima Daiichi accident was mainly caused by the station black out and ultimate heat sink failure due to the earthquake and external flooding by the Tsunami. So the major failure modes could be identified. In addition, long time has passed after the accident, and the plant status might be changed greatly due to the harsh environment caused by the severe accident. So at the present state, it is difficult to get useful information for PSA from the site examination.

A.4.1.5 Ongoing R&D Activities

An assessment of the lessons learnt from the Fukushima Daiichi nuclear accident for implementing PSA methodology will address some foundational notions related to a number of factors, as highlighted by the event: these topics should be conveniently addressed and worked out in the context of post-Fukushima Daiichi follow-up activities, in order to implement their impact on nuclear safety.

Consequently, they should be incorporated into a well-structured frame in order to handle all possible risk contributors and their effects consistently and efficiently within one framework. This requires an international effort in order to harmonise the PSA methodology towards the implementation of the new models.

Finally, one has to note that these topics are currently being addressed internationally at the NEA level, as regards specific groups as TGNEV (Task Group on Natural External Events) and WGRISK (Working Group on Risk), within the frame of European Union supported project ASAMPSA_E [1] or discussed during workshops on specific topics [2] and at various national levels.

In particular the ASAMPSA_E project (2013-2016) is aimed at identifying gaps and implementing methods and practices of PSA in the light of Fukushima Daiichi accident, including the evaluation of external hazards and their combination, the risk associated to multi-unit site and the extended accident scenario.

As for the research activities in Korea, since Korea has similar regional condition and has NPP sites with multi-unit operated, a project for multi-unit risk assessment performed by the Korean Atomic Energy Research Institute was started in 2012. The main objective of the project is to develop the multi-unit PSA methodology by integrating various internal/external PSA models for each unit in the same NPP site. Also, total site risk profiles including most of external events are evaluated using the developed multi-unit PSA methodology.

NRA of Japan is currently developing seismic probabilistic risk assessments model for two-unit PWR site using Monte Carlo calculation code, SECOM2-DQFM, assuming seismic correlation coefficients between SSCs of different units, in addition to two-unit BWR site model which has already been developed, to indicate site CDF as well as simultaneous core damage fraction for both units. TEPCO is also performing the seismic induced multi-unit PSA risk research.

NRC of the United States is also performing a Level 3 PSA project which includes the site risk assessment.

A.4.1.6 References

[1] E. Raimond, A. Wielenberger, M. Kumar Post Fukushima Lessons learned for Probabilistic Safety Assessment, Proceedings of ESREL 2015, Zurich, Switzerland, 7-10 September 2015

[2] International Technical Workshop on Multi-Unit Probabilistic Safety Assessment (PSA), November 17-20, 2014, Ottawa, (Canada), jointly organized by CNSC and NEA/WG Risk

A.4.2 *Robustness of Electrical Systems*

A.4.2.1 Background

In July 2006, the INES 2 event which occurred at the Forsmark NPP in Sweden confirmed the safety importance of internal electrical systems of NPPs and more specifically, their sensitivity to loss of off-site power, exceptional transients, and order of events progression, latent faults, and maintenance errors.

In March 2011 the Fukushima Daiichi accident further illustrated the susceptibilities of the current power systems to external events. Consequently, improvements to the robustness and reliability of electrical systems (power sources and distribution) have been required. The Committee on Nuclear Regulatory Activities (CNRA) Senior Task Group on Fukushima Daiichi ranked the topic of robustness of electrical systems as number one (1) in the list of potential CSNI activities in the aftermath of the Fukushima Daiichi accident.

In total, more than ten (10) station blackout events have been reported to IAEA so far. More recently, two significant incidents (Byron, United States, 2012; Forsmark, Sweden, 2013) have been caused by an open phase event on the three-phase networks. Those incidents illustrate that electrical systems may exhibit complex behaviours which have not been sufficiently considered during their design.

Probabilistic Safety Assessment (PSA) has for a long time recognised the importance of electrical systems; the loss of electrical sources being the major contributor to core damage frequency. The electrical systems are unique in the safety demonstration of a plant. Firstly, the systems rely on the external power source (e.g. the grid) which is not safety classified and secondly, that power is supplied to plant equipment, irrespective of the safety class. Electrical systems are hence by nature prone to failure propagation and require a specific approach regarding defence in depth.

Lastly, the technology of electrical systems is rapidly evolving with a growing use of solid state power electronics and digital instrumentation and control components. In particular “smart” devices containing embedded software are making those systems more and more complex thus raising new safety issues.

Following the Forsmark incident, and after a first workshop in Stockholm in September 2007, the NEA launched a joint CSNI/CNRA task group in January 2008 to “examine the defence in depth of electrical systems and grid interaction with nuclear power plants”. This task group called DIDELSYS organised a workshop in Paris in May 2009 and produced a report in November 2009 [1]. The activity of this task group was then extended for a second phase (DIDELSYS II) until December 2011 leading to a workshop in May 2011 and a technical opinion paper [2].

The recommendations from DIDELSYS were used by regulatory bodies to provide guidance and direction to licensees (Belgium, Germany, United Kingdom, France…), and the recommendations have been incorporated into regulatory documents (e.g. German safety requirements and guidelines). Conclusions from DIDELSYS were included in IAEA guidance documents.

In December 2012, the CSNI approved the creation of a task group on the robustness of electrical systems (ROBELSYS) with the objective to “identify and discuss the lessons learnt from the Fukushima Daiichi accident as concerns the electrical systems and the provisions taken by various countries in terms of requirements and design in order to enhance the robustness of these electrical systems, especially as regards the protection against extreme external hazards”. This group organised a workshop in Paris in April 2014. Over a hundred participants from twenty five (25) countries attended this workshop during which thirty four (34) papers were presented. Based on the papers presented at the workshop and the panel discussions which occurred during the workshop, a comprehensive set of recommendations have been identified in the Proceedings of the workshop issued in March 2015 [3] [3].

A.4.2.2 Safety Research Interest

The safety research interest is low, except for systems identified as important for mission time and system survivability. Some investigations could be performed during the decommissioning of the Fukushima Daiichi units about the failure modes of electrical equipment due to the tsunami. Nevertheless, the general robustness of electrical systems against external events should be sought through an improved general design of systems, protective measures of equipment, better functional and geographical diversity, use of mobile equipment, and not by improving the resistance of the equipment itself. The insights provided by examinations on damaged units is hence probably of limited interest.

A.4.2.3 Decommissioning Interest

It seems that there is **low interest for this area** of research in nuclear safety as far as decommissioning is considered.

A.4.2.4 Potential Examinations

None are proposed.Samples and analyses of some electrical components or cables would be of limited value:

- It would be difficult to reconstitute the environmental conditions (temperature, pressure, dose…) that these components have undergone during the course of the accident;
- The behaviour of electrical components and cables under severe environmental conditions is very dependent upon their type and detailed characteristics and conclusions would be therefore difficult to transpose to other components and cables;
- Dedicated experiments to assess the behaviour of electrical components and cables against severe environmental conditions can be established with reasonable effort.

A.4.2.5 Ongoing R&D Activities

The activities planned within the CSNI Working Group on Electrical Power Systems (WGELEC) are the following:

Objective 1: Enhancement of the robustness of electrical systems

The robustness of the electrical systems has been challenged by recent events, identifying several areas of vulnerability. The following topics have been identified as areas for further study:

- Approaches to detecting, managing and mitigating degraded power supplies, including Open Phase Condition
- Good practices for the design and qualification of AC/DC systems and portable equipment to support extended station blackout
- Approaches for isolating the safety systems from external events

3. In December 2015, the CSNI approved the creation of a new Working Group on Electrical Power Systems (WGELEC).

- Addressing the challenges to off-site power in light of the increasing contribution from renewable sources (small generators of ~MW)
- Approaches to commissioning and routine testing of electrical equipment and systems
- Approaches to increasing independence and diversity of power sources and distribution to safety systems
- Evaluate protective features to ensure adequate equipment function during emergency situations
- Approaches to minimising the propagation of grid perturbations to the NPP.

Objective 2: Development and improvement in the analysis and simulation of the behaviour of NPP's Electrical systems

The complex behaviour of electrical systems exhibited during the recent incidents involving asymmetric three-phase electrical faults and the trend to introduce more digital components performing sophisticated functions illustrate the need for computer tools to perform the safety reviews of those systems. During the ROBELSYS workshop it was identified that electrical transient studies using such computer simulation tools are being developed in several member countries but there has been little international exchange on this topic. As a result, there is benefit to sharing information on codes and software used for dynamic power flow studies.

The following topics have been identified as areas for further study:

- Identification of reliable simulation tools and methods for making better assessments of electrical faults (asymmetric three-phase ; one/two-open-phase issue);
- Best practices for the simulation of plant response to AC/DC bus failures to recognise undesirable failure modes and compensatory measures
- Validation of modelling techniques
- Development of standardised transient voltage wave forms for use in qualifying on-site electrical system components. (These wave forms could replace or supplement the present lightning and switching impulse test wave forms used.)
- Good practices in monitoring electrical parameters for diagnosis and simulation models.

Objective 3: Safety challenges related to the use of power and software-based electronics in electrical power systems.

Digital components are increasingly replacing analogue devices for control and protection of electrical systems as it becomes more and more difficult to obtain components based upon analogue technology. During the ROBELSYS workshop it was acknowledged that such digital components raise specific safety challenges and require a rigorous and time consuming functional qualification. Indeed, digital components can provide increasing functionalities, but show a higher level of complexity leading to safety challenges. Due to the more complex structure, digital components show the potential for new failure mechanisms and an increasing number of failure possibilities, including the potential for common cause failures. Additionally digital components used in electrical systems have often been developed for general industry purposes and thus not according to nuclear safety standards. This often leads to the increased use of COTS ("Commercial Off The Shelf"). Difficulties to access design documentation may require the use of an alternative safety demonstration based on an ad hoc mix of experience feedback, third party assessments, additional tests and analysis, etc. that may still lead to unanticipated events.

More specifically the following topics of interest have been identified:

- Best practices for specifying and qualifying digital electronics in response to degrading power supplies
- Sharing the operating experience on software embedded electrical equipment
- Development of test requirements for enhancing the ruggedness of analogue and digital equipment
- Understanding the vulnerabilities of smart circuit breakers or motor operated valves and software-based protection relays
- Study of the relative merits of using analogue and digital systems and their application.

A.4.2.6 References

[1] OECD/NEA, Defence in Depth of Electrical Systems and Grid Interaction, Final DIDELSYS Task Group Report, (NEA/CSNI/R(2009)10), November 2009

[2] OECD/NEA, Defence in Depth of Electrical Systems, CSNI Technical Opinion Paper No. 16, 2013, NEA No. 7070

[3] OECD/NEA, Robustness of Electrical Systems of Nuclear Power Plants in Light of the Fukushima Daiichi Accident (ROBELSYS), Workshop Proceedings, Paris, France, 1-4 April 2014, (NEA/CSNI/R(2015)4), March 2015

A.4.3 *Spent Fuel Pool*

A.4.3.1 Background

Light water reactors are equipped with an on-site storage facility for fuel elements that were unloaded from the reactor core after having reached their target burn-up and fresh fuel elements waiting to be loaded into the core during an outage. The storage facility is usually constructed as an open rectangular cavity filled with water. The fuel elements are stored vertically in racks inside this pool. Spacing and absorber materials are included such that criticality is excluded. The water level is kept several meters above the fuel to provide for radiation shielding. Decay heat is removed from the fuel by an active cooling circuit. The fuel in this pool, a full core load or more, contains a large amount of radioactivity that has to be confined by the fuel rod cladding tubes. There is no pressure boundary around the fuel in most of the NPPs as when in the reactor; the spent fuel pool (SFP) is open to the atmosphere of the fuel building or the reactor building as in the case of the Fukushima Daiichi units.

In the event of a failure of all cooling systems, the pool water would gradually heat up to the boiling point and then slowly evaporate with a rate depending on the total decay heat generated in the pool. When the fuel elements become partially uncovered, cladding heats up because the steam flow is low and not capable of removing the decay heat by convection. With the water level further down, Zircaloy-cladding material will be oxidised and hydrogen will be generated under steam environment.

Calculations show that this process takes several days to develop. If cooling cannot be restored, the fuel rods will fail. Under the extreme assumption of fast draining of the pool, such as through cracks in the pool walls, the fuel cladding could reach temperatures high enough for igniting, i.e. a Zircaloy fire.

During the slow evaporation process, steam production may not be high enough to prevent the uncovered part of the fuel assemblies to be exposed to air or a mixture of steam and air. Concerning impacts on safety, the presence of air can lead to accelerated oxidation of the Zircaloy cladding compared to that in steam, owing to the faster kinetics, while the 85% higher heat of reaction drives this process further. Air ingress is typically associated with poor heat transfer; the combined effect of these factors can give rise to an increased rate of assemblies degradation. Furthermore, the exposure of UO_2 to air at elevated temperatures can lead to increased release of some fission products, notably the highly-radiotoxic ruthenium while the effect of air is likely to further weaken the oxidised cladding as a barrier against fission product release.

A.4.3.1.1 CSNI Status Report on Spent Fuel Pool under Accident Conditions

As part of the CSNI activities motivated by the Fukushima Daiichi accident, WGAMA and the Working Group on Fuel Safety have produced a “CSNI Status Report on Spent Fuel Pool (SFP) under loss of cooling and loss of coolant accident conditions” [1]. The main objectives were: (1) to produce a brief summary of the status of Spent Fuel Pool accident and mitigation strategies to better contribute to the post-Fukushima Daiichi NPP accident decision-making process; (2) to provide a brief assessment of current experimental and analytical knowledge about loss of cooling accidents in spent fuel pool (SFP) and their associated mitigation strategies; (3) to briefly describe the strengths and weaknesses of analytical methods used in codes to predict spent fuel pool accident evolution and assess the efficiency of different cooling mechanisms for mitigation of such accident; and (4) to identify and list additional research activities required to address gaps in the understanding of relevant phenomenological processes, to identify where analytical tool deficiencies exist, and to reduce the uncertainties in this understanding.

A.4.3.1.2 Experiments with relevance to SFP cooling accidents

Separate and integral effect tests have been conducted since the 1980s to better understand the fuel behaviour and degradation under severe accident conditions. The main objective of these tests was to provide data for model development and validation of computer codes used for reactor safety analysis. A number of the tests, while not originally developed for SFP accidents, provided valuable data and insights for application to SFP accident phenomenology. For example, the international PHEBUS Fission Product programme, conducted in France, provided insights and data on the fission product release and late phase melt progression for LWRs. Most of the findings of these tests are directly applicable to accident progression in SFPs.

Another set of integral tests, suitable for model validation and application to SFP accident conditions, are QUENCH-10 and QUENCH-16, conducted in Germany. These tests not only provided an improved understanding of the oxidation phenomena, but also examined the phenomena associated with recovery and quenching of overheated fuel rods. Also the experiments and tests carried out to investigate the 2003 Paks cleaning tank incident have provided useful data.

The only integral tests specifically targeted for SFP loss of cooling accidents were conducted at Sandia National Laboratories, United States, partly within the NEA Sandia Fuel Project [2]. The main objective of the experimental work was to provide basic thermal-hydraulic data for completely uncovered and air cooled fuel assemblies for boiling and pressurised water reactors, and facilitate severe accident code validation and reduce modelling uncertainties. The accident conditions of interest for the SFP were simulated in a full-scale prototypic fashion.

The experimental programme is based on a scenario where the pool loses all water in a short time and the fuel elements are exposed to an atmosphere of steam and air. Fast heat-up and igniting is expected and the propagation of the Zr-fire is investigated. This is a scenario quite different from the events in Fukushima. The analytical activities, however, may link both scenarios. They are part of the joint research project and involve calculations with severe accident codes, such as MELCOR, ATHLET or ASTEC.

A large number of separate effect tests have been done to characterise high temperature air oxidation of various cladding materials, and further tests have been performed or are underway [3-6]. These tests provide necessary data for modelling cladding degradation and zirconium fire initiation in SFP accidents. Most of the tests were done under isothermal conditions and studied the phenomena of oxidation kinetics in air and air/steam environments, oxidation breakaway, and nitriding.

There have also been various separate effect tests to examine the fission product release characteristics of the fuel under air-rich conditions, where enhanced release of otherwise low-volatile species like ruthenium can become important. Major recent experiments of this kind include VERCORS [7] and VERDON [8], conducted in France, the VEGA programme [9] in Japan, and the CRL programme [10] at AECL where both high burn-up UO_2 fuel and $(U,Pu)O_2$ fuel were investigated.

Finally, there were also some tests conducted in Korea that were designed for the evaluation of siphon breaker performance.

These experiments are summarised in the following table, from the CSNI report [1].

Table A.16. Experiments related to SFP cooling accidents

Phenomena	Related experiments	
	Separate effects tests	Integral tests
Thermal-hydraulics	Siphon breakers: (Korean Atomic Energy Research Institute)	Boildown: QUENCH (KIT) Reflooding: QUENCH (KIT), PARAMETER (LUTCH)
Fuel behaviour, fuel assembly and rack degradation	Cladding high temperature oxidation in air: (ANL, IRSN, JAEA, KFKI AEKI, KIT) B4C interaction: Becarre (IRSN)	PHEBUS FP (IRSN) QUENCH (KIT) CODEX AIT (KFKI AEKI) PARAMETER (LUTCH)
Fission product (FP) release and transport	FP release in air-rich environment: (CEA, CRL, JAEA, ORNL, UKAEA)	PHEBUS FP (IRSN)

A.4.3.1.3 Simulation tools

Simulation tools applied to SFP accidents include computer programs developed for analysis of thermal- hydraulics, nuclear criticality, fuel rod behaviour and severe accidents. For the simulation of SFP thermal-hydraulics, CFD tools can be used in cases where 3D phenomena/regimes are important. They have the capacity to address problems at the local scale in 3D. However, SFP analyses are usually done at a larger scale, and the large simulation domain necessitates simplified modelling of the storage racks (porous medium approximation) and relatively coarse meshes in the CFD simulations. Thermal-hydraulics system codes are mostly applied for accident analysis at a large scale. System codes make use of 1D or 2D representations of the considered geometry, but they are being further developed to 3D.

Computational tools used for evaluation of the nuclear criticality safety of SFPs calculate the effective neutron multiplication factor of the SFP for any static configuration described in terms of geometry, material compositions, and extra information regarding cladding degradation, debris formation and physical state and level of the cooling water. These codes can in fact be used for both operational and accident conditions. Three types of calculation schemes are employed: a purely stochastic, a purely deterministic, and a hybrid scheme. A high level of accuracy in the results can typically be obtained by any of the schemes. The burn-up dependent fuel composition can be provided by dedicated codes, which perform an in-core fuel depletion and fission products build-up analysis.

The fuel rod behaviour during the early phase of a loss of cooling incident or accident, up to the loss of rod-like geometry, can be simulated with transient fuel behaviour codes, which simulate the thermos-mechanical phenomena and the changes in fuel pellet and cladding in detail. However, they usually lack models for cladding high temperature oxidation in air-containing environments.

Severe accident codes have been developed for reactor applications by extending existing thermal-hydraulic codes with models for simulating phenomena in the reactor core during severe accidents. These codes are also used for analyses of SFP cooling accidents, because the major phenomena in severe reactor accidents are fundamentally the same as in severe SFP cooling accidents. However, the geometry and conditions expected in SFP accidents differ from those in reactor accidents, and the applicability of models in different severe accident codes is currently being verified for SFP conditions.

A.4.3.1.4 Ability of reactor core severe accident codes to simulate SFP severe accidents

The European Severe Accidents Research Network SARNET investigated the capabilities of severe accident codes to analyse SFP accidents [11]. This investigation comprised: (1) the state of knowledge, especially in regard to phenomena related to oxidation in air of the fuel rod claddings, (2) the state of code assessments on integral tests like QUENCH or PARAMETER; tests allowing to study accidental transients of oxidation in air of fuel rod claddings, ending by reflooding; and SFP tests allowing to study the behaviour of one of several fuel assemblies for representative transients of loss of coolant SFP accident, inducing fuel claddings oxidation in air and burn propagation, and (3) the assessment of different SFP accidents with different severe accident codes for various SFP geometries, various scenarios, and various levels and reparations of the residual power on fuel assemblies.

The first two tasks clearly identified lacks in knowledge – and therefore on physical relevance of available models in severe accident codes – regarding the phenomena related to the oxidation in air or steam/air mixtures of the fuel claddings (especially the role of nitrogen in the acceleration mechanisms of cladding degradation) and on the mechanical behaviour of oxidised/nitrided claddings. Moreover, difficulties were revealed to model correctly the real 3D geometry and heterogeneity of fuel assemblies with the 2D cylindrical geometry usually applied by severe accident codes.

Concerning simulations of SFP transients, five different severe accident codes were used, namely: ASTEC, MELCOR, ATHLET-CD, ICARE/CATHARE, RELAP/SCDAPSIM. The simulations have shown the impact of modelling assumptions such as the number of nodes used to represent the fuel building, which can have strong impact on the gas flow between the different parts of the building. They also raise questions about the reliability of some results obtained with these severe accident codes, regarding in particular:

- The phenomena related to the cladding behaviour in the presence of air or a steam / air mixture, such as oxidation, nitriding and embrittlement;
- The phenomena of natural convection and boiling in the fuel building. In fact, the conclusions on the coolability of fuel assemblies can be very different depending on the calculations. Some studies show, for a loss of water transient (conducting to fast dewatering and air ingress in the fuel assemblies), that air flow is sufficient to remove the power, for other studies this conclusion depends on the air flow that could actually flow in the fuel assemblies;
- The conditions of air ingress in the assembly, according to the water depth, the assembly power, and the intensity of boiling. Some studies show that for certain conditions, during the phase of fuel assembly dewatering, the air ingress flow through the top of the assembly (counter-current of steam flow) can cool down the upper part of the fuel assembly.
- The coolability of dewatered fuel assemblies with water injections.

A.4.3.2 Safety research interest

There are seven SFPs on the Fukushima Daiichi site. The SFPs of Units 1-4 were of safety concern during the accident. The initial earthquake has not caused apparent damage to the pools. Some water may have been lost by sloshing. The exact amount of leakages is not known, but is not significant compared to the water lost by evaporation during several periods of failing heat removal by cooling systems.

Concern has been mostly with the SFP of Unit 4. This unit was in shutdown state during the accident, but the SFP contained 1 535 fuel elements, 204 of them were fresh fuel. The day after the hydrogen explosion that damaged the upper part of the reactor building, it was observed from a helicopter that the pool water level was being maintained, but it was no clear at the moment which phenomena caused the hydrogen production leading to the explosion. The lowest water level in this pool, about 1.5 m above the fuel racks, was reached around April 20, 2011. There was no evidence of bulk boiling in any of the pools; measured peak water temperatures ranged from 62 to 92 °C. Eventually, pool water cooling by the alternative cooling system was started for all SFPs, and the water temperature has thereafter been maintained below 40 °C. The results from analysing nuclides from the SFP and visual inspections have revealed that Unit 4's SFP remains nearly undamaged and that the explosion was caused by hydrogen transferred from unit 3 through common SGTS ventilation system pipeworks. The fuel has been now extracted from the SFP and seems to be mostly intact; it was probably never uncovered.

Therefore no significant findings on the behaviour of spent fuel under accident conditions seem to be derived from examinations at Fukushima Daiichi. **The safety research interest is considered as low**.

A.4.3.3 Decommissioning Interest

There is **low interest for this area** of research in nuclear safety as far as decommissioning strategy is considered. Some visual inspections of spent fuel assemblies and nuclides analyses of SFP water are necessarily to be performed in order to prepare the removal of fuel from the SFP's in Units 1-3, but it is not expected that significant damage to fuel assemblies occurred. Nevertheless, longer-term corrosion and degradation mechanisms of spent fuel removed and transported to common pool may be important to decommissioning. **Generally speaking, the decommissioning interest is considered as low**.

A.4.3.4 Potential Examinations

None are proposed.

A.4.3.5 Ongoing R&D Activities

A.4.3.5.1 In Europe

The DENOPI project, operated by IRSN and supported by the French government in the framework of post-Fukushima Daiichi activities, is devoted to the experimental study of SFPs under loss of cooling and loss of coolant accident conditions [12]. The project is divided into 3 parts:

- Two-phase convection phenomena in SFPs under loss of cooling conditions: The approach proposed in the DENOPI project is to conduct experiments on models of an SFP at reduced scale to contribute to the development and validation of two-phase flow convection models across the entire SFP.
- Physical phenomena at the scale of a fuel assembly under loss of coolant conditions: Experiments will be performed with partially uncovered fuel assemblies in order to study: (1) the conditions for air penetration into the fuel assemblies; (2) the void fraction in the fuel assemblies during boil-off, which is an important parameter in the evaluation of criticality issues; and (3) the efficiency of a water spray to cool the fuel assemblies in case of a loss of coolant accident.
- Oxidation of zirconium by an air/steam mixture: Experiments on oxidation and nitriding of zirconium alloy fuel cladding will be performed in order to better estimate the margin to runaway of these exothermal reactions, leading to the destruction of the cladding.

Separate effect test programmes are performed at Karlsruhe Institute of Technology (KIT) in order to better understand the detailed mechanisms and effects of nitrogen on the oxidation kinetics of zirconium alloy cladding. Extensive high-temperature oxidation test series have recently been run in mixed oxygen-nitrogen and steam-nitrogen atmospheres.

KIT is also planning to perform another semi-integral bundle test in the QUENCH facility, with special focus on SFP conditions, including steam-air mixtures. Such a test could be e.g. conducted in the framework of the EC-sponsored Severe Accident Facilities for European Safety Targets (SAFEST) programme.

The AIR_SFP project, launched recently in the framework of the European NUGENIA platform, is dedicated to the application of accident codes to spent fuel pools, with three main objectives:

- Improving severe accident code models to simulate air oxidation phenomena,
- Defining recommendations to the use of severe accident codes for SFP accident applications,
- Defining more precisely needs of R&D on different topics like large-scale flow convection, impact of partial dewatering or air flow on thermal runaway and fuel degradation**.**

A.4.3.5.2 In Japan

NRA has been carrying out a spray test programme for BWR spent fuel to obtain quantitative spray effects for accidental situations in SFP since 2014. The target scenarios are loss of coolant accidents (LOCAs) in SFP. Water spray is injected from a spray nozzle located above the fuel assemblies when spent fuel assemblies are uncovered fully or partially due to abnormal decrease in water level. In the tests, important knowledge of spray effects such as thermal hydraulic characteristics of liquid droplets atomisation, counter-current flow and heat transfer between fuel rods and liquid droplets/liquid film will be obtained by measurements of fuel rod temperature, liquid velocity and void fraction inside/outside spent fuel assemblies. The tests will start in 2016 after the test facility which consists of a storage tank, spent fuel assemblies (Single bundle or multi bundles), storage racks and spray injection system is fabricated.

A.4.3.5.3 At NEA

In 2016 a Phenomena Identification and Ranking Table (PIRT) exercise on Spent Fuel Pool (SFP) under loss of cooling or coolant accidents conditions will be launched under the CSNI auspices. A particular emphasis will be placed on mitigation strategies.

A.4.3.6 References

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[2] Characterisation of thermal-hydraulic and ignition phenomena in prototypic, full-length boiling water reactor spent fuel pool assemblies after a postulated complete loss-of-coolant accident Report NUREG/CR-7143, USNRC

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A.4.4 *Human Performance*

A.4.4.1 Background

The accident of the Fukushima Daiichi NPPs nuclear power plant has raised several issues in terms of the PSA technique being currently used in various countries. One of the significant issues is the performance of human operators (or operating personnel) who are faced with beyond design basis accidents resulting from extreme external hazards. In this light, a couple of research topics are worth emphasising because they are deduced not only from the phenomena of the Fukushima Daiichi accident but also from its lessons learnt.

A.4.4.2 Safety Research Interest

Three kinds of research issues could be highlighted with respect to the lessons learnt from the Fukushima Daiichi accident. The first research issue is human performance under extreme external hazards. The experience from the Fukushima Daiichi accident confirmed that the role of human operators (working in a main control room) and operating personnel (working in a local area or technical support centre) is crucial for responding an extreme external hazard. This means that the variation (or degradation) of human performance with respect to the context of an extreme external hazard should be properly understood.

The second research issue is the training needed to enhance the cognitive abilities of human operators. According to the review of major sequences, it is revealed that human operators working in the main control room intentionally turned off the Isolation Condenser (IC) system because they were not able to access critical information representing the ongoing status of the Fukushima Daiichi Unit 1. It is reasonable to expect that most of human operators who are faced with this situation will draw the similar decision. However, it is also true that the status of the Fukushima Daiichi Unit 1 could be better if human operators let the IC system go on. This strongly implies that the training of human operators (and operating personnel) of which the purpose is to develop a cognitive ability to properly respond to uncertain situations caused by insufficient information is very important, in particular:

- The use of ad hoc strategies that could be effective for bringing nuclear power plants to a safe shutdown condition during beyond design basis accidents;
- General training for enhancing effective management of the situation at hand.

The last research issue is to predict the response of human operators with respect to a given context. A more challenging issue is how to predict (or estimate) the response of human operators who are faced with various situations caused by beyond design basis accidents. It is concluded that **the overall safety research interest is medium** as far as the research supports improvements to severe accident management.

A.4.4.3 Decommissioning Interest

Since all the research issues being described in the above focused on the response of human operators exposed to extreme conditions at operating NPPs, it is not proper to reconsider them in terms of decommissioning. **So, the decommissioning interest is low.**

A.4.4.4 Potential Examinations

With respect to the research issues being suggested in the above, the catalogue of information to be gained from the Fukushima Daiichi accident could be identified. First of all, in the case of the first research issue, the effect of harsh environments on the performance of human operators should be collected. For example, at the early phase of the Fukushima Daiichi accident, operating personnel did their best to restore the electrical power struggling with a harsh environment such as a high radiation and darkness. In addition, they felt a high degree of psychological stress such as fear and anxiety for their personal safety. Moreover, due to the delay of additional resources from off-sites, operating personnel had to cope with other psychological stressors such as fatigue (i.e. continuing their work for a long time). Unfortunately, although the effect of a harsh environment caused by physical stressors (e.g. high radiation) on the performance of human operators is available from existing literatures, the effect of psychological stressors (e.g. fear and fatigue) on their performance is still insufficient. For example, it is rare to find out sufficient information with respect to the variation of the degree of fear and/or fatigue with respect to the size of an aftershock. Accordingly, it can be said that the feasibility of the information examination being required in the first research issue could be low.

In terms of the second research issue, it is necessary to collect the cognitive capabilities of human operators such as (1) capability for reasoning with missing, conflicting, and misleading information, (2) capability for reasoning under data overload conditions, and (3) capability to develop and/or implement a plan that are not fully managed by given procedures and guidelines. However, since it is very hard to gather information related to these capabilities from the Fukushima Daiichi accident, it should be said that the feasibility of the information examination is low. Although simulation experiments are helpful for collecting the associated information, it is also true that we do not have a reliable and feasible simulator that allows us to emulate the task environment of harsh conditions such as beyond design basis accidents.

Regarding to the last research issue, the responses of human operators should be gathered from the Fukushima Daiichi accident, which could be different with respect to the nature of tasks available in a given situation, such as (1) procedure-guided tasks (e.g. conducting emergency operating procedures), (2) procedure-informed tasks (e.g. severe accident management guidelines), and (3) knowledge-driven tasks (e.g. no available procedures and guidelines). In addition, as the experience of the Fukushima Daiichi accident clearly showed, the response of human operators is very sensitive to the number and structure of distributed teams that have different roles and responsibilities. However, it can be expected that the feasibility of the information examination being considered in this research issue is medium if we are able to consider the performance of human operators who are faced with similar contexts with those of the Fukushima Daiichi accident.

A.4.4.5 Ongoing R&D Activities

Regarding the first research issue, the NEA Working Group on Human and Organisational Factors (WGHOF) has worked for the last two years in order to get useful information on the performance of human operators under an extreme condition. In this light, on 24-26 Feb. 2014, the WGHOF and the Swiss Federal Nuclear Safety Inspectorate (ENSI) hosted a workshop in Brugg, Switzerland, to gain insights on human performance under extreme conditions. More detailed information on this workshop can be found from a website of the NEA (www.oecd-nea.org/general/mnb/2014/march.html), and a draft report dealing with important issues, good practices and research needs from the perspective of human, organisation and infrastructure was published in the early of 2015. Similarly, under the programme of the Halden Reactor Project (HRP), IFE (Institutt For Energiteknikk) is now conducting a research project entitled *Decision making in situations with degraded information*, of which one prime objective is to study the behaviour of human operators in decision-making and problem solving under stressful conditions with limited information and reduced resources.

In the case of the last research issue, a conceptual framework that can be used to estimate the response times of human operators who are exposed to a seismic event is suggested by Korea Atomic Research Institute (Park et al., 2015 [1]). It is worth emphasising that the collection of human performance data under various kinds of situations is one of the crucial research directions to properly understand the response of human operators being faced with extreme conditions, such as the Fukushima Daiichi accident. In this regard, it is worth emphasising that the US NRC (Nuclear Regulatory Commission) is currently sponsoring a research activity to identify the impacts of environmental conditions associated with flooding and another research to improve the manner in which human performance is modelled.

In addition, a research project was launched by Korean Institute of Nuclear Safety to strengthen the regulatory system on human and organisational factors in 2014 (Lee et al., 2015 [2, 3]).

A.4.4.6 References

[1] Park J., Kim, Y., Kim, J. H., Jung, W., and Jang S. C., 2015. Estimating the response times of human operators working in the main control room of nuclear power plants based on the context of a seismic event – A case study, Annals of Nuclear Energy, vol. 85, p. 36-46.

[2] D.H. Lee, D.S. Kim, D.J. Kim, M.Y. Kim, M.C. Kim, S.H. Kim, J.T. Kim, H. Kim, Y.S. Yoon, Y.M. Lee, Y.H. Chung, Y.S. Choi, 2015. Analysis of Technical Issues on Plant Operation and Management in the Light of Beyond Design Basis Events and Severe Accident, KINS/RR-1297 [In Korean]

[3] D.H. Lee, D.S. Kim, D.J. Kim, M.Y. Kim, M.C. Kim, S.H. Kim, J.T. Kim, H. Kim, Y.S. Yoon, Y.M. Lee, Y.H. Chung, Y.S. Choi, 2015. Analysis of Regulatory Issues on Human Factors Engineering in the Light of Beyond Design Basis Events and Severe Accident, KINS/RR-1298 [In Korean]

A.4.5 *Seismic Response (of buildings and components)*

After a nuclear accident like the one at Fukushima Daiichi site, no regulation or experience exist when addressing the following actions against seismic related challenges: evaluating structural integrity of damaged SSCs, selecting a robust method to recover fuel debris, taking into account the situation of the RPVs and/or PCVs (with or without water) and building new SSCs to process or storage contaminated water. Before starting these actions, it would be wise to confirm site seismic parameters, and topics or relevant information necessary to ensure long-term structural integrity of both, damaged and new SSCs, against seismic related events as summarised below.

A.4.5.1 Background

According to available information, the earthquake did not have a serious impact on plant safety [1]. It did not cause a loss of coolant accident (LOCA) or loss of emergency diesel generator (EDG) functions in Fukushima Unit 1, as speculated after the accident and indicated in the Report of the National Diet of Japan. It has been concluded that pipe breaks causing leakage on a scale that would have affected the development of the accident did not occur; and concerning the loss of EDG functions, recorded data shown that this malfunctions occurred immediately after the loss of function of the seawater pumps, which is considered to have been caused by the tsunami.

Despite the discussion on earthquake impact on plant safety and waiting for a forensic evaluation, after a severe accident like the one at Fukushima Daiichi, damaged nuclear fuel assemblies have to be continuously cooled until they are removed from the reactors. To do that it will be necessary to build new SSCs, and to implement decommissioning activities to design a robust methodology to remove fuel debris from the reactors.

Regardless of the Periodic Safety Review process scope, the availability of new scientific or technical findings, external operational experience, and even updated seismic re-evaluations, and from a general viewpoint, an updated comprehensive reassessment of the site characterisation has not been required by regulators. In other words: ‘back checking’ has been usually performed, instead of a comprehensive site reassessment or ‘back fitting’. If this is the case (“back checking” but not “back fitting”), then the Safety Analysis Report (SAS) will remain written in accordance with existing regulations several decades ago.

Events experienced by some Japanese NPPs (Kashiwazaki Kariha, KK 2007 and Fukushima Daiichi, FK 2011) show that, before performing a seismic analysis (Soil Structure Interaction (SSI) for instance), determination of site effect uncertainty will be needed, especially in multiunit sites. Relevant differences between design and recorded seismic levels from one site to another, and even variability between records obtained in different units in the same site, as show the records obtained at KK[4] seven units during the Niigata-Chuetsu-Oki earthquake (2007) and at Fukushima Daiiichi/Daiini[5] units during the Great East Japan Earthquake (2011), are significant and cannot be predicted using conventional tools. This leads to the necessity of performing a consistent site effect analysis against seismic response challenges.

Another issue to be highlighted is related to the practice to cope with seismic design of safety SSCs. Additionally to dynamic forces; the Japanese practice incorporates several approaches like application of more than three times the static seismic loads of the Japanese conventional building code. This result in a robust design of safety SSCs that

4. Recorded levels from 2 to 5 times the design levels.
5. Different effects at Fukushima Daiichi than at Fukushima Daiini, with very similar distance to the epicenter.

clearly prevented significant consequences beyond a seismic design exceedance; but the contribution from dynamic or static design levels to this success are not yet identified and evaluated. This is a significant issue for plants not concerned for this practice.

A.4.5.2 Safety Research Interest

Site effect is identified as a safety research issue of interest, in selecting with confidence earthquake levels to cover long term seismic challenges. Nevertheless the **safety research interest is low** as better data from other plants (not subject to accident) can be gained.

Another topic is comparing seismic design inputs and recorded levels, to obtain lessons learnt for seismic engineering practical purposes.

Likewise, another safety research issue of interest will be exploring and, if possible, to separate the contribution of dynamic and static loads, to the successful behaviour of safety SSCs response to loads beyond design values.

A.4.5.3 Decommissioning Interest

The **decommissioning interest is low**: Information on seismic response at the accident is not really needed for planning decommissioning strategies, while there is need to ensure seismic stability during decommissioning activities, but the information on the seismic event of 2011 is not really needed for decommissioning.

A.4.5.4 Potential Examinations

A comprehensive reassessment of site seismic parameters of the Fukushima Daiichi plant using recorded data and current knowledge needs to be done.

A.4.5.5 NEA Ongoing R&D Activities

At the NEA level, a previous activity concerning site effect has been developed and published as a report [2] that can be taken into account as starting activity.

A good plan should be to develop a benchmark exercise using Japanese records in order to assess site effect uncertainties. In this sense a proposal was already presented at the Seismic Subgroup of the WGIAGE through the corresponding CAPS (WGIAGE (2014) – Seismic input definition and its control point: a practical application), but this initiative is waiting for a leader to carry it out.

A.4.5.6 References

[1] IAEA. Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety - The Accident at TEPCO's Fukushima Nuclear Power Stations, June 2011.

[2] OECD/NEA, Current Practices in Defining Seismic Input for Nuclear Facilities, NEA/CSNI/R(2015)9, May 2015

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