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Reactor Core and Containment Cooling Systems: Long-Term Management and Reliability (RCCS-2021)

NEA Specialist Workshop
Proceedings
18-20 October 2021
Virtual Meeting

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NUCLEAR ENERGY AGENCY
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NUCLEAR ENERGY AGENCY

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The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes;
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

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

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The Committee on the Safety of Nuclear Installations (CSNI) addresses Nuclear Energy Agency (NEA) programmes and activities that support maintaining and advancing the scientific and technical knowledge base of the safety of nuclear installations.

The Committee constitutes a forum for the exchange of technical information and for collaboration between organisations, which can contribute, from their respective backgrounds in research, development and engineering, to its activities. It has regard to the exchange of information between member countries and safety R&D programmes of various sizes in order to keep all member countries involved in and abreast of developments in technical safety matters.

The Committee reviews the state of knowledge on important topics of nuclear safety science and techniques and of safety assessments, and ensures that operating experience is appropriately accounted for in its activities. It initiates and conducts programmes identified by these reviews and assessments in order to confirm safety, overcome discrepancies, develop improvements and reach consensus on technical issues of common interest. It promotes the co-ordination of work in different member countries that serve to maintain and enhance competence in nuclear safety matters, including the establishment of joint undertakings (e.g. joint research and data projects), and assists in the feedback of the results to participating organisations. The Committee ensures that valuable end-products of the technical reviews and analyses are provided to members in a timely manner, and made publicly available when appropriate, to support broader nuclear safety.

The Committee focuses primarily on the safety aspects of existing power reactors, other nuclear installations and new power reactors; it also considers the safety implications of scientific and technical developments of future reactor technologies and designs. Further, the scope for the Committee includes human and organisational research activities and technical developments that affect nuclear safety.

Foreword and acknowledgements

The Nuclear Energy Agency (NEA), its Committee on the Safety of Nuclear Installations (CSNI) and the Working Group on Analysis and Management of Accidents (WGAMA) acknowledge the significant contributions of those individuals who had a key role in the preparation of the summary report, and those who had a leadership role in the conduct and success of the NEA Specialist Workshop on Reactor Core and Containment Cooling Systems – Long-Term Management and Reliability (RCCS-2021).

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Additional thanks are extended to the organising committee members in support of the workshop local organisation, and to the scientific committee members and external reviewers for their support in the review process.

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List of abbreviations and acronyms

AWI	Alternative water injection
BDBA	Beyond design-basis accident
BWR	Boiling water reactor
CANDU	Canada Deuterium Uranium reactor
CCC	Containment cooling condenser
CCS	Containment cooling system
CEA	Commissariat à l'énergie atomique et aux énergies alternatives (French Alternative Energies and Atomic Energy Commission)
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (Centre for Energy, Environmental and Technological Research, Spain)
CNL	Canadian Nuclear Laboratories
CNPRI	China Nuclear Power Technology Research Institute
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners' Group
CRIEPI	Central Research Institute of Electric Power Industry (Japan)
CSNI	Committee on the Safety of Nuclear Installations (NEA)
CSS	Containment spray system
CFS	Cavity flooding system
DBA	Design basis accident
DEC	Design extension conditions
DEC-A	DEC without significant fuel degradation or core melt
DEC-B	DEC with core melt
ECC/ECCS	Emergency core cooling/emergency core cooling system
EDF	Électricité de France
ENEA	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EPR	European pressurised water reactor
ERVC	External reactor vessel cooling
ESG	Energy Safety Group (Ukraine)
ETSON	European Technical Safety Organisation Network
FCVS	Filtered containment venting system
FP	Fission product
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit

IAEA	International Atomic Energy Agency
IJS	Institut Jožef Stefan (Jožef Stefan Institute, Slovenia)
IRWST	In-containment refuelling water storage tank
IRSN	Institut de radioprotection et de sûreté nucléaire (French Institute for Radiological Protection and Nuclear Safety [ASNR since January 2025])
IVR/IVMR	In-vessel retention/in-vessel melt retention
JAEA	Japan atomic energy agency
KAERI	Korea Atomic Energy Research Institute
KTH	Royal Institute of Technology (Sweden)
LBLOCA	Large break loss-of-coolant accident
LHS	Latin hypercube sampling
LOCA	Loss-of-coolant accident
LTCCS	Long-term containment cooling system
LWR	Light water reactor
MCCI	Molten corium-concrete interaction
NEA	Nuclear Energy Agency
NNSA	National Nuclear Safety Administration (China)
NRA	Nuclear Regulation Authority (Japan)
OC	Organising committee
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation (United Kingdom)
PAR	Passive autocatalytic recombiner
PHWR	Pressurised heavy water reactor
PSI	Paul Scherrer Institute
PWR	Pressurised water reactor
REPAS	Reliability evaluation of passive safety
RCS	Reactor coolant system
RPV	Reactor pressure vessel
RWST	Reactor water storage tank
SASS	Severe accident spray system
SC	Scientific committee
SEC NRS	Scientific and Engineering Centre for Nuclear and Radiation Safety (Russia)
SMART-100	System integrated modular advanced reactor, Korean small modular reactor
SMR	Small modular reactor

TSO	Technical support organisation
UJD SR	Nuclear Regulatory Authority of the Slovak Republic
US NRC	United States Nuclear Regulatory Commission
VVER	Water water energy reactor (Russian PWR)
WGAMA	CSNI Working Group on Analysis and Management of Accidents

Executive summary

The specialist workshop entitled "Reactor Core and Containment Cooling Systems: Long-term Management and Reliability (RCCS-2021)" was organised by the Nuclear Energy Agency (NEA) as part of the activities of the Working Group on Analysis and Management of Accidents (WGAMA) of the Committee on the Safety of Nuclear Installations (CSNI).

The activity was co-ordinated by the French Institut de radioprotection et de sûreté nucléaire (IRSN) and attracted 188 experts from more than 28 countries representing regulatory organisations, reactor designers, operators, consultancies, engineering companies, technical support organisations (TSOs) and research establishments.

The workshop was organised as a follow-up to NEA activities on establishing a knowledge base for long-term core cooling reliability (NEA, 2013) and on the long-term management and actions for a severe accident in a nuclear power plant (NEA, 2021). It aimed at providing an update on recent developments related to practices for maintaining the core and containment cooling functions in the long term, following an accident in a nuclear power plant, with the intent of including most recent upgrades as well as lessons learnt from the Fukushima Daiichi accident. The workshop mainly addressed two key issues that may affect the reliability of the cooling systems:

- the degradation of components and/or systems essential to the cooling function in accident conditions under the combined effects of temperature, pressure, humidity, radiation doses and chemical environment;
- the clogging of filters on systems supplying water to the core and containment because of material transported into the sump (i.e. fibrous insulation, latent debris within the containment, corrosion material, paint or concrete).

In addition, new and foreseen technical solutions to ensure long-term cooling in different reactor types were discussed.

The workshop included several keynote lectures on regulatory aspects, cooling systems, accident management and lessons learnt from the Fukushima Daiichi accident. Technical sessions on four topics were set to address the above-mentioned issues:

- accident management measures and strategies;
- new systems and designs for post-accidental long-term core/corium and containment cooling;
- filtration system performance and clogging issues;
- debris sources, with a focus on chemical conditions in cooling waters.

It was noted that collaborative, experimental and/or analytical investigations are effective in developing the technical bases needed to demonstrate the long-term reliability of equipment after a severe accident. This is of interest for existing designs as well as new designs, including small modular reactors (SMRs).

Regarding experimental investigations, it appears that experimental facilities and expertise have been developed for investigations mostly related to the long-term reliability of cooling systems in loss-of-coolant accidents (LOCAs) without significant core damage. Additional efforts are needed to address more specifically material response for severe accident prevailing conditions with combined chemistry and radiation effects. Challenges may be related to defining experimental investigations of interest to various designs and accident scenarios, covering realistic severe accident conditions and ultimately addressing scaling.

It is therefore recommended to conduct a ranking exercise to prioritise the phenomena to be investigated, establish research plans in the area while defining experimental investigations and assess whether the methodology and calculation tools developed for LOCAs without significant core damage are applicable to severe accidents or need to be completed. In this exercise, the following aspects would need to be considered:

- Establishing potential debris sources for severe accidents, e.g. considering the degradation of heat insulation and organic materials (e.g. paints, composite liners, cables), the degradation/corrosion of structural material from normal operation and accident conditions and the contribution to debris/particle formation of core debris. Learnings from the Fukushima Daiichi accident would be valuable, though variability in accident scenarios and system designs would have to be addressed.
- The formation of chemical precipitates from non-metallic insulation debris including thermal effect, buffer composition, pH effect, ionic strength, effect of chemical additives.
- Possible combined corrosion, chemical and radiation effects on structural materials and components of cooling systems and the possible effect on cooling systems.

To this end, small-scale experimental investigations of the separate and combined effects of the corrosion, chemistry and radiation of materials on material debris may help address these challenges.

The level of sharing data internationally could be increased, particularly for design specific investigations conducted on a national basis, such as for downstream effects investigations. Those effects comprise all phenomena that apply to components after the water/debris mixture has passed the sump strainer. This could reduce the need for additional testing requirements.

As regards components that contribute to containment leak tightness in a severe accident (e.g. liners, seals), it is recommended that experts share information to further assess the technical bases that can be used to predict the components' response during accidents and potential containment leak tightness failures. Collectively identifying key knowledge gaps and defining and developing approaches to address them from the severe accident and ageing management perspectives could help enhance the life management of these components and reduce the risks of failure and of radioactive release in accidents.

Significant modelling uncertainties remain in the prediction of formed core material debris configurations and debris bed properties in severe accidents. The development of models to address long-term core debris bed coolability and scaling, ranging from sump clogging separate and integral tests to reactor application, could benefit from collaborative efforts.

1. Introduction

1.1. Objectives, scope and background of the specialist workshop "Reactor core and containment cooling systems: Long-term management and reliability"

Core and containment heat removal is one of fundamental safety functions to be maintained during accident conditions in a nuclear power plant. The Fukushima Daichi accident highlighted the importance of maintaining the cooling function and the difficulty of managing a large amount of contaminated water; this in turn underscored the need for reliable and sustainable recirculation strategies to ensure core and containment heat removal for a long period of time. Similarly, the publication *Long-Term Management and Actions for a Severe Accident in a Nuclear Power Plant* (NEA, 2021) underlined the need to address open issues and technological gaps related to maintaining long-term cooling with:

- the development of knowledge on the long-term reliability of systems and equipment involved in maintaining a coolable configuration;
- the assessment of failure risks due to induced mechanical weakness, clogging with debris and corrosion-erosion reactions;
- the development of knowledge on material degradation under SA conditions (high dose and temperature) that may form significant quantities of debris, posing challenges for long-term cooling;
- the development of strategies for flooding and cooling the corium to avoid the transfer of contaminated water outside the confinement;
- the development of knowledge on the use of water with controlled chemistry to limit the risk of re-criticality, fission product (FP) remobilisation, corrosion, clogging, and to facilitate water management in the long term[1].

Previous related NEA activities, such as those of the task group on sump clogging (NEA, 2013), had mostly focused on loss-of-coolant accident (LOCA) conditions. The task group recognised the knowledge base for long-term core cooling reliability in a LOCA had been enhanced significantly on chemical effects and downstream effects and that test facilities were available to provide insights into fundamental phenomena and to perform plant specific testing. At the time, the group had considered that collaborative projects could have been beneficial to further establish the knowledge base. However, since then, research efforts have mostly been conducted nationally.

Considering the lessons learnt from these activities, as well as those resulting from the Fukushima Daiichi-related actions and projects (Nakayoshi et al., 2020) (Pellegrini et al., 2020), it was decided to organise the specialist workshop entitled "Reactor Core and Containment Cooling Systems: Long-term Management and Reliability (RCCS-2021)" under the auspices of the NEA. The aim of the workshop was to promote international exchanges of information and practices and to identify knowledge gaps related to maintaining the core and containment cooling functions under long-term accident conditions in a nuclear power plant.

The RCCS-2021 workshop was announced in April 2020 with the objective of holding it in Levice, Slovak Republic, in February 2021. A visit of the VUEZ experimental facility

1. "Long term" here refers to accident management actions implemented after a plant has reached a stabilised and controlled state.

VIKTORIA was planned for participants of the workshop. Because of travel restrictions caused by the COVID-19 pandemic in Europe, the organising committee decided to postpone the RCCS-2021 workshop and hold it online on 18-21 October 2021.

The workshop focused on the knowledge that needed to be developed on the degradation in long-term accident conditions (under the combined effects of temperature, pressure, radiation dose and chemical environment) of material and components that can impair the maintenance of cooling and the effects of corrosion-erosion reactions in sensitive components in the long term.

The workshop scope was broadened to include a comprehensive survey of the existing safety standards and rules related to cooling reliability in the long term as well as a review of the existing knowledge established through testing (including qualification) of material and components relevant to accident conditions. The workshop also addressed clogging in cooling systems' upstream filters and downstream in sensitive components (e.g. heat exchangers, valves), as well as new systems and designs for post-accidental long-term core/corium and containment cooling.

1.2. Organisation, programme and structure

An organising committee (OC) oversaw the planning and organisation of the meeting on behalf of the WGAMA. The OC members are presented in the table below:

Table 1.1. Organising committee members

OC member	Organisation	Country
Ahmed Bentaib	IRSN	France
Luis E. Herranz	CIEMAT	Spain
Ali Tehrani	ONR	United Kingdom
Noreddine Mesmous	CNSC	Canada
Lubica Kubisova	UJD SR	Slovak Republic
Ivan Vicena	VUEZ	Slovak Republic
Viktoria Valachovicova	VUEZ	Slovak Republic
Juraj Kubica	VUEZ	Slovak Republic
Martina Adorni	NEA	
Didier Jacquemain	NEA	

The workshop agenda (see Annex A) comprised short presentations and discussions. The presenters were requested to provide pre-recorded presentations that were made available to attendees prior to the workshop.

The workshop featured seven keynote lectures, four technical sessions and 29 peer-reviewed papers addressing the following:

- Topic 1 - Accident management measures and strategies.
- Topic 2 - New systems and designs.
- Topic 3 - Sump clogging issues.
- Topic 4 - Debris formation and chemical conditions in cooling waters.

A summary session was organised on the last day to present and discuss the preliminary conclusions and to draw comprehensive thoughts and ideas for future research programmes.

To ensure the relevance and quality of the contributions, the abstracts and extended abstracts submitted by the presenters were examined with the support of nearly 30 people, including the members of the scientific committee (SC), the OC members and external reviewers. The workshop programme was established based on selected extended abstracts.

To paint a global picture of recent developments and practices adopted for core and containment long-term cooling, considering LWR and PHWR technologies, the OC invited the following keynote speakers to open the plenary session:

- Jorge LUIS HERNANDEZ, IAEA, with the presentation "Reactor core and containment cooling systems regulatory aspects from IAEA standards".
- Bruno TOURNIAIRE, EDF, France, with the presentation "Reactor core and containment cooling systems in French nuclear power plants".
- Karim OSMAN, Ontario Power Generation, Canada, with the presentation "SAMG long term core and containment cooling strategy application and management in Canadian nuclear power plants".
- Pingting JIANG, CNPRI, China, with the presentation "Long term management strategies of containment pressure control during severe accidents".
- Randy BUNT, Southern Nuclear, United States, with the presentation "Reactor core and containment cooling issues in DBA and SA differences – BWR".
- Hossein ESMAILI, US NRC, United States, with the presentation "Post-Fukushima severe accident research activities at NRC - Containment vents and hydrogen control".
- Shinya MIZOKAMI, TEPCO, Japan, with the presentation "Lessons learned from 1F accident by focusing on the long-term cooling".

The 22 technical presentations were divided into four technical sessions dedicated to different topics, as shown below:

Table 1.2. Number of contributions per topic

Topic N°	Session title	Number of contributions
1	**Assessment of accident management measures and strategies**	3
2	**New systems and designs**	8
3	**Sump clogging issues**	5
4	**Debris formation and chemical conditions in cooling waters**	6

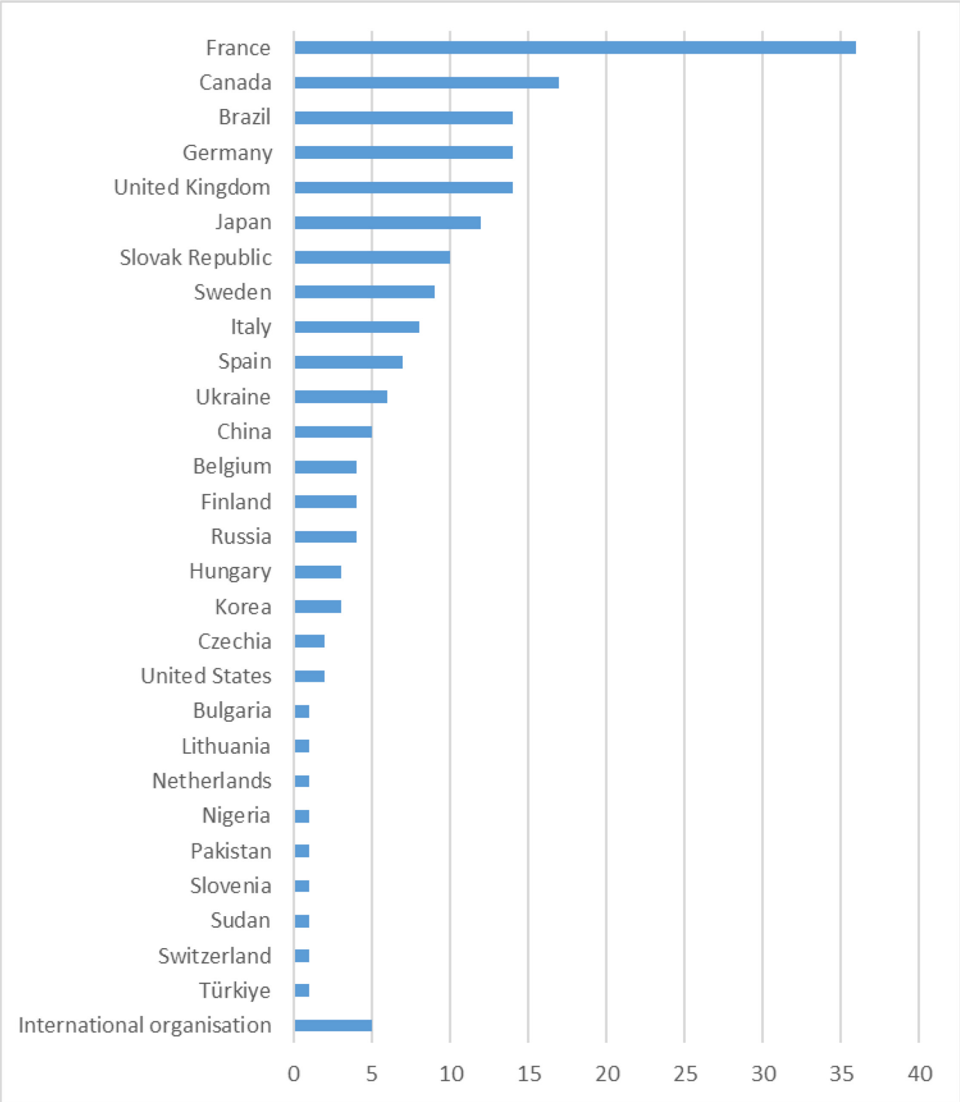
In addition, the summary session, organised on the last day, served to collect the participants' feedback and views of the workshop, as well as the session chairpersons' evaluations and syntheses. The summary session also allowed for thorough discussions on specific topics and issues that emerged during the workshop.

1.3. Attendance

The specialist workshop "Reactor Core and Containment Cooling Systems: Long-Term Management and Reliability" (RCCS-2021) gathered 188 participants representing more than 28 countries and 3 international organizations. Participants included experts from regulatory organisations, reactor designers, operators, consultancies, engineering companies, TSCs/TSOs supporting the regulators and research establishments and international organisations. The geographical origin of the participants is shown in Figure 1.1.

The diversity in participation is an indication of international interest. However, the online meeting mode occasionally precluded the thorough technical discussions that face-to-face meetings allow.

Figure 1.1. Geographical origin of RCCS-2021 participants



1.4. Structure of the report

The chair of each plenary and technical session was asked to prepare a short synthesis report of the session based on the written papers, presentations and discussions. Based on this written material, Chapter 2 summarises the main outcomes and discussions held during the sessions. Chapter 3 provides the concluding remarks and recommendations.

2. Summary of the workshop sessions/topics

2.1. Keynote lectures

Session Chairs: Luis Enrique Herranz (CIEMAT, Spain) and Ahmed Bentaib (IRSN, France)

The keynote lectures aimed to provide an overview of the regulation for cooling systems based on IAEA standards, strategies/practices used for cooling the core and/or cooling and maintaining the containment after an accident considering PWR, BWR and PHWR technologies. The lessons learnt from the Fukushima Daiichi accident were given special attention.

Accordingly, the session was organised in three items:

Item 1: General regulatory aspects (paper # **1010**)

Item 2: Long-term management for different nuclear power plant technologies:

- Practices for PWR (paper # **1020**).
- Practices for BWR (paper # **1050**).
- Practices for PHWR (paper # **1030**).

Item 3: Lessons learnt from the Fukuhisma Daiichi accident (paper #**1060** and paper # **1070**)

The list of the keynote speakers is provided in Table 2.1:

Table 2.1. Keynote lectures list

#1 010	*Reactor core and containment cooling systems regulatory aspects from IAEA standards* Dr Jorge LUIS HERNANDEZ, IAEA
#1 020	*Reactor core and containment cooling systems on French nuclear power plants* Dr Bruno TOURNIAIRE, EDF, France
#1 030	*SAMG long-term core and containment cooling strategy application and management in Canadian nuclear power plants* Mr Karim OSMAN, Ontario Power Generation, Canada
#1 040	*Long-term management strategies of containment pressure control during severe accidents* Dr Pingting JIANG, CNPRI, China
#1 050	*Reactor core and containment cooling issues in DBA and SA differences – BWR* Mr Randy BUNT, Southern Nuclear, United States
#1 060	*Post-Fukushima severe accident research activities at NRC - containment vents and hydrogen control* Dr Hossein ESMAILI, US NRC, United States
#1 070	*Lessons learnt from 1F accident by focusing on the long-term cooling* Dr Shinya MIZOKAMI, TEPCO, Japan

Item 1: General regulatory aspects

Concerning regulatory aspects at the international level, Dr Hernandez provided an overview of the updated design safety requirements and recommendations in the IAEA safety standards applicable to new large nuclear power plants equipped with water-cooled reactor designs. Although the IAEA requirements and recommendations on the design were highlighted as technology-neutral, the vast operating experience gained throughout the years might have affected their current formulation. Particularly relevant to RCCS-2021 was the IAEA recommendation to ensure residual heat removal from the reactor core by avoiding the clogging of sump strainers and filters and by pushing forward strategies for in-vessel melt retention and ex-vessel corium cooling.

As for those nuclear power plants designed in accordance with former standards, the application of all the IAEA requirements and all IAEA recommendations might need to be assessed (e.g. through periodic safety reviews, licence renewals or design review processes) to define which reasonably practicable safety improvements could be made to further enhance the safe operation of the nuclear power plant.

For nuclear power plants equipped with other reactor types, such as non-water-cooled reactor technologies or water cooled small modular reactors, the IAEA requirements and recommendations on the design might not be fully applicable as they are; further engineering judgement may be required.

Regarding long-term severe accident management, Dr Hernandez mentioned the IAEA Action Plan on Nuclear Safety to define a comprehensive working programme, including the reinforcement of research and development, particularly in the management of severe accidents. Two strategies were identified: in-vessel melt retention and ex-vessel corium cooling/stabilisation. Both strategies were said to need further knowledge of the associated phenomena to reduce the uncertainties concerning their effective implementation.

Dr Hernandez finished his talk by mentioning the IAEA safety guides and recommendations related to other long-term strategies, such as the control of combustible gases in the atmosphere of the containment, the control of the pressure inside the containment and the control of radioactive releases. He also mentioned the IAEA recommendations related to the use of portable equipment; their maintenance and inspection; waste management due to long-term actions such as water treatment; limits to dose rates to ensure the feasibility of the operator actions and availability of electrical, compressed air or water sources.

Item 2: Long-term management for different nuclear power plant technologies

Regarding long-term management for PWR technologies, the presenters described the methodologies adopted to deal with long-term cooling.

Dr Tourniaire provided an overview of the reactor and containment cooling systems already implemented in the French generation 2 (GEN2) fleet and those implemented in the EPR. Both design extension condition without significant fuel degradation (DEC-A) and with core melt (DEC-B) situations were covered.

For GEN2, the ongoing upgrade ensures the core and containment cooling in the long term, based on a system named ultimate containment spray system (CSS). CSS consists of a pump injecting water from the reactor water storage tank (RWST) into the primary circuit and the sumps of the reactor building and then recirculating it to the sumps. In addition, the heat exchanger is capable of evacuating the residual power from the reactor building with its own mobile ultimate heat sink, as implemented by the Nuclear Rapid Action Force. This system is supplied by an emergency diesel generator, which is diversified from the main diesel generators and is qualified for severe accident conditions (in terms of temperature, irradiation, pressure and debris) and designed to withstand extreme external hazards.

Dr Tourniaire indicated that, in case of ultimate CSS failure in the long term, the filtered venting system is credited to remove the residual heat power.

For GEN3 reactors, Dr Tourniaire explained, dedicated systems are implemented to remove residual power without the need for containment venting.

As regards other new PWR designs, Dr Pingting's lecture focused on their containment pressure control strategies, with the major subject being how and when to use the systems properly in long-term management strategies. Relevant systems that help control the containment pressure, like containment heat removal systems and containment venting systems, were introduced and their effectiveness was assessed based on numerical studies including sensitivity analysis.

As regards PHWRs, Mr Osman presented an overview on the strategies adopted by the Canadian utilities along with CANDU Owners Group (COG) to ensure sustained core and containment cooling post core degradation. These strategies are developed as guidelines that provide various means of adding water to the core or the shield tank to cool the fuel or corium to maintain in-vessel retention (IVR). Core cooling after the onset of severe accident conditions can be achieved by adding water to the heat transport system or moderator for in-vessel core cooling and/or the shield tank to maintain ex-vessel cooling. These strategies are designed to provide water flowrates that minimise steaming while cooling the core and preventing pressurisation of the reactor building to prevent radioactive releases. To support the water addition strategies, computational aids were created to estimate the rate of water addition at various times into the event.

As regards BWRs, Mr Bunt presented an overview on the guidance developed by the BWR Owner Group to accomplish reactor core and containment cooling in both design basis events and under severe accident scenarios in BWRs. This keynote presentation highlighted the features of the BWR design and event response capabilities related to reactor core and containment cooling.

Item 3: Lessons learnt from the Fukushima Daiichi accident

Regarding the lessons learnt from the Fukushima Daichi accident, Dr Esmaili provided a summary of several analytical severe accident research activities undertaken by the NRC to support the regulatory basis for filtered containment venting and hydrogen risk management. As described below, the containment cooling strategy depends on the reactor design:

- A combination of containment venting and water addition is required to prevent containment failure and to mitigate radiological releases for the BWR Mark I and II containments.
- Anticipatory venting (before core damage) is beneficial to reduce the containment pressure and delay the radionuclide release to the environment.
- For BWR Mark I and II containments, containment venting is efficient in purging hydrogen and non-condensable gases. Water injection is also helpful in maintaining a steam-inerted atmosphere which can preclude energetic hydrogen combustion. The releases into the environment from a BWR Mark II containment are generally comparable to or lower than those from a BWR Mark I containment.

- For both PWR ice condenser and BWR Mark III containment types, igniters are important for providing adequate containment integrity past the 24-hour mark. The igniters would delay but not alleviate a potential containment overpressure failure.
- PWR ice condenser: Without igniters, containment fails soon after hot leg rupture (~24 hours), while use of igniters can control hydrogen and limit the containment pressure so that a containment failure is not likely within three days.
- BWR Mark III: Without igniters, the containment fails by overpressure soon after lower head rupture (~18 hours). With the igniters, containment failure by overpressure is significantly delayed (> one day). Activation of containment sprays at the time of lower head failure further delays the containment overpressure failure.

Dr Mizokami described the reactor and debris cooling systems and alternative approaches used during the Fukushima Daiichi accident. In his presentation, Dr Mizokami discussed the behaviour of each piece of equipment used in the three damaged units and the differences observed in their evolution. He highlighted that the reactor responses observed during the Fukushima Daiichi accident to the management measures taken have the potential to substantially help improving the safety of the existing and future nuclear power plants, particularly in the frame of accident management.

General conclusions and recommendations from the keynote session

The presentations and discussions led to the following conclusions:

- it is beneficial to upgrade the core and containment cooling safety systems with respect to the latest IAEA standards for large nuclear power plants;
- there is a need for further engineering judgement to satisfy the IAEA standards for nuclear power plants such as non-water-cooled reactor technologies or water cooled small modular reactors;
- it is relevant for long-term cooling strategies to include the use of portable equipment, their maintenance and inspection, the waste management due to long-term actions such as water treatment, the limits to dose rates to ensure the operator actions and availability of electrical, and compressed air or water sources;
- safety improvements have been made to existing and future nuclear power plants based on the lessons learnt from the Fukushima Daiichi accident, such as the use of mobile ultimate heat sinks and the implementation of emergency diesel generators credited for severe accident conditions – temperature, irradiation, pressure, debris – and designed to withstand extreme hazards.

Even though significant nuclear power plant safety enhancements have already been achieved based on Fukushima Daiichi accident investigations, ongoing research on the matter will likely result in further optimisation.

2.2. Technical sessions

2.2.1. Topic 1: Assessment of accident managements measures and strategies

Topic Chairs: Ali Tehrani (ONR, United Kingdom) and Noreddine Mesmous (CNSC, Canada)

The Topic 1 technical session was dedicated to the assessment of accident management measures and strategies. The presentations covered a broad range of areas, providing a better understanding of the impact of different parameters and how these will affect accident progression. The presentations also highlighted the contributions made by the post-accident analysis of the Fukushima Daiichi plant, supported by several international initiatives. The presentations are listed in Table 2.2:

Table 2.2. Topic 1 presentation list

#1 110	Review and perspective of GRS' long term accident analyses with consideration of structure mechanical aspects K. HECKMANN, M. SONNENKALB, J. SIEVERS, T. STEINRÖTTER, S. PALAZZO, J. ARNDT, C. BLÄSIUS (GRS, Germany)
#1 120	Long term containment cooling in Fukushima Unit 1: Consequences and sensitivity L.E. HERRANZ, R. BOCANEGRA (CIEMAT, Spain)
#1 150	Assessment of long-term containment conditions using MELCOR and ASTEC codes M. DI GIULI, P. DEJARDIN, M. AUGLAIRE (Tractebel Engineering (SA) Engie, BELGIUM)

These presentations have helped to reach a better understanding of some of the key issues and are summarised below.

Paper # 1110: **"Review and perspective of GRS' long term accident analyses with consideration of structure mechanical aspects", Heckmann, K., C. Bläsius, T. Steinrötter, S. Palazzo, M. Sonnenkalb and J. Sievers; GRS Germany**

The authors presented analyses of the reliability of long-term provision of cooling that might be affected by damage caused by DBA or BDBA scenarios. The authors noted that from the structural integrity perspective, the assessment of long-term cooling requires an analysis of "short-term" ageing due to challenging environmental conditions and failure of the pressure-retaining boundary, including sealing, with consideration of material properties far from design conditions. Consideration of the increase of radioactivity in the coolant loops, corrosion effects due to impaired water quality, debris in the coolant and potential clogging of safety systems should be included in the assessment.

The authors presented the relevant phenomena and developed methods for the assessment of reliability with relevant aspects for the long-term management of cooling. For loss-of-coolant accident scenarios, clogging phenomena, including the impact of zinc borate, and recommendations for mitigation were highlighted. In addition, the computational approach for the high-temperature damage of components during the BDBA transient phase and the recovery of systems/components in severe accident management measures were highlighted. The authors noted relevant gaps concerning methodology and knowledge, as well as possible improvements to reduce uncertainties.

The remaining challenge in this work is to characterise the materials' behaviour under long-term accident conditions far from design conditions, and to enhance structure mechanical computing tools to allow reliable assessment of component degradation. This also requires a validation of structural simulation approaches for extreme conditions. Additional research is required to enable further consideration of these effects in accident analyses and potential recommendations.

Paper #1 120: **"Long term containment cooling in Fukushima Unit 1: Consequences and sensitivity", L.E. Herranz and R. Bocanegra, CIEMAT, Spain**

The authors investigated the long-term cooling operation following the accident at unit one of the Fukushima Daiichi Nuclear Power Plant. They provided a summary of sensitivity analyses carried out within the BSAF project with the latest version of MELCOR 2.2. The analyses presented focused on evaluating the impact of the alternative water injection (AWI) on the containment pressure and on fission product release to the environment. The reference case was supported by two sensitivity studies. The first focused on the cooling magnitude, and the second addressed the AWI timing.

According to the results presented, once containment integrity is breached and flow paths established, effects of a later water injection would not be heavily dependent on timing but the amount and rate of water injection might substantially affect the radioactive release on the environment.

Paper # 1150: **"Assessment of long-term containment conditions using MELCOR and ASTEC codes", Di Giuli, M., P. Dejardin and M. Auglaire from Tractebel ENGIE, Brussels, Belgium**

The authors presented an overview of factors governing the containment atmosphere composition and conditions post severe accident involving reactor pressure vessel failure. They went on to point out that the phenomenon with the greatest impact on containment conditions is the molten corium concrete interaction (MCCI), which in turn affects the amount of steam and gases released following the concrete ablation. The studies are performed using MELCOR and ASTEC codes in stand-alone mode evaluating the impact of different severe accident management (SAM) strategies on long-term containment conditions (10-30 days) for several distinct scenarios in a PWR plant. The calculated results highlighted the main parameters affecting the diverse SAM strategies reproduced and provided useful indications of the code capabilities utilised.

The authors provided a high-level overview of the methodology that can be readily applied when assessing the impact of different SAM guidelines (SAMGs) to identify the maximum loading on the containment for a variety of ex-vessel conditions as well as to bypass existing uncertainties related to corium debris coolability. The work is continuing, particularly relating to the capabilities of the simulation codes being used, and will be made available in appropriate venues.

General conclusions and recommendations for Topic 1.

The main discussions focused on the suitability of the assumptions for prolonged post-accident conditions in the analyses presented. Thus, the following was highlighted:

- The assessment of long-term cooling requires an analysis of "short-term" ageing due to challenging environmental conditions and failure of the pressure-retaining boundary, including sealing, with consideration of material properties far from design conditions.
- There is a need for further knowledge development to address the remaining challenge related to the characterisation of the materials' behaviour under long-term accident conditions far from design condition, and to enhance structure mechanical computing tools to allow reliable assessment of component degradation.

2.2.2. *Topic 2: New systems and designs*

Topic Chairs: L. Kubisova (UJD SR), M. Shawkat (CNSC) and T. Van Rompuy (BelV)

The Topic 2 session was dedicated to the new systems and designs introduced to prevent the escalation of design basis accidents into severe accidents and/or mitigate the consequences of any severe accidents in nuclear power plants. The presentations are listed in Table 2.3.

Table 2.3. Topic 2 presentations list

#1 160	Development of severe accident mitigation system and technology for Korean small integral reactor R. PARK, S. KIM (KAERI, South Korea)
#1 170	VVER-440/V213 containment cooling system for severe accidents P. MATEJOVIC, M. BARNAK, Z. TUMA (IVS, Slovak Republic and Dukovany Nuclear Power Plant, Czechia)
#1 180	Challenges in the design of long-term containment cooling system at Paks Nuclear Power Plant G. LAJTHA, E. TOTH, L. TARCZAL, T. SIKLOSSY (NUBIKI, Hungary)
#1 190	Fully passive long term containment cooling of VVER440-V213 reactors M. HUPP, M. BRAUN, N. LOSCH (Framatome, Germany)
#1 210	Study of the additional upgrade’s implementation for long-term heat removal from the containment of nuclear power plants with VVER-1000 reactor O. KOCHARYANTS, O. MAZUROK, V. IVANOV, O. MIHAYLENKO (ESG, Ukraine)
#1 220	Study of the additional upgrade’s implementation for long-term heat removal from the confinement of nuclear power plants with VVER-440 reactor O. KOCHARYANTS, O. MAZUROK, V. IVANOV, O. MIHAYLENKO (ESG, Ukraine)
#1 340	Qualification and testing of a water turbine driven pump for use in a long-term containment cooling system E. KOSTOV, A. PÉREZ-SALADO KAMPS, C. HARTMANN, D. LAUER, D. ZENIUK, AND H. TROSELIUS (Westinghouse and KSB SE & Co. KGaA, Germany)

The presentations covered new systems and designs proposed for SMART-100, VVER-440/V213, and VVER-1 000 reactors. Most of the presentations focused on the new systems designed to resolve the concerns regarding the integrity of VVER-440/V213 containment due to slow and long-term (beyond three days) pressurisation following severe accidents. All VVER-440/V213 aim at preventing reactor vessel failure by in-vessel corium retention through external reactor vessel cooling (IVR-ERVC), but proposed solutions for long-term containment cooling are different. Below is a summary of each presentation

Paper #1160: **“Development of severe accident mitigation system and technology for Korean small integral reactor”, Rae-Joon Park, Sang Ho Kim, KAERI, Korea**

The paper presented the severe accident mitigation systems available in the Korean small integral reactor SMART-100 (system-integrated modular advanced reactor) designed to produce 365 MW(th). The main concept used to terminate the progression of severe accidents is to prevent reactor vessel rupture using IVR-ERVC. This is achieved using the cavity flooding system (CFS) with an in-containment refuelling water storage tank (IRWST). In addition, SMART-100 incorporates other systems to rapidly depressurise the reactor coolant system (RCS) to avoid high-pressure melt ejection and hydrogen control systems to mitigate the risk of hydrogen combustion. Long-term cooling of the containment is to be ensured by the ancillary containment spray system.

Paper #1170: **"VVER-440/V213 containment cooling system for severe accidents", Matejovic, P., M. Barnak, Z. Tuma, Inzinierska Vypoctova Spolocnost [IVS], Trnava, Slovak Republic and Dukovany Nuclear Power Plant, Czechia**

The paper discussed, based on a simulation performed with the ASTEC v2.1 code, the performance of the new systems added to Dukovany Nuclear Power Plant VVER-440/V213 to cope with the consequences of severe accidents. These systems include IVR-ERVC, PARs, and a new depressurisation line on the pressuriser, the latter currently under implementation. The paper discussed the results of ASTECv2.1 simulations of the containment response following LBLOCA without availability of any containment heat removal system to estimate the time margin to reaching the containment ultimate strength. Another set of simulations examined the performance of a new independent containment cooling system (CCS) that utilises turbo-pumps and heat exchangers inside the containment. These are used to circulate the sump water through sprayers while being driven/cooled by an external cooling circuit. Implementation of the new CCS is under preparation in all four units of Dukovany Nuclear Power Plant.

Paper #1 180: **"Challenges in the design of long-term containment cooling system at Paks Nuclear Power Plant", Lajtha, G., E. Toth, L. Tarczal, T. Siklossy, Safety Analyses Division, NUBIKI and Paks Nuclear Power Plant Ltd., Hungary**

The authors discussed the challenges in the design of a long-term containment cooling system for VVER-440/V213 at Paks Nuclear Power Plant. Based on MAAP4-VVER simulations, they investigated various possible solutions for dealing with the issue of containment pressurisation. Following a severe accident and initiation of reactor external cooling of the vessel, the steaming from the cooling water is expected to pressurise the containment and a long-term cooling system will be required to ensure containment integrity. Different possible methods were examined and, based on the results, it was decided to implement a new severe accident spray system (SASS). The SASS allows steam condensation using cooled water droplets injection into the atmosphere of the localisation shaft of the containment. The water (condensate and droplets) is collected in the sump and driven by the pumps to be cooled by air-cooled heat exchangers located outside the localisation shaft (on the roof). Operation of the SASS ensures the conditions for external cooling of the reactor vessel, the leaching of fission products from the containment atmosphere and the reduction of the containment pressure and temperature to achieve a safe state. Adequacy of the selected system design is supported also by the results of L1 and L2 PSA. The proposed system is expected to be able to operate for up to a year, if needed.

Paper #1 190: **"Fully passive long-term containment cooling of VVER440-V213 reactors", Hupp, M., M. Braun, N. Losch, Framatome, Germany**

The paper presented an alternative technical solution to ensure safe long-term containment heat removal following a severe accident in a VVER-440/V213. Instead of using any kind of active spray system to spray water to the containment atmosphere, Framatome GmbH proposed a containment cooling condenser (CCC). It is designed as a passive system that circulates water from a back-cooling pool (e.g. the shielding/storage pool) into the containment through tilted tube-bundle heat exchanger where steam is condensed on the outer surface of the tubes. The phase-change within the heat exchanger tubes drives a natural two-phase convection through the tube-bundle. The steam leaving the heat exchanger is then condensed in a back-cooling pool. The CCC system was tested at Framatome's experimental facilities and its performance in a VVER-440/V213 containment during severe accident was simulated using MELCOR. The results show that a moderately sized system should be sufficient for VVER-440/V213 needs.

Papers #1 210 and #1 220:

"Study of the additional upgrades implementation for long-term heat removal from the containment of nuclear power plants with VVER-1 000 reactor", Kocharyants, O., O. Mazurok, V. Ivanov, O. Mihaylenko, ESG, Ukraine

"Study of the additional upgrades implementation for long-term heat removal from the confinement of nuclear power plants with VVER-440 reactor", Kocharyants, O., O. Mazurok, V. Ivanov, O. Mihaylenko, ESG, Ukraine

These two papers discussed the preliminary results of investigations on possible technical solutions for containment long-term cooling following severe accidents in a VVER-1 000 (Zaporizhzhia Nuclear Power Plant Unit 1) and VVER-440/V213 (Rivne Nuclear Power Plant Units 1 and 2). All Ukrainian VVER-1 000 reactors are equipped (or being equipped) with filter containment venting systems (FCVS) and PARs. With operation of FCVS and PARs during a severe accident, the integrity of the containment is ensured for 72 hours. To keep its integrity for a longer period, a long-term containment cooling system (LTCCS) is needed. A LTCCS concept with turbine driven pumps, in-containment heat exchanger and spray nozzles (like the one proposed in RCCS-2 021-1 170 and RCCS-2 021-1 340) was analysed with MELCOR, considering variability in the FCVS operation regime. The effect of simultaneous FCVS and LTCCS operation on containment behaviour should be further investigated to avoid the danger of deep negative pressures after loss of non-condensable gases due to FCVS operation.

For the VVER-440/V213, two concepts of long-term containment heat removal were investigated to resolve the concern of slow pressurisation inside the containment following external cooling of the reactor vessel: (i) a passive containment cooling condenser system similar to the one described in RCCS-2021-1 190; and (ii) an externally driven turbo-pump to circulate the sump water through a heat exchanger and sprayers (similar to the one discussed in RCCS-2021-1 170). The advantages and disadvantages of both systems were compared and investigated through analyses with MELCOR code. Based on preliminary results it was judged that the externally driven turbo-pump system with sprays is more adequate for a long-term containment cooling system (LTCCS). The preliminary MELCOR simulations demonstrated that the selected system would ensure reasonable temperature and pressure transients within the containment following the severe accident.

Paper # 1 340: **"Qualification and testing of a water turbine driven pump for use in a long-term containment cooling system", Kostov, E., A. Pérez-Salado Kamps, C. Hartmann, D. Lauer, D. Zeniuk1, and H. Troselius, Westinghouse and KSB SE & Co. KGaA, Germany**

The paper presented a new water turbo-driven pump developed by Westinghouse Electric, Germany, to ensure long-term containment cooling. The main concept is similar to the one to be used in RCCS-2 021-1 170 and one suggested for use in RCCS-2 021-1 220. The authors summarised the pump qualifications and results of the performed tests. The results suggest that the system allows removing the decay heat of well more than 6.1 MW from the VVER-440 containment during a severe accident with core melt and in-vessel melt retention (IVMR) available. The pump can operate for prolonged periods of time (over six months). The pump has been designed and tested to be qualified for operation under severe accident conditions together with the pump manufacturer KSB.

General conclusions and recommendations for Topic 2.

The Topic 2 technical session was dedicated to the new systems and designs introduced to prevent the escalating of design basis accidents into severe accidents and/or to mitigate the consequences of severe accidents in nuclear power plants. The paper RCCS-2 021-1 160 described the severe accident mitigation systems of the new SMART reactor design (including IVMR through ERVC and an ancillary containment spray system to maintain containment integrity). The authors informed that, to verify effective operation of the proposed mitigation technologies, further investigation is needed that should comprise:

- experiments on the IVR-ERVC, focusing on two-phase natural circulation between the outer reactor vessel wall and the vessel insulation;
- a detailed analysis of the hydrogen behaviour in the lower containment area of the SMART, due to large subdivision in this part of the containment.

All other contributions focused on new systems designed and/or tested to ensure long-term containment cooling of existing VVER plants (VVER-440/V213 and VVER-1 000). The sump drains and filters to prevent filter clogging were reconstructed earlier (in the late 1990s) in these plants, thus any potential of sump clogging is considered as resolved.

Four of those other contributions investigated solutions that are under implementation or consideration for the following nuclear power plants: Dukovany, Paks, Rivne and Zaporizhzhia.

Apparently, all VVER-440/V213 aim at preventing failure of a reactor vessel by IVMR-ERVC, but solutions for containment cooling differ. For VVER-1 000, however, corium will be cooled ex-vessel and the LTCCS concept with in-containment heat exchanger and turbine driven spray pumps has shown successful long-term containment heat removal capability. Two of those other contributions discussed development, qualification and testing performed for one of the following long-term containment cooling systems each: containment cooling condenser (a passive system) and turbo-driven pump with combination of heat exchanger.

For all nuclear power plants discussed during this session, the use of a spray system was the selected method to extract heat from the containment atmosphere.

The main discussions concerned some technical details of the SMART design, the choice of SA scenarios assumed for the modelling activities, remaining items to be considered and/or resolved before long-term containment cooling solutions can be applied in nuclear power plants and the extent to which the approach to the justification of certain aspects of the proper operation of the proposed long-term containment cooling solutions can and/or should be analytical or experimental.

Investigations related to reliability of new heat evacuation systems, including passive ones, mostly focus on heat evacuation capacity, including for long periods of time. There is a need to review and assess boundary conditions that could affect these systems' functioning in a severe accident and conduct appropriate reliability studies.

From works presented within Topic 2 it can be concluded that various aspects are to be carefully investigated to ensure successful implementation of any potential technical solution for a long-term containment heat removal system. Among others, these aspect cover:

- an estimation of available time margins before reaching the ultimate limits of the containment strength (for overpressure as well as under-pressure);

- an estimation of available time margins before reaching limits related to coolant inventory, which are critical for maintaining circulation, either natural or forced, as applicable, in the cooling loops;
- careful assessment of residual heat to be removed from the containment in then long-term;
- the capacity of equipment used (pumps, heat exchanger, etc.) and parameters of coolant (temperature, flow rate, etc.) required;
- consideration of harsh environmental conditions to be withstood by the proposed heat removal system.

The design of any proposed long-term heat removal system should comply with the following conditions:

- the system should be simple and able to operate under circumstances of a severe accident (contaminated atmosphere and water, higher temperatures and pressures, etc.), meaning its appropriate qualification is needed;
- the system should perform as expected even under extreme weather conditions;
- the system should be qualified to withstand seismic events in a similar manner as the containment;
- it must not affect normal plant operation nor the operation of emergency systems;
- it must work together with the other severe accident management systems.

2.2.3. Topic 3: Sump clogging issues

Topic Chairs: B. Tourniaire (EDF, France) and I. Vicena (VUEZ, Slovak Republic)

The Topic 3 session was dedicated to sump clogging issues. The presentations covered a broad spectrum of activities including experimental activities on small and large scales as well as the French TSO conclusions related to long-term sump performance and new REPAS application on sump clogging. The presentations are listed in Table 2.4.

Table 2.4. Topic 3 presentations list

#1 290	VIKTORIA experiments on sump filtration during Loss of Coolant Accident G. REPETTO, B. MIGOT (IRSN, France), V. SOLTESZ (VUEZ, Slovak Republic)
#1 300	Lifetime extension of 900MWe nuclear power plants: French TSO main conclusions regarding long term sump performance after a loss of coolant J.F. TRIGEOL, E. PARIAUD, E. DIXNEUF, A. DUPRAT, G. REPPETO (IRSN, France)
#1 310	Analytical tests to study potential chemical effects on sump clogging C. ALVAREZ, W. LE SAUX, M. PRADIER, L. BOSLAND, L. CANTREL, M.O. SIMONOT (IRSN, France)
#1 320	Lifetime extension of French 900 MWe nuclear power plants: French TSO main conclusions regarding long term sump performance during a severe accident G. CENERINO, L. CANTREL (IRSN,France), V. SOLTESZ (VUEZ, Slovak Republic)
#1 330	REPAS application on sump clogging issue in long term core cooling A. BERSANO, G. AGNELLO, F. D'AURIA, E. ZIO, F. MASCARI (ENEA, Italy)

The presentations covered new experimental results and their use in safety assessment process. Below is a summary of each presentation.

Paper #1 290: **"VIKTORIA experiments on sump filtration during loss of coolant accident", Repetto, G., B. Migot, IRSN (France), V. Soltesz (VUEZ, Slovak Republic)**

Paper #1 300: **"Lifetime extension of 900MWe nuclear power plants: French TSO main conclusions regarding long-term sump performance after a loss of coolant", Trigeol, J.F., E. Pariaud, E. Dixneuf, A. Duprat, G. Reppeto, IRSN (France)**

Paper #1 310: **"Analytical tests to study potential chemical effects on sump clogging", Alvarez, C., W. Le Saux, M. Pradier, L. Bosland, L. Cantrel, M.O. Simonot IRSN (France)**

Paper #1 320: **"Lifetime extension of French 900 MWe nuclear power plants: French TSO main conclusions regarding long term sump performance during a severe accident", Cenerino, G., L. Cantrel, IRSN (France), V. Soltesz (VUEZ, Slovak Republic)**

Two papers are related to experimental investigations and two to safety assessments in French nuclear power plants.

In the first two papers (#1 290 and #1 310), the presenters described the experimental programmes that have been performed to study the phenomena involved in the filter's sump clogging issue and discussed the conclusions drawn from these experimental programmes in the frame of the fourth periodic French 900 MWe safety assessment.

The adopted experimental approach includes both global tests performed in the VIKTORIA facility (study of sump filtration with representative debris including insulation materials and painting chips) and separate effect tests performed in the COPIN (Clogging Of sumps In Nuclear industry) facility.

In both cases, LOCA and severe accidents representative conditions were looked for. This includes thermohydraulic parameters such as temperature, filters design, flow rates, size of the debris and chemical parameters such as water composition of the sumps (boric acid and soda concentrations in water) and chemical compositions of the debris (fibres of materials used in nuclear power plant are selected).

The global VIKTORIA experiments have shown the influence of several of those parameters on the sump filters' performance and to first order the design of the filter (rectangular cartridges or planar grid types). The separate effects tests aim at better identifying and understanding the chemical phenomena which could be involved in the formation of gels and precipitates that could contribute to the clogging of filters. Since performed in a medium-scale facility, new phenomena such as the effect of radiation leading to the radiolysis of gas and water are planned to be studied in the frame of the analysis of the severe accident case.

The review of these two papers indicates that a significant experimental database is available and could be used to study the phenomena involved in sump clogging issues. The COPIN facility can provide analytical data to better understand each phenomenon and the VIKTORIA tests make it possible to study the coupling of these phenomena and their consequence on representative filters. The facilities are in operation so that new experiments in conditions different from those investigated until now (e.g. using new filters) can probably be performed if needed.

The two papers (#1 300 and #1 320) present the approach which was followed to assess the risk of filter sump clogging at the reactor scale in the frame of the fourth periodic safety assessment of the French 900 MWe power reactor and the conclusions of the assessment which was provided by IRSN in 2020. The paper (#1 320) describes what was done for LOCA accidents and the other paper (#1 300) what is currently performed for severe accident conditions. Both works mainly rely on the results of the global experiments performed on the VIKTORIA facility and new tests being planned on the COPIN facility for severe accident conditions. In case of LOCA conditions, the safety assessment by IRSN has led to significant modifications on 900 MWe French nuclear power plants such as the replacement of fibrous insulation materials at steam generator bottom, the removal of all reflective metal insulation and a reduction in the quantity of fibre insulation materials. Thus, these papers clearly illustrate how R&D results have impacted the design of French nuclear power plants to improve safety.

Paper #1 330: **"REPAS application on sump clogging issue in long term core", Bersano, A., G. Agnello, F. D'auria, E. Zio, F. Mascari (ENEA, Italy)**

The paper describes a methodology and an application to characterise the sump clogging issue in case of long-term cooling. It is based on the use of the REPAS methodology (developed in the past by ENEA University of Pisa, Polytechnic of Milan and University of Rome) and of the TRACE code (best estimate thermal-hydraulic system code) developed by US NRC. The main purpose of the paper is to show that the REPAS methodology, which was developed to study the reliability of passive systems, can also be used for active systems. The REPAS methodology considers both deterministic and probabilistic cases.

General conclusions and recommendations for Topic 3

Papers on experimental investigations address both separate and integral tests relevant for clogging issues. The obtained experimental results are useful for model development and validation that could be used for safety assessment purposes.

However, the modelling aspect is not addressed. In the case of the filter sump clogging issue, none of the papers mention the development or use of modelling, neither as a natural continuation of the experimental work nor as a support activity to the safety assessment. Since the discussions highlight that there is no consensus on the way to extrapolate these results to the reactor case for safety assessment, this raises several points such as:

1. The best way to scale the global experiments for a relevant extrapolation to the reactor case in case safety assessments would only be based on such extrapolation.
2. The need to develop modelling activities to help the extrapolation to the reactor case.
3. The identification of separate effect tests in support of the understanding of global experiments and modelling activities.
4. The way to use the results of separate effect tests given the strong coupling with other phenomena (e.g. reaction of major ions in solution [Ca, Si, Na and B] with debris and waters in reactor's sump).

2.2.4. Topic 4: Debris formation and chemical conditions in cooling waters

Topic Chairs: D. Jacquemain (NEA) and S. Bechta (KTH, Sweden)

The Topic 4 session (debris formation and chemical conditions in cooling waters) covered two main items:

- Item 1: Fuel debris formation and coolability (four papers); and
- Item 2: Chemical conditions in cooling waters (two papers).

The presentations are listed in Table 2.5.

Table 2.5. Topic 3 presentations list

#1 240	Extension of debris bed cooling evaluation code DPCOOL for evaluating uncertainties in long-term debris coolability A.HOTTA, W. KIKUCHI (NRA, Japan)
#1 230	Extension of molten jet breakup evaluation code JBREAK by improving droplet agglomeration model and validation based on DEFOR-A test W. KIKUCHI, A. HOTTA, A. MORITA (NRA, Japan)
#1 260	Development of evaluation framework for ex-vessel core coolability T. MATSUMOTO, Y. IWASAWA, T. SUGIYAMA (JAEA, Japan)
#1 270	Simulation of Potential Ex-Vessel Corium Behaviour during a CANDU Reactor Severe Accident using MELTSPREAD and CORQUENCH Codes F. FORGUES, J. SPENCER (CNL, Canada)
#1 250	Assessment of Boric Acid Compounds Solubility at VVER's Primary Coolant Conditions N.L. KHARITONOVA, S.A. GURBANOVA (SEC NRS, Russia)
#1 280	Radiation-chemical consideration of effect of seawater for cooling reactor cores at severe accident on structural material corrosion R. NAGAISHI, T. ITO, R. KUWANO (JAEA, Japan)

For both items, a summary of each presentation is given below.

Item 1: Fuel debris formation and coolability

Paper #1 240: **"Extension of debris bed cooling evaluation code DPCOOL for evaluating uncertainties in long-term debris coolability", A. Hotta and W. Kikuchi (NRA)**

Demonstration of successful long-term debris cooling and in-vessel or ex-vessel corium stabilisation includes sufficient confirmation that the decay heat is stably removed, and the geometry of the debris bed will not change remarkably so that the debris bed is securely confined even if there is a local dry out or molten pool for a certain period. To consider the morphology and other properties of the debris bed, it is necessary to know how the debris bed is formed, including the effects of interactions with the coolant under a wide range of cooling phenomena, fragmentation, and solidification of oxidic and metallic melts undergoing liquid immiscibility.

Based on this background, the NRA conducts a collaborative research project and tries to expand the knowledge of the important phenomena and uncertainty factors relevant in the evaluation of the long-term cooling. In parallel, it is developing a code system, THERMOS, for the evaluation of ex-vessel debris bed formation and cooling. The code consists of four modules, JBREAK, MSPREAD, DPCOOL and REMELT.

DPCOOL is a module that deals with the cooling of heterogeneously configured debris beds submerged in the water pool. The three-dimensional two-fluid model is developed for the water pool region, whereas the model of volumetrically heated porous medium cooled by two-phase flow is used for the debris bed region. The code has been validated based on non-heated and heated homogeneous particle layer experiments in the air-water and steam-water conditions.

Currently, the code has been extended to deal with (a) multi-component heterogeneous debris beds, (b) the heat transfer coefficient model between the structure walls, the particles, and the water pool, and (c) the multi-dimensional particle self-levelling. Herein, these newly implemented features were introduced with the equations including the variables and parameters that represent their theoretical basis. Validations results have been discussed based on three experiments: VTT's COOLOCE-13, IKE's DEBRIS with the central downcomer, and KTH's PDS-C.

Priorities for future DPCOOL improvement are highlighted in the paper considering the analysis results, and potential experimental/modelling uncertainty factors.

An uncertainty factor common to all three experiments is attributed as the two-phase flow pattern both in the bulk and near-wall region. The two-phase flow pattern map for saturated water at 0.1 MPa is discussed, developed and implemented in DPCOOL to connect bubbly, slug or annular flow regimes with characteristic particle diameters and void fractions.

The contact heat transfer coefficient between the debris and the structure can be affected by high temperature mechanical interactions between partially liquid or viscoplastic particles and metal surfaces. Such interactions, which can promote contact heat transfer, are currently studied by the paper authors in collaboration with CRIEPI. In this experiment, a mock-up particle debris – wall contact section will be shaped by the 3D printer. The heat transfer components by contact and coolant convection are measured with varying the contact area and visualising the two-phase flow pattern near the contact surface.

The following model improvements are recognised from the viewpoint of the long-term debris cooling.

(i) It is presently assumed that all debris bed pores that contribute to the porosity form the flow network that relates to the pool region. The importance of the closed porosity has been recognised and measured in DEFOR-A and PULiMS conducted at KTH. The closed porosity should be modelled as not accessible for two phase flow, which can reduce coolability. The closed porosity and cracks can also influence mechanical properties of solidified particles, cakes and crusts isolating melt from the coolant. Degradation of mechanical properties may play an important role in the short term, e.g. considering possible fracture or melting through the crust that holds up the molten pool, or in the long term, e.g. considering secondary fragmentation of debris bed particles after long-term leaching and corrosion in water/air.

(ii) When evaluating the heat removal by air flow as the ultimate form of long-term cooling, it is necessary to consider the heat transfer between particles and air that does not accompany the dry out or the two-phase flow pattern.

(iii) The metal oxidation model is not currently modelled but must be implemented if necessary for judging success of the air cooling.

(iv) It is necessary to consider remelting and melt progression inside the debris bed, including formation of the local molten pool, melt infiltration, blockage of the flow path of the molten phase, etc. The interface for coupling analysis with THERMOS-REMELT is currently under development to realise this goal.

Finally, it is concluded that the model extension and the three validation results described above show that DPCOOL can evaluate the heat removal in actual complex debris configurations. These results gave the authors the expectation that coupling with other THERMOS modules would be a useful analysis tool in evaluating the long-term debris cooling under various accident scenarios and investigating TEPCO's Fukushima Daiichi Nuclear Power Station accident.

Paper #1 260: **"Development of evaluation framework for ex-vessel core coolability", Matsumoto, T., Y. Iwasawa and T. Sugiyama (JAEA)**

JAEA has developed an approach involving probabilistic analyses to assess the success of fuel debris cooling ex-vessel during a severe accident. It was applied to a BWR with a wet cavity strategy.

Firstly, probabilistic distributions of the melt conditions (e.g. melt temperature, cross sectional area of RPV breach, duration and flowrate of melt discharge, total mass of discharged molten material), when ejected from the RPV, were obtained by iterative analyses with the MELCOR code. Three hundred input parameter sets were generated by Latin hypercube sampling (LHS) for five parameters relating with the in-vessel degradation phenomena (four related to melts relocation and one to the RPV failure temperature).

These probabilistic distributions were then used to generate for three different water depths in the cavity (0.5, 1 and 2 metres) 59 input cases for the probabilistic analyses with the JASMINE code, which calculates, in a second step, the formation of the accumulated debris on the floor of the cavity, predicting the melt spreading on the floor and the mass fraction of agglomerated debris. Using this approach, the configuration of the accumulated debris in the cavity (shape and height) can be calculated addressing variability on the melt discharge and on the debris formation.

The debris coolability is considered successful when the accumulated debris height remains below a maximum coolable height and when the debris temperature at the boundary with concrete remains below the concrete ablation temperature. All cases studied, except one, resulted in coolable configurations.

The authors recognised that the results may be refined by further investigating the effects of melt temperature on melt spreading and debris formation in the cavity and that results are very much dependent on the debris agglomeration model implemented in the JASMINE code.

Paper #1 230: **"Extension of molten jet breakup evaluation code JBREAK by improving droplet agglomeration model and validation based on DEFOR A test", W. Kikuchi and A. Hotta (NRA))**

The NRA is developing the high-resolution modular code system THERMOS for quantitative evaluations of the debris bed formation and cooling and related uncertainties, in which JBREAK is a major module.

The JBREAK solves the molten jet breakup, formation and cooling of debris beds based on a three-dimensional interface tracking method. Droplets are generated from the molten jet using an empirically fitted size distribution. Droplets are then cooled and solidified during sedimentation in the water pool and deposited on the cavity floor while a part of them can be bonded together during agglomeration.

The Kelvin Helmholtz model is applied as a droplet entrainment model for the side surface while the Rayleigh Taylor model is applied for the tip. Furthermore, the droplet settling trajectories are solved by the random walk method by considering the turbulent flow field, interactions with different droplet groups, and collisions against walls and obstacles.

Droplet agglomeration is expressed based on the semi-empirical model, where the agglomeration mass fraction is expressed as a function of the partially solidified droplet mass fraction. The model was modified by considering the mass and energy balance of newly settling droplets and already settled particle debris at different elevations in the water pool.

Validation of models is presented for three experiments selected in the DEFOR A test series recently performed at KTH: A24, A26 and A27. The three experiments include a wide range of simulant melt superheat and flow rate of the molten jet.

In the A24 and A26 tests, the characteristic particle size becomes smaller with an increase of the distance from the water pool surface to the particle catcher. However, the opposite tendency is observed in the case of A27, which was performed under the highest melt superheat. It can be inferred that the diameter of the solidified particles increases with the distance in A27 because high temperature droplets spread on the catchers and form agglomerated debris that includes small particles. This assumption was supported by the debris bed shape in the catchers measured by the laser scanner.

Since JBREAK does not consider melt spreading in the catchers, the code predicts a steeper slope near the impinging point and could not predict the specific shape of the debris bed sufficiently well. It can be deduced that this modelling drawback also led to an uncertainty in the agglomeration fraction.

The authors noted that the predictability of the number of particles reaching the pool floor can be improved by considering the disturbance due to boiling near the melt and correction of the turbulent flow effect. It is important to improve the model from a comprehensive point of view, considering the synergistic effects of the various phenomena described so far.

Paper #1 270: **"Simulation of potential ex vessel corium behaviour during a CANDU reactor severe accident using MELTSPREAD and CORQUENCH codes", F. Forgues and J. Spencer (CNL)**

The authors from the CNL address corium behaviour after a failure of in-vessel retention (IVR) during a hypothetical severe accident scenario in a pressurised heavy water reactor (PHWR), such as the CANDU reactor. If IVR is lost, corium could relocate onto the concrete floor of the calandria vault beneath the calandria vessel. To develop understanding of ex-vessel corium behaviour and evaluate the coolability of core debris, the paper investigates corium spreading behaviour, molten core concrete interaction (MCCI), and the solidification process.

Two computer codes developed by the Argonne National Laboratory are used:

- MELTSPREAD-3.0 1D FEM to calculate the gravity driven transient spreading of core melt after the calandria vessel failure and associated melt release ex-vessel. The input parameters of MELTSPREAD are widely customisable by the user via an input file that includes a series of test specifications and modelling assumptions such as: (i) melt pour conditions (flow rate, temperature, corium composition including material in oxidic e.g. UO_2 and ZrO_2 and metallic e.g. Zr phases), (ii) cavity geometry (complex and customisable nodalisation of spreading geometry), (iii) cavity flooding information (dry vs pre-existing water layer), (iv) cavity atmospheric conditions (temperature, pressure) and (v) material of spreading surface (steel, concrete or user specified materials such as ceramic).

- CORQUENCH-4.1 to capture MCCI and the debris cooling behaviour. The end conditions of MELTSPREAD simulations are provided as an input in the CORQUENCH to evaluate the long-term core concrete interaction behaviour following the transient spreading stage. Results of the CORQUENCH simulation are not yet available but are currently part of ongoing work.

Numerical results are presented and discussed for different corium oxidation levels (from 22 to 100 rel%) and melt flow rates (constant flow rate of 1 800 kg/s for 50 seconds in the first set and varied flow rate of 500 kg/s for 200 seconds, 1 000 kg/s for 100 seconds and 2 000 kg/s for 50 seconds considering the fixed total mass of released corium as 100 t in the second set of calculations).

In this first set of test cases with constant melt flow rate, employing five different corium compositions with different levels of zirconium, the melt fully solidifies earlier for the corium with low zirconium oxidation, except for the fully oxidised corium composition, which solidifies much faster than all the other corium compositions.

In the second set of test cases with varied melt flow rate, it is visible that the slower the corium pour rate, the longer it takes for the melt to cool down (presumably due in large part to the geometry of the resulting melt configuration). Another interesting finding is that the initial rate of ablation is higher for higher flow rates near the pour location, but ablation persists for longer periods with slower pours, ultimately resulting in more integral mass of material ablated.

Future work is directed towards utilisation of the end conditions of the MELTSPREAD simulation as input for the CORQUENCH code to evaluate the long-term core concrete interaction behaviour following the transient spreading stage. Different spreading geometries, pouring and cavity conditions will also be explored.

As finally highlighted, this is the first study of ex-vessel corium behaviour in CANDU reactors reported using MELTSPREAD and constitutes an advancement in the knowledge base pertaining to corium behaviour and long-term coolability of debris in a hypothetical severe accident scenario.

General discussion on fuel debris formation and coolability

Four papers in the session addressed melt jet fragmentation in water (melt from fusion of in-core material), fuel debris formation and coolability, mainly during the ex-vessel phase of a severe accident. No contributions on other types of debris (e.g. resulting from degradation of in-containment material) which could affect long-term cooling were provided for the session.

It was discussed and recognised that significant modelling uncertainties still exist in the prediction of formed debris configurations and debris bed properties due to:

- Uncertainties accumulated during in-vessel accident progression modelling and specifically modelling of RPV failure mode, time and characteristics.
- Variability in melt release scenario, e.g. one or more pours with different locations and melt composition, melt pour flow rate, temperature and composition vs time, variability in cavity configurations and system parameters, e.g. possible pressure escalation/oscillations – even if energetic steam explosion does not take place.
- The fact that debris formation and agglomeration models, as well as debris coolability ones, are not completely validated on representative tests reproducing full ranges of major parameter variation in the conditions expected. There is visible lack of experimental data for that.
- Variability and complexity of melt compositions and poor knowledge of the properties of liquid and solid corium.

Enhancing probabilistic analyses could help in better assessing debris formation and coolability ex-vessel.

Long-term aspects of fuel debris bed behaviour and coolability appeared to be novel and quite challenging applications for the models and codes traditionally developed in reactor severe accident research. Regarding long-term cooling, no contribution addressed the potential effects of the presence of other types of debris and precipitates/sediments in cooling waters that could potentially affect the fuel debris cooling (e.g. by blocking porosities in accumulated fuel debris) but also contribute to blocking/clogging cooling systems, as discussed in Session 3.

For several important phenomena, experimental data do not cover all needs of model development and code validation. Several new important directions of experimental research are proposed in the papers on model development; some of the suggested experiments are already initiated.

Models and codes developed in different organisations give a good starting point for advanced ex-vessel analyses, but it is still challenging to predict cooling or remelting of accumulated debris and MCCI arrest.

The NEA ROSAU project should provide new data, reducing some of these uncertainties.

Addressing the full span of debris/sediments that can be formed under severe accident conditions appears very challenging and further assessment of possible debris sources (e.g. heat insulation material, paints and composite liners) and their potential detrimental effects on the long-term management of an accident would be needed to prioritise any significant R&D in the field. Explorations inside the containment of the three damaged units at Fukushima Daiichi have evidenced the presence of large amounts of ununiformly distributed structural debris and sediments originating from various sources. The extent, nature and characteristics of debris formed during a severe accident will anyway significantly depend on the accident progression in and ex-vessel and reached conditions (e.g. temperature, radiation dose, chemical conditions).

Another open question is connected with the fuel debris ageing with very high accumulated radiation dose in water/air environment, which can result in accelerated leaching of specific elements (including FPs), corrosion damage and degradation of material properties, all contributing into the risk of debris fragmentation, formation of fine particles, and associated secondary FP releases, but also challenges to predict and to manage debris bed macro- and micro-structures. Some activities on FP leaching studies employing synthetic corium samples produced in the labs and plant samples from Chernobyl are in progress in the NEA TCOFF project, the second phase of which is expected to start soon, but available results mostly cover FP releases during leaching, i.e. with the dissolved species, not corium ageing and degradation of material properties.

Item 2: Chemical conditions in cooling waters

Paper #1250: **"Assessment of boric acid compounds solubility at VVER's primary coolant conditions", N.L. Kharitonova and S.A. Gurbanova (SEC NRS, MEPhI)**

SEC NRS and MEPhI provided a contribution related to assessing the risk of boric acid compound precipitation/crystallisation in the core and RCS during an accident. Boric acid precipitation could affect long-term cooling by reducing or blocking cooling flows. Higher concentrations of boric acid may be desirable in operation with longer cycles and for sump pH control in accidents (e.g. maintaining an alkaline pH to reduce iodine release) but the related precipitation risk needs to be assessed.

The contribution discussed data on the stability and solubility of major borate compounds for RCS-relevant conditions and the status of models of key chemical reactions for precipitation of borate compounds with alkali metals (Na, Li, K), which are additives in VVER.

The authors recognise that though data and models are available to address borate compounds precipitation in well-characterised conditions, more complex chemical conditions may be expected in accidents and chemical reactions with other compounds may affect precipitation.

An approach could be to list major chemical compounds which precipitation would have to be considered in priority and modelled for accidental conditions. There could be co-precipitation of various borate compounds.

Such knowledge should also be useful to develop technical solutions (e.g. control of chemical conditions) to prevent possible blocking of core and RCS cooling by precipitation in case of accidents.

Paper #1 280: **"Radiation-chemical consideration of effect of seawater for cooling reactor cores at severe accident on structural material corrosion", Nagaishi, R., T. Ito and R. Kuwano (JAEA)**

JAEA provided a contribution related to the effects of the use of seawater as coolant during the Fukushima Daiichi accident. The addition of seawater is affecting H_2 generation by radiolysis and structural material corrosion with a particular effect of Cl^- and Br^- ions under radiation (radiation-catalysed corrosion).

Experimental data were gained on seawater radiolysis and radiolytic corrosion of structure material. H_2O_2, HClO and ClO_2^- are major oxidant species promoting the corrosion.

Authors consider, based on the obtained data, that at Fukushima Daiichi, before desalination, HClO and ClO_2^- mostly contributed to corrosion. After desalination, H_2O_2 would be the predominant species affecting corrosion. They also consider the desalination should have been effective in limiting the radiolytic corrosion.

General discussion on chemical conditions in cooling waters

Severe accident chemical conditions being extremely complex to represent in detail, a key question is how to prioritise R&D efforts to investigate possible chemical effects on long-term cooling in a severe accident as many materials can contribute to the formation of sediments/particulates, precipitates and result in clogging, e.g. related to:

- degradation of heat insulation and organic materials (e.g. paints, composite liners, cables), degradation/corrosion of structural material from normal operation/accident conditions, contribution to debris/particulate formation of fuel debris:
- effects of chemical additives in coolant waters;
- effects of some fission products (e.g. iodine) on radio-catalysis of corrosion of structural materials.

In this area, it appears challenging to define what research and chemical models are needed as a priority to further assess how chemical conditions can affect precipitate formation and clogging risks during accidents. As discussed earlier, this may also depend on the accident development and the conditions reached in and ex-vessel.

Countermeasures, such as "control" of chemical conditions in cooling waters, have been applied in some nuclear power plants to limit corrosion and precipitation risks. It would be appropriate to share best practice and experience in the field to eventually establish guidance and recommendations.

3. Conclusions and recommendations

3.1. Conclusions

The specialist workshop on "Reactor Core and Containment Cooling Systems: Long-Term Management and Reliability" (RCCS-2021) provided an opportunity to discuss recent developments and practices adopted for core and containment cooling for design basis accidents and severe accidents. The workshop primarily focused on LWR and PHWR technologies and the reliability of cooling systems and components to support the continued safety functions in the long-term accident conditions.

Experts from the international nuclear community provided updates on the recent research developments to improve understanding of the technical bases related to maintaining the cooling function in accident scenarios in a nuclear power plant. The workshop addressed, among other aspects, two key issues that affect the reliability of these cooling systems: the degradation of systems or components in accident conditions, and the robustness of cooling systems related to clogging of the filters on systems supplying the water to the core and containment cooling systems.

From the presentations and discussions held during the workshop some conclusions may be drawn:

- **The technical bases and methodology to address the long-term reliability of cooling systems in LOCA scenarios, without significant core damage, are considered well established** with the determination of the upstream debris source term, of the sump filtration performance, of chemical effects, and of downstream debris and their potential impact on fuel and core cooling degradation. Further, limiting the debris sources (e.g. by reducing or eliminating material inventories such as fibre in containment to limit the level of debris being formed in accident scenarios) and enhancing the filtration capacities have been considered and implemented for several plants. In addition, several countries reported that additional measures have been considered to address the sump clogging issue.
- **The applicability and sufficiency of the LOCA technical bases and methodology to address the long-term reliability of cooling systems in severe accidents need to be assessed.** Debris sources, chemical effects, filtration capacities, erosion-corrosion effects and downstream effects must be re-evaluated for severe accident conditions and scenarios. The applicability of the method developed for LOCA conditions should be assessed. Lessons being learnt from the Fukushima Daiichi accident can be informative, although the need to consider a variety of designs and accident scenarios requires additional insights, which is likely to benefit from a well-defined and targeted collaborative research effort in the area. A ranking exercise could be conducted to prioritise phenomena to investigate and to establish research plans in the area with the definition of experimental investigations of generic interest considering the variability of designs and accident scenarios. The ranking exercise would also be valuable for assessing the applicability of the LOCA method to severe accidents.
- **Safety enhancements and new systems have been added to improve long-term core and containment cooling in severe accident scenarios**. Among the latter, the use of passive systems and portable equipment (stored on site or brought to site) is a common feature. The Fukushima Daiichi accident sped up both the implementation of such systems and fostered design studies of new systems. It would be beneficial to further establish the technical bases for demonstrating the

long-term reliability of these systems in challenging severe accident conditions. This would support licensing of new designs.

- **Data collected from the Fukushima Daiichi accident have provided meaningful insights into long-term cooling during severe accidents.** The most relevant information includes the available time margins before reaching the ultimate limits of the containment strength (for overpressure as well as under-pressure), the coolant inventory, level, etc., the assessment of residual heat to be removed from the containment in long-term, the capacity of equipment used (pumps, heat exchanger, etc.) and coolant parameters (temperature, flow rate, etc.), or the consideration of a harsh environment to be withstood by heat removal systems. Preliminary information on debris formed during the accident has been shared in joint NEA, post-Fukushima projects such as PreADES and more information can be expected in the future with new debris sample analysis campaigns.
- **Substantial R&D work has been conducted for LOCA conditions, without significant core damage, to gain a better understanding of the reliability of long-term cooling systems with the development of dedicated experimental facilities.** These facilities provide opportunities for collaborative investigation of aspects related to demonstrating cooling systems' long-term reliability extending to severe accident conditions, addressing e.g. debris sources or chemical and irradiation effects. In addition, data sharing on equipment testing might help to focus the investigation targets and should be promoted. Currently, small and integral scale facilities are available in several laboratories to address open issues.

In addition, the workshop participants raised some important points that are considered key to address in order to support the reassessment of the safety of large and small (SMRs) nuclear power plants according to the updated IAEA requirements and recommendations:

- **Significant modelling uncertainties still exist in the prediction of formed core material debris configurations and debris bed properties in severe accidents.** Several reasons might be given, from the accumulation of uncertainties on knowledge of in-vessel and ex-vessel accident progression, to the limited knowledge of complex corium mixtures properties, which are strongly linked to melt composition variability. Variability in vessel rupture modes and in the transfer of core melt and debris from the vessel should also be considered. Dedicated collaborative research, such as the NEA ROSAU project, is ongoing to reduce related uncertainties. Model development to address long-term core debris bed coolability could benefit from collaborative efforts.
- **R&D efforts are needed to investigate possible combined chemical and radiation effects relevant for severe accident conditions.** Formation of debris/sediments/particles in cooling waters and their evolution with time, e.g. whether they remain in suspension or sediment, are expected to strongly depend on combined chemical and radiation effects. A limited number of investigations have addressed the combined effects of challenging chemical and radiation severe accident conditions. Existing irradiation facilities provide the opportunity to expand related testing.
- **Knowledge on materials should be extended to ensure the leak-tightness of containment (e.g. liners, seals) and to anticipate their mechanical response to anticipated prevailing conditions in severe accidents.** This would support enhancing structural mechanical assessments of important containment components.

- **The scaling from sump clogging separate and integral tests to reactor applications needs to be addressed.** The applicability of models developed and/or under development should be demonstrated by addressing scaling.

3.2. Recommendations

Following these conclusions, it appears essential to further establish in a collaborative process the technical bases needed to support the demonstration of the long-term reliability of cooling systems in severe accidents, including for new systems added after the Fukushima Daiichi accident and newly designed systems, including for SMR technologies.

It is recommended that a ranking exercise be conducted in prioritising the phenomena to investigate, establish research plans in the area with the definition of experimental investigations, and assess whether the methodology and calculation tools developed for LOCA accidents without significant core damage are applicable for severe accidents or need to be completed. In this exercise, the following aspects would need to be considered:

- The potential debris sources for severe accidents, e.g. considering the degradation of heat insulation and organic materials (e.g. paints, composite liners, cables), the degradation/corrosion of structural materials from normal operation and accident conditions, and the contribution to debris/particle formation of core debris. Learnings from the Fukushima Daiichi accident would be valuable, though variability in accident scenarios and systems designs would have to be addressed.
- The formation of chemical precipitates from non-metallic insulation debris including thermal effect, buffer composition, pH effect, ionic strength, effect of chemical additives.
- The possible combination of corrosion and chemical and radiation effects on the structural materials and components of cooling systems and the possible effect on cooling systems.

Regarding experimental investigations, several experimental facilities have been developed and expertise has been established for investigations mostly related to the long-term reliability of cooling systems in LOCA accidents without significant core damage. Other facilities, such as irradiators, can address more specifically the materials' response to prevailing severe accident conditions with combined chemistry and radiation effects. These facilities offer opportunities for extending investigations to severe accident conditions. There may be challenges related to defining experimental investigations of interest to various designs and accident scenarios, covering realistic severe accident conditions and, ultimately, addressing scaling. Small-scale experimental investigations of separate and combined effects of the corrosion, chemistry and radiation of materials on material debris may help addressing these challenges.

The level of data sharing internationally where design-specific investigations are conducted on a national basis – such as investigations of downstream effects, identified as all phenomena that apply to components after the water/debris mixture has passed the sump strainer – could be increased. This could reduce the need for additional testing requirements.

Moreover, the workshop highlighted the need for reliable containment component degradation assessment using validated structure mechanical computing tools and considering "short-term" ageing due to challenging exposure conditions in severe accidents. It also emphasised the need to better assess the risk of failure of the pressure-retaining boundary, including sealing, with consideration of material properties far from design conditions.

Regarding components contributing to ensuring containment leak tightness in a severe accident (e.g. liners, seals), it is recommended that ageing and severe accident communities connect to further assess existing technical bases that can be used to predict the components' response in-accident and the potential containment leak tightness failures. Collectively identifying key knowledge gaps and defining and developing approaches to address them from the perspective of both severe accidents and ageing management could support enhancements to the life management of these components and reduce risks of failure and of radioactive release in accidents.

Significant modelling uncertainties remain in the prediction of formed core material debris configurations and debris bed properties in severe accidents. Collaborative efforts could help in the development of models to address long-term core debris bed coolability and scaling from sump clogging separate and integral tests to reactor application.

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Annex A. Workshop agenda

DAY 1	Monday 18 October 2021	
13:00-13:30	*Chair : Ahmed BENTAIB, IRSN, France*	
	Welcome	Véronique ROUYER, Head, Division of Nuclear Safety Technology and Regulation, Nuclear Energy Agency, OECD Patrice GIORDANO, Nuclear Safety Research Director, IRSN, France Ivan VICENA – Head of Technological Systems Dpt. and Member of BoD, VÚEZ Luis E. Herranz (CIEMAT, Spain), Chair of OECD/NEA Working group of Analysis and Management of Accident (WGAMA)
13:30-14:30	**General Session- Keynote addresses** *Chairs: L. E. Herranz, A. Bentaib*	
RCCS-2021-1010	Jorge LUIS HERNANDEZ (IAEA)	Reactor core and containment cooling systems regulatory aspects from IAEA standards
RCCS-2021-1020	Bruno TOURNIAIRE (EDF, France)	Reactor core and containment cooling systems on French nuclear power plants
RCCS-2021-1030	Karim OSMAN (Ontario Power Generation, Canada)	SAMG long-term core and containment cooling strategy application and management in Canadian nuclear power plants
14:30-14:50	**Break**	
14:50-16:00	**Topic 3**	**Sump clogging issues** *Chairs: B. TOURNIAIRE, I. VICENA*
RCCS-2021-1290	G. REPETTO, B. MIGOT, V. SOLTESZ (IRSN, France)	VIKTORIA experiments on sump filtration during Loss of Coolant Accident
RCCS-2021-1300	J.F. TRIGEOL, E. PARIAUD, E. DIXNEUF, A. DUPRAT, G. REPPETO (IRSN, France)	Lifetime extension of 900MWe nuclear power plants: French TSO main conclusions regarding long term sump performance after a loss of coolant accident
RCCS-2021-1320	G. CENERINO, L. CANTREL, V. SOLTESZ (IRSN, France)	Lifetime extension of French 900 MWe nuclear power plants: French TSO main conclusions regarding long term sump performance during a severe accident
RCCS-2021-1330	A. BERSANO, G. AGNELLO, F. D'AURIA, E. ZIO, F. MASCARI (ENEA, Italy)	REPAS application on sump clogging issue in long term core cooling
RCCS-2021-1310	C. ALVAREZ, W. LE SAUX, M. PRADIER, L. BOSLAND, L. CANTREL, M.O. SIMONOT (IRSN, France)	Analytical tests to study potential chemical effects on sump clogging

Overall Q/A		
16:00	**SEMINAR DAY 1 CLOSES**	
DAY 2	**Tuesday 19 October 2021**	
13:00-13:40	**General session- Keynote addresses** *Chairs: L. E. Herranz, A. Bentaib*	
RCCS-2021-1040	P. JIANG (CNPRI, China)	Long term management strategies of containment pressure control during severe accidents
RCCS-2021-1050	R. BUNT (Southern Nuclear, United States)	Reactor core and containment cooling issues in DBA and SA differences – BWR
13:40-13:50	**Break**	
13:50-15:10	**Topic 2**	**New systems and designs** *Chairs: L. KUBISOVA, M. SHAWKAT, T. VAN ROMPUY*
RCCS-2021-1160	R. PARK, S. KIM (KAERI, South Korea)	Development of severe accident mitigation system and technology for Korean small integral reactor
RCCS-2021-1170	P. MATEJOVIC, M. BARNAK, Z. TUMA (Dukovany Nuclear Power Plant, Czechia)	VVER-440/V213 containment cooling system for severe accidents
RCCS-2021-1180	G. LAJTHA, E. TOTH, L. TARCZAL, T. SIKLOSSY (NUBIKI, Hungary)	Challenges in the Design of Long-Term Containment Cooling System at Paks Nuclear Power Plant
RCCS-2021-1190	M. HUPP, M. BRAUN, N. LOSCH (Framatome, Germany)	Fully passive long term containment cooling of VVER440-V213 reactors
RCCS-2021-1210	O. KOCHARYANTS, O. MAZUROK, V. IVANOV, O. MIHAYLENKO (ESG, Ukraine)	Study of the additional upgrades implementation for long-term heat removal from the containment of nuclear power plants with VVER-1000 reactor
RCCS-2021-1220	O. KOCHARYANTS, O. MAZUROK, V. IVANOV, O. MIHAYLENKO (ESG, Ukraine)	Study of the additional upgrades implementation for long-term heat removal from the confinement of nuclear power plants with VVER-440 reactor
RCCS-2021-1340	E. KOSTOV, A. PÉREZ-SALADO KAMPS, C. HARTMANN, D. LAUER, D. ZENIUK, AND H. TROSELIUS (Westinghouse, Germany)	Qualification and Testing of a Water Turbine Driven Pump for Use in a Long-Term Containment Cooling System

Overall Q/A		
15:10-15:20	**Break**	
15:20-16:00	**Topic 1**	**Assessment of accident management measures and strategies** *Chairs: A. TEHRANI, N. MESMOUS*
RCCS-2021-1110	K. HECKMANN, M. SONNENKALB, J. SIEVERS, T. STEINRÖTTER, S. PALAZZO, J. ARNDT, C. BLÄSIUS (GRS, Germany)	Review and Perspective of GRS' Long Term Accident Analyses with Consideration of Structure Mechanical Aspects
RCCS-2021-1120	L.E. HERRANZ, R. BOCANEGRA (CIEMAT, Spain)	Long term containment cooling in Fukushima Unit 1: consequences and sensitivity
RCCS-2021-1150	M. DI GIULI, P. DEJARDIN, M. AUGLAIRE (Tractebel Engineering (SA) Engie, BELGIUM)	Assessment of long-term containment conditions using MELCOR and ASTEC codes
Overall Q/A		
Virtual group pictures		
16:00	**SEMINAR DAY 2 CLOSES**	
DAY 3	**Wednesday 20 October 2021**	
13:00-13:40	**General session - Keynote addresses** *Chairs: L. E. Herranz, A. Bentaib*	
RCCS-2021-1060	H. ESMAILI (US NRC, United States)	Post-Fukushima Severe Accident Research Activities at NRC - Containment Vents and Hydrogen Control
RCCS-2021-1070	S. MIZOKAMI (TEPCO, Japan)	Lessons learned from 1F accident by focusing on the long term cooling
13:40-13:50	**Break**	
13:50-15:00	**Topic 4**	**Debris formation and chemical conditions in cooling waters** *Chairs: D. JACQUEMAIN, S. BECHTA*
RCCS-2021-1240	A.HOTTA, W. KIKUCHI (NRA, Japan)	Extension of debris bed cooling evaluation code DPCOOL for evaluating uncertainties in long-term debris coolability
RCCS-2021-1230	W. KIKUCHI, A. HOTTA, A. MORITA (NRA, Japan)	Extension of molten jet breakup evaluation code JBREAK by improving droplet agglomeration model and validation based on DEFOR-A test

RCCS-2021-1260	T. MATSUMOTO, Y. IWASAWA, T. SUGIYAMA (JAEA, Japan)	Development of evaluation framework for ex-vessel core coolability
RCCS-2021-1270	F. FORGUES, J. SPENCER (CNL, Canada)	Simulation of Potential Ex-Vessel Corium Behaviour during a CANDU Reactor Severe Accident using MELTSPREAD and CORQUENCH Codes
RCCS-2021-1250	N.L. KHARITONOVA, S.A. GURBANOVA (SEC NRS, Russia)	Assessment of Boric Acid Compounds Solubility at VVER's Primary Coolant Conditions
RCCS-2021-1280	R. NAGAISHI, T. ITO, R. KUWANO (JAEA, Japan)	Radiation-chemical consideration of effect of seawater for cooling reactor cores at severe accident on structural material corrosion
Overall Q/A		
14:40-15:00	**Break**	
15:00-16:00	**Closing remarks provided by the sessions Chairs**	
Workshop closes		