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In-vessel Melt Retention and Ex-vessel Corium Cooling

Summary of a Technical Meeting

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IN-VESSEL MELT RETENTION AND EX-VESSEL CORIUM COOLING

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SUMMARY OF A TECHNICAL MEETING

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FOREWORD

As a follow-up of the International Experts Meeting on Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant, held in Vienna from 16 to 20 February 2015, the IAEA organized a meeting on Post-Fukushima Research and Development Strategies and Priorities from 15 to 18 December 2015. The objective of that meeting was to provide a platform for experts from Member States and international organizations to exchange perspectives and information on strategies and priorities for R&D regarding the Fukushima Daiichi accident and severe accidents in general. The experts discussed R&D areas requiring further attention and the benefits of possible international cooperation, and confirmed that R&D on in-vessel melt retention and ex-vessel corium cooling/stabilization remained one of the highest priority areas. The participants emphasized that more phenomenological knowledge was needed to reduce the significant uncertainties regarding the effectiveness of the strategy of in-vessel retention of molten core to avoid challenging the integrity of the containment. This strategy had been used in the design of new reactors (e.g. AP1000, APR1400, CAP1400) and was already considered by some Member States to be part of the justification for lifetime extension of operating reactors.

The meeting in December 2015 led to the Technical Meeting on Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling, held from 17 to 21 October 2016, in Shanghai, China. That meeting, which is the subject of the present publication, provided a platform for detailed presentations and technical discussions on recent progress in R&D activities on in-vessel melt retention and ex-vessel corium cooling during severe accidents at water cooled reactors. It helped to facilitate the exchange of relevant R&D results; foster worldwide collaboration on R&D activities; enhance communication among industry (e.g. utilities, vendors), regulatory bodies and research organizations; and support the discussion and updating of scientific and engineering knowledge in this area.

The present publication constitutes a detailed report of the meeting and includes summaries of the presentations given and the follow-up discussions; the presentations given at the meeting are provided in the supplementary files available on-line. The conclusions and recommendations for further work presented are those of the meeting participants.

The IAEA officers responsible for the preparation of this publication were A. Amri of the Division of Nuclear Installation Safety and K. Yamada of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

As a result of the Fukushima Daiichi accident in March 2011, the Director General of the IAEA convened a Ministerial Conference on Nuclear Safety in Vienna, Austria, in June 2011. The Conference adopted a Ministerial Declaration that requested the Director General to prepare a draft Action Plan. The Action Plan sets out a comprehensive programme of work, in 12 major areas, to strengthen nuclear safety worldwide. Under one of these areas, entitled: Action 12: Effectively utilize research and development, the IAEA Secretariat was requested to provide relevant stakeholders with assistance as appropriate to conduct necessary research and development in nuclear safety, technology and engineering, including that related to existing and new design-specific aspects.

After the Fukushima Daiichi accident, new R&D activities have been undertaken by many countries and international organizations relating to severe accidents at NPPs. As part of the Action Item 12, the IAEA held, in cooperation with the OECD Nuclear Energy Agency (OECD/NEA), the IEM on Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant at its Headquarters, from 16 to 20 February 2015. The objective of the IEM was to facilitate the exchange of information on these R&D activities and to further strengthen international collaboration among Member States and international organizations.

As one of the follow-up meetings for the IEM, the IAEA held a TM on Post-Fukushima Research and Development Strategies and Priorities at its Headquarters in Vienna, Austria, from 15 to 18 December 2015. The objective of that TM was to provide a platform for experts from Member States and international organizations to exchange perspectives and information on strategies and priorities for R&D regarding the Fukushima Daiichi accident and severe accidents in general. The experts discussed R&D areas that need further attention and the benefits of possible international cooperation. It has been highlighted during the IEM and confirmed at the TM that the R&D area regarding in-vessel melt retention (IVMR) and ex-vessel corium cooling/stabilization (EVCC) is one of the highest priority areas. The participants emphasized that more phenomenological knowledge needs to be gained to reduce the significant uncertainties which remain as to the effectiveness of the strategy of in-vessel retention of molten core to avoid challenging the integrity of the containment. This strategy is used in the design of new reactors (e.g. AP1000, APR1400 and CAP1400) and is already considered by some Member States for the justification of lifetime extension of operating reactors. This led to a Technical Meeting on Phenomenology and Technologies Relevant to IVMR and EVCC that was conducted on 17-21 October 2016, in Shanghai, China.

The purpose of the meeting was to provide a platform for detailed presentations and technical discussions on recent progress in R&D activities on IVMR and EVCC during severe accidents at water cooled reactors (WCRs). The meeting aimed at facilitating the exchange of relevant R&D results, fostering worldwide collaboration in R&D activities, enhancing communication between industry (utilities, vendors, etc.), regulatory bodies and research organizations, and discussing and updating scientific and engineering knowledge in this area.

1.2. OBJECTIVE

This publication is based on the outputs and outcomes of the TM on Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling.

The purpose of this publication is to capture the current state of knowledge related to phenomenology and technologies as well as the challenges and pending issues relevant to IVMR and EVCC for WCRs by summarizing the information from the TM, as provided by the meeting participants, in enough detail that Member States can refer to it. The publication includes, online as supplementary files, all the presentations provided during the meeting. Any recommendations and conclusions presented in this publication are those of the meeting participants.

1.3. SCOPE

This publication covers IVMR and EVCC for WRCs. It summarizes the technical sessions and discussion sessions that took place during the TM on Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling. The TM had the following objectives with an emphasis on phenomenology and technologies relevant to IVMR and EVCC for WCRs:

- To discuss recent progress in the strategies, phenomenology and technologies of IVMR and EVCC;
- To exchange information on technologies for cooling the core melt and corium during severe accidents at WCRs and on their application to existing and future NPPs;
- To collect information on fundamental phenomenological research that can help to understand corium melt progression; and
- To discuss possible international collaboration in R&D activities on IVMR and EVCC.

The meeting included also discussion sessions to enable participants to contribute to the summary and highlights of the meeting and to make recommendations to the IAEA on future activities in this area.

1.4. STRUCTURE

This publication contains three sections and two annexes, as well online supplementary files.

Section 1 recalls the background, the objectives and the organisation of the TM, as well as the structure of the TECDOC.

Section 2 provides a detailed summary of the technical and discussion sessions. Each session summary is based on the summaries that were prepared by the Scientific Secretaries, presented to, and agreed with the meeting participants during the TM.

In Section 3, general conclusions and recommendations by the participants to the TM are presented.

Annex 1 and Annex 2 provide the meeting programme and the list of the participants to the TM, respectively.

The online supplementary files include all the presentations provided during the meeting.

2. SUMMARY OF MEETING TECHNICAL SESSIONS

The TM was co-organized by the Department of Nuclear Energy and the Department of Nuclear Safety and Security. It was hosted by the Government of China through the Shanghai Nuclear Engineering Research and Design Institute (SNERDI), Shanghai, China. In total, 52 nominated participants from 18 Member States and 11 observers from the host country participated in the meeting.

2.1. OPENING SESSION

The meeting consisted of the Opening Session, 6 Technical Sessions, 2 Discussion Sessions, 2 Summary Sessions and the Closing Session. Thirty-three (33) presentations were provided at the Technical Sessions, each of which was followed by very active technical discussion. Several issues related to the session topic were discussed at the end of each Technical Session. General issues related to the two main topics of the meeting, i.e. IVMR and EVCC, were discussed at the Discussion Sessions, respectively.

2.2. SESSION 1A: GENERAL CONSIDERATIONS ON IN-VESSEL MELT RETENTION STRATEGY

The first Technical Session was dedicated to general considerations on IVMR. Seven (7) presentations were provided on feedback from the international seminar on IVMR; modeling and analysis of IVMR and EVCC in the U.S; application of IVMR strategy in Korea; IVMR for WWER; IVMR at Loviisa NPP; R&D activities for IVMR evaluation performance at KAERI, and IVMR technical basis for CANDU reactors.

2.2.1. Summary of the presentations

(a) Summary and Conclusions from the International Seminar on In-Vessel Retention Strategy, F. Fichot, IRSN, France

The International Seminar was co-organized by JRC and IRSN, with the sponsorship of ETSON on 6-7 June 2016, and was hosted by IRSN in Aix-en-Provence (France).

The seminar covered all the important issues related to strategies of in-vessel corium retention, from the physical understanding to the safety demonstration and regulatory frames. Experts' presentations provided an overview of the current knowledge and understanding about the physical processes associated with corium and its interaction with the reactor vessel. The major points of the safety demonstration were pointed out and discussed. Some industrial aspects were also addressed and discussed by reactor designers.

One of the objectives of the workshop was to provide a basis for the orientation of new research projects such as the European project IVMR. This project started in 2015 and is coordinated by IRSN. It aims at developing knowledge and tools to estimate the efficiency of the measures dedicated to stabilizing corium in the reactor vessel in case of severe accident with core melting in reactors of 1000MW(e) or more.

The following similarities between the various reactor designs and the design options are considered as positive for IVMR success:

- The passive reflooding of the reactor cavity, up to the level of the primary loops (as far as PWRs are concerned).
- The design of vessel insulation to allow water circulation between the vessel wall and insulation and to allow steam venting from the top part.
- The simultaneous in-vessel water injection (direct or not) to reduce the risk of focusing effect[1].
- The presence of a large mass of steel in the lower plenum which would most likely avoid the formation of a heavy metal layer and would increase the thickness of the top metal layer, thus reducing the risk of focusing effect.
- The absence of penetrations (as in several PWR and PHWR designs).
- The large water inventory (including safety injections and storage tanks) leading to delay the time of corium arrival in the lower plenum and therefore to reduce the residual power to extract (up to one day before corium arrival for some designs).

Though significant progresses were made, there is still lack of knowledge and need to continue the research in different domains. Indeed, currently, there are still missing data to be able to completely understand the phenomenology of the processes involved in IVMR. It includes thermochemical effects, high Rayleigh number turbulent convection, surface effects on the critical heat flux (CHF) and mechanical behavior of the thin (cold shell) resulting from ablation of the vessel wall. Therefore, the standard analysis for safety demonstration of IVMR suffers from two opposite drawbacks: it neglects transient situations which could lead to higher heat fluxes than those in steady-state situations and it uses excessively simplified models (such as focusing effect) which might be too conservative.

The current approach followed by most experts for IVMR is a compromise between a deterministic approach using the significant knowledge gained during the last two decades and a probabilistic approach (see, for example, the presentation summarized in section 2.5.1 (a)) to consider large uncertainties due to lack of data for some physical phenomena (as listed above) and due to excessive simplifications of models.

It was concluded that a harmonization of the positions of safety authorities on the IVMR strategy is necessary to allow decision making based on scientific knowledge and not on personal views. For that end, a consensus on several issues has to be reached among R&D experts. This includes the issue of the transient evolution of oxide and metal layers in the lower plenum and the issue of the long term mechanical behavior of the ablated vessel wall following for instance, a re-pressurization of the RV.

Additional R&D is encouraged, as well as international collaborations on the topic. In Europe, the IVMR project aims at providing new experimental data and a harmonized methodology for IVMR. It will also include an activity on innovations dedicated to increase the efficiency of the IVMR strategy. More generally, this workshop has shown that industrial actors make constant efforts to take into account existing knowledge and uncertainties to propose more effective implementation of the IVMR strategy.

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[1] In most configurations of molten corium in the reactor vessel lower head, there is a metal layer on top of a large oxidic pool (see summary 2.4.1 (a) below). The focusing effect mainly involves this metal layer; if the radiative heat transfer on the top of the metal layer is not enough to discharge the thermal power received from the oxidic pool, the temperature of the metal layer is expected to increase and the metal layer to transfer its heat to the reactor vessel wall. This focusing effect is expected to increase as the metal layer thickness decreases.

There is a clear distinction between existing reactors for which the implementation of IVMR strategy is a serious challenge and new reactor designs for which several options can be implemented to increase the safety margins, such as delaying corium arrival in the lower plenum, increasing the mass of molten steel or implementing measures for simultaneous in-vessel water injection.

It was finally proposed to organize a similar workshop in 3 or 4 years to share new data and estimate the progress made towards a harmonized approach of IVMR.

(b) Modelling and Analysis of In-Vessel Melt Retention and Ex-Vessel Corium Cooling in the U.S, E. Fuller, USNRC, USA

This presentation provided an overview of modelling and analysis of IVMR and EVCC in the United States. He noted that successful strategies for in-vessel retention or termination of ex-vessel core melt progression depend on what equipment can be utilized and which operator actions can be taken to add sufficient coolant water and remove the decay heat. Such actions, along with appropriate use of plant design features, can prevent containment failure or mitigate radioactive releases.

The importance of in-vessel retention and termination of ex-vessel core-melt progression has been recognized for decades in the U. S. The path from the Reactor Safety Study (WASH-1400) to the establishment of the Technical Basis for the Containment Protection and Release Reduction (CPRR) Rulemaking decision went through several steps including the Industry Degraded Core Rulemaking Program (IDCOR), the development of the initial, and then revised Technical Basis for Severe Accident Management Guidelines (SAMG); mechanistic analyses of core-melt progression were performed during these steps and continue to be carried out.

Analyses show that water addition prior to significant relocation of molten core material into the lower plenum can improve the likelihood for IVMR. As the core material (in the form of melt jets) relocates to the lower plenum, water there (either pre-existing or as a result of injection from external sources) helps break up the melt into fragments and quench it. Without water addition, vessel failure is highly likely either due to lower head creep rupture or penetration failures. If core-melt progression cannot be arrested in-vessel, then core debris would be released to the containment, where it would interact with concrete and other structural materials, resulting in concrete erosion. Water addition ex-vessel in this scenario can stabilize the core debris and prevent basemat or containment liner or failure. It is important to know, however, when to add water and how much to add as part of an effective accident management strategy.

Phenomenological research in the 1980s and 1990s investigated in-vessel and ex-vessel melt progression experimentally. The resulting data were used to develop models to simulate accident progression. Over time, the models have been improved steadily from their original versions as more and more experimental information became available.

The most-current code versions of severe accident codes (MELCOR 2.1 and MAAP 5.03) are used by USNRC and EPRI respectively to simulate both in-vessel and ex-vessel water addition to cool molten corium. Indeed, during the CPRR analyses for the Mark I BWR, simulations of accident sequences where water was added and the containment was vented, predicted core debris cooling that prevented containment failure either by liner melt-through or by over-pressurization. Furthermore, for some scenarios, simulation results predicted that vessel integrity could be maintained. If water is added at or shortly after RV failure, present simulations for Mark I and Mark III BWR analyses showed that it can be possible to minimize molten core concrete interaction. Results from these analyses have provided valuable insights for not only providing a water source and means to deliver it to the vessel required, but the pressure must be low enough for the water to enter. In existing light water reactors (LWRs), operator actions can be carried out to reduce the pressure. Failing this, it is possible that there would be a failure of some kind (such as a stuck-open valve, instrument tube melting, pump seal loss of coolant accidents (LOCAs), or a hot leg creep rupture) to cause depressurization. If the RV is depressurized sufficiently before core damage, then timely water injection into the RV can arrest core damage, reduce fission product release and prevent vessel failure.

Model validation continues and the integrated severe accident codes continue to be improved. Research needs are identified regarding modelling of the late phase of melt progression, hydrogen risk, ex-vessel melt behaviour, fission product chemistry and transport, and pool scrubbing under saturated conditions. Modelling of mitigation systems as well as reduction of residual uncertainties are other needs to be addressed. The challenges posed by workforce ageing and budget reality could be addressed with knowledge preservation, R&D optimization and increased international cooperation.

(c) Application of IVMR-ERVC Strategy in Korea, K. H. Lim, KINS, Republic of Korea

The speaker's presentation was on the experimental and analytical R&D activities carried out in support to the implementation of IVMR-ERVC (External Reactor Vessel Cooling) strategy in the design of APR1400 NPP. Experiments and analyses to justify the feasibility of the strategy were the main topics in the mid-2000s when the design of the NPP was under development. SBLB (Subscale Boundary Layer Boiling) experiment was performed to examine the CHF at the external vessel wall of the APR1400 reactor lower head from 2003 to 2004. The effect of flow conditions such as inlet sub-cooling, ex-vessel wall surface coating, and impact of design modification of the insulator on the CHF were examined using the reduced reactor scale (1:2) test facility. The thermal margin at the various accident scenarios was analysed using the severe accident system codes (SCDAP/RELAP5-3D and SCDAP/RELAP5-MOD3.3), Lumped-parameter code (VESTA), and a Computational Fluid Dynamics (CFD) code (LILAC) based on CHF values provided by SBLB.

Several experimental approaches to examine the ex-vessel coolant boiling and recirculating behaviour were considered. K-HERMES-HALF (Hydraulic Evaluation of Reactor Cooling Mechanism by External Self-induced flow-HALF scale) experiment was performed by using the half-size test facility of the APR1400. This experiment used the non-heating method with air bubble injection which did not consider any boiling phenomena. Another facility, CASA SC, using a 1/10 scaled-down three-dimensional hemispheric lower head vessel and insulator provided CHF results based on scaling analysis. However, those results were limited at the scaled down geometry and at metallic pool region. KAIST-CHF experiments were performed under forced convection conditions to examine the effect of mass flux, inlet sub-cooling, exit quality, heater material, and additive on CHF. The maximum experimental CHF correlation was derived at the 1st campaign as a function of mass flux, sub-cooling at the inlet, and exit quality by using the test specimen with actual size and increasing heat flux conditions according to the inclination. At the 2nd campaign, the maximum CHF correlation was developed as a function of mass flux, void fraction, and exit quality with reduced radius of the test specimens 0.15~0.5m. Recently, more generic CHF model including implicit relations among variables such as slug length, vapor velocity, liquid velocity, and liquid micro thickness has been developed based on the previous experiments.

Based on the previous work, MAAP5 analysis was performed by KEPCO E&C for Shinkori 3 and 4 NPPs as APR1400 type. As a result, the maximum heat flux at the ex-vessel wall is less than CHF by SBLB experiment at the limiting accident scenario, LBLOCA (Large Break Loss of Coolant Accident). However, the integrity of the ICI (In-Core Instrumentation) penetrations is not guaranteed. The analyses also showed that if water is injected to the reactor in an appropriate time, IVMR strategy is successful by top cooling effect without failure of ICI penetrations. Currently, KAERI is performing VESTA-S experiment to examine the failure mechanism of APR1400 ICI penetration using ICI penetration specimen; the results are used to develop a tube ejection failure evaluation code based on MAAP5. KINS is developing the IVMR-ERVC assessment methodology which consists of transient analysis using MELCOR 2.1 according to various accident scenarios, steady state analysis using lumped parameter method under bounding initial and boundary conditions, and uncertainty analysis. Comparison of analysis capability of MAAP5 and MELCOR 2.1 is expected to be performed in terms of IVMR-ERVC strategy assessment.

(d) Corium Retention Strategy on VVER under Severe Accident Conditions, Y. Zvonarev, Kurchatov Institute, Russian Federation

This presentation reported on analyses of in-vessel and ex-vessel corium retention strategies for WWER under severe accident conditions. Results from RASPLAV and RASPLAV-2 projects, as well as from MASCA and MASCA-2 projects were used to develop HEFEST-ULR computer code to simulate thermal and chemical processes in RV lower head and in the core catcher in case of severe accident with core melting.

IVMR strategy was analyzed using SOCRAT/ HEFEST and ASTEC computer codes for WWER-600, WWER-1000, and the two designs of WWER-1200, in case of LBLOCA with SBO (Station Black Out). Given the scenario, water supply to the RV was provided by the safety injection tanks only; IVMR consisted in flooding the reactor cavity and removing the decay heat from the corium through the RV wall to external water. Sensitivity analyses were performed regarding for example, initial melt temperature, mass of the melt, composition of the melt, and decay heat decrease due to fission product releases. It was shown that (i) It is possible to efficiently use the IVMR strategy for WWER-600 through external RPV cooling; (ii) the IVMR strategy for WWER-1000 without additional measures on intensification of external cooling was not sufficient to prevent RV failure in most cases; (iii) RV failure cannot be avoided for WWER-1200 and it is necessary to use core catcher for severe accident management.

The core catcher (CC) is designed to reduce to a safe level of radiological consequences of severe accidents. The main idea of this concept is the core melt localization in a special device (CC), where the melt interacts with the sacrificial material whose composition was determined based on experimental and theoretical investigation. As a result, volumetric decay heat decreases and there is an inversion of corium pool: oxide fraction, saturated by sacrificial materials, moves upwards, while metal part moves downwards. In the whole process, the system remains sub-critical owing to the addition of gadolinium oxide to the sacrificial material. Corium is cooled by water boiling on the external CC casing and on the corium pool upper surface. This concept justification is based on SACR experiments on interaction of corium with a sacrificial material. Calculations were performed using HEFEST-ULR code.

(e) In-Vessel Melt Retention at Loviisa NPP: Past and present Activities, T. Laato, Fortum, Finland

The speaker provided an overview on the past and ongoing R&D regarding IVMR strategy for Loviisa NPP. In Loviisa NPP (2 WWER-440 reactors) the IVMR strategy has been selected to arrest the downward progression of molten corium in case of a severe accident. The IVMR approach was first proposed for Loviisa NPP in 1988 by T.G. Theophanous. In early 1990's an extensive research programme was conducted to ensure the applicability of IVMR to Loviisa and in 1995 the concept was approved by the regulatory body. The necessary plant modifications were made during annual outages of reactor 1 and reactor 2 in 2000 and 2002 respectively.

To justify IVMR at Loviisa, ULPU and COPO experiments were conducted. In ULPU experiments the heat transfer from the vessel wall to the coolant was examined to determine critical heat flux at which the vessel wall would no longer be coolable. In COPO experiments heat transfer within corium pool was examined to determine the heat flux distribution from the molten pool to the vessel wall.

The ULPU facility was 1:1.24 height scale heated pipe with diameter and flow restrictions at the top comparable to slice of Loviisa reactor cavity. In the early tests, only the vertical part of the vessel wall was considered. Due to limitations of the heater, CHF was not achieved; it was concluded, that the CHF for the vertical wall is at least 1.2 MW/m^2. During the approval process by the authority the ULPU experiments were continued to determine the CHF values along a hemispherical vessel wall in full height facility.

The COPO facility was a 2D, 1:2 scale slice in the shape of the WWER-440 lower head. Salt water with Joule heating was used as a simulant for corium and the boundary conditions were kept constant by varying coolant flow in the cooling modules at the boundaries. The average heat fluxes were found to be consistent with widely used convective heat transfer correlations by Mayinger for the upward heat flux and Steinberner & Reineke for the sideward heat flux.

After the IVMR approach was already approved by the regulatory body, confirmatory COPO II experiments were conducted. In COPO II the simulant material was salt water with Joule heating similarly to previous experiments, but the cooling was applied using liquid nitrogen. This approach made it possible to achieve strictly isothermal boundary conditions due to formation of ice. In COPO II also hemispherical lower head was used (1:2 scale of AP600 lower head). Some experiments were also carried out with 2-layer configuration using pure water separated with aluminum sheet as a top layer. It was concluded that the heat transfer correlation by Churchill & Chu was applicable especially for a thick unheated top layer and the heat flux distribution from the metal layer to the wall is rather flat.

After RASPLAV and MASCA projects, the question about more complicated pool configurations has been raised. Therefore, it was considered, that further research needs to be done to examine the effects of 3-layer configuration to the applicability of IVMR at Loviisa. Calculations were performed with in-house program IVMRSYS. It was found, that in most cases the IVMR is still applicable with large margin to the failure criteria. In case of LBLOCA scenario simulated with MELCOR, the comparatively low zirconium oxidation degree combined with reduced melting steel mass lead to a situation, in which the CHF will be exceeded. Nonetheless the low zirconium oxidation degree could be considered unlikely in WWER-440/213 reactors and the applicability of widely used heat transfer correlations for a thin top metal layer is questionable.

Regarding IVMR at Loviisa NPP, currently transient phenomena including corium relocation into the lower plenum and proper modelling of thin top metal layer are considered by Fortum as the most important topics of research. R&D is pursued with Royal Institute of Technology in Stockholm (KTH) under the EC IVMR project. At KTH, a new facility called SIMECO-II will be built. It will be a 2D-slice of a hemispherical lower plenum with transparent side walls to study heat transfer in 2- and 3-stratified molten pool at high temperature using high temperature melts of salt and metal as simulants of oxide and metal parts of corium respectively. Moreover, the transparent sidewalls will enable the possibility to measure flow velocities within the transparent salt layer. Preliminary calculations of one- and two-layer cases have been performed as well as material investigations of the simulant materials' properties.

(f) R&D Activities for the Evaluation of In-Vessel Corium Cooling Performance at KAERI, H. Y. Kim, KAERI, Republic of Korea

This presentation reported on the five-year R&D programme funded by the Korean government and launched in 2012 at KAERI. The purpose of the programme is to develop and verify an Evaluation Technique for In-Vessel Corium Cooling Performance. This technology is considered a key for the severe accident mitigation. It is expected that a practical severe accident mitigation plan is established and can be used for the evaluation and verification of severe accident management strategy of the operating NPPs.

The R&D programme includes the topics for RPV safety in case of an accident with core melting, i.e. evaluation of in-vessel corium behavior, evaluation of in-vessel corium coolability[2] and evaluation of ICI penetration nozzle failure.

The evaluation of in-vessel corium behavior consisted of (i) the development of a phenomena identification and ranking table (PIRT) for in-vessel coolability covering core heat-up and loss of geometry, corium relocation of upper plenum and core relocation to lower head; (ii) development of SIMPLE (Severe In-vessel Melt Progression in Lower Plenum Environment) computer code; and (iii) SIMPLE verification and validation.

In-vessel Corium Coolability included the development of in-vessel coolability map based on MELCOR calculations, SAMG modifications considering PSA Level 1 main sequences leading to core melting, and enhancement of ex-vessel cooling with outer RPV surface metal coating. The latter involved testing with microporous coating to investigate impact on heat transfer. Water sub-cooling and orientation effects on heat transfer have been investigated as well.

To investigate ICI penetration nozzle failure, welding material ablation tests and ICI penetration failure tests have been performed in VESTA-S test facility with ZrO_2; other tests are planned with ZrO_2 and UO_2. A computer code, PENTAP+, was developed to support penetration failure analysis (weld failure and tube ejection), data provided by SIMPLE code (tube temperature profile) being used at each time step. This activity will be pursued for a few more years while KAERI R&D programme will mainly focus on ex-vessel debris coolability.

[2] The term coolability is used to indicate the ability of core melt/debris to be cooled and kept for a long time below a certain temperature. For in-vessel corium coolability, this temperature is such that the reactor vessel integrity is preserved. For ex-vessel corium coolability, this temperature is around 1000 °C. Under this temperature, fission product release and non-condensable gas generation stop, and containment integrity is not seriously challenged anymore if containment cooling has been established (see Ref. [1]).

(g) In-Vessel Retention (IVMR) Design Features, Phenomenology and Technical Basis in the CANDU Reactor, R. McLean, Bruce Power, Canada

The speaker provided an overview of IVMR design features, phenomenology and technical basis for CANDU reactors. IVMR is the state of containing degraded reactor core constituents (corium) inside the calandria vessel (CV) for CANDU reactors by deploying internal CV cooling (with D_2O), external CV cooling (with H_2O) or a combination of both CV cooling modes, prior to CV failure. IVMR of degraded core materials in the CANDU CV is considered as crucial severe accident management (SAM) guidance strategy and represents a success of mitigating actions to recover control in the event of a severe accident.

CANDU reactors may benefit from adopting the IVMR strategy owing to their geometry and large heat sink availability (particularly external cooling of CV by the reactor vault or shield tank water), and materials composition. Considerable research and development effort has been expended by the Canadian nuclear industry in recent years to improve the understanding and technical basis for IVMR as a terminal corium state in response to a severe accident challenge. For a successful IVMR demonstration in a CANDU reactor, the following is required to be demonstrated:

- The molten corium will not stratify into layers of low density and high thermal conductivity producing metallic layers that would float to the top of the melt;
- The decay heat from molten corium formed at the bottom of the CV can be removed by the reactor vault or shield tank water and the tube sheet cooling water without exceeding the CHF in either location; and
- The solidified molten corium in the crust layer will have very low rates of chemical corrosion with the CV wall so that it does not weaken the vessel wall thickness as it is retained inside the vessel for a prolonged period.

The CANDU IVMR strategy also involves use of emergency mitigating equipment (EME), which consists of portable pumps and portable AC power generator, and related EME and SAM guides to facilitate retention of core debris in various physical configurations and forms within the CV.

The CANDU IVMR phenomena and challenges are separated into two broad categories:

1. Dynamic phenomena/challenges: These are short-lived (transient) processes, events or loads that occur when hot core materials are brought into contact with water or structures. Typically, dynamic IVMR phenomena occur during core degradation and disassembly (e.g. hot materials entering water or in contact with a wall) or when SAM provisions have just been applied (e.g. vaporization of water coming into contact with hot materials). In-vessel mechanical loads and in-vessel heat transfer characteristics are of importance. The first challenges are addressed with structural/stress analysis to demonstrate the integrity of the vessel. The second ones whose concerns include e.g. local heat flux, crust formation, require thermal-stress analyses for relevant accident scenarios.
2. Sustained phenomena/challenges: These are long-lasting (quasi-stable) processes or loads that occur when a degraded core configuration and CV conditions have stabilized. Sustained phenomena pertain to in-vessel heat transfer characteristics, chemical interactions, ex-vessel heat transfer characteristics, heat transfer at the vessel ends, and horizontal and vertical penetrations.

Material Interaction tests for CANDU reactors (MATICAN) were completed using CANDU corium to assess the potential for corium stratification and heat focusing at the CV wall, as well as to investigate rate and depth of corrosion between corium and VC steel. In this frame, tests which have been performed at RASPLAV-3 facility in Russia under varying conditions of oxidation, Zr/U ratio and temperature showed that (i) corium melt will not result in stratification within the range of zirconium oxidation expected in CANDU reactors (i.e. in the range of 0 to 35%) meaning vessel failure due to corium melt formation is unlikely to occur; and (ii) there is no loss of wall thickness by chemical corium-wall interaction at a wall temperature up to 900 ºC and a reduction in wall thickness of 4.5 mm at 1200 ºC is observed after 21 hours.

Ongoing ex-vessel heat transfer tests are being performed to investigate ex-vessel CHF in pool boiling and CHF at vessel ends (e.g. steel balls filled end shield). The results will be used to confirm the validity of MAAP CANDU severe accident analysis for IVMR.

2.2.2. Summary of the session

It is commonly recognized that the IVMR strategy achieved by ERVC and/or in-vessel flooding is one of the most effective measures to prevent the progression of severe accidents at water-cooled reactors. Several operating nuclear power reactors and new ones use or aim to use IVMR strategy, and some of them have dedicated systems.

A lot of R&D has been performed and are still on-going to develop IVMR strategy and technologies at national, regional and international level, and the Fukushima Daiichi accident revitalized it. The assumptions for the justification of IVMR application must be re-evaluated based on new findings.

Most of the efforts have been made in understanding of key phenomena with experiments and numerical analyses, code improvement/validation, and application of IVMR strategy to specific reactors and its optimization. The insights from the International Workshop revealed several areas where more research efforts are needed. Most of them were discussed at this Technical Meeting more in detail.

There are still uncertainties, and they mainly come from insufficient knowledge in phenomenology related to accident progression and limitations of experimental facilities and instrumentation. Probabilistic approaches or assumptions are considered necessary as complement for deterministic approach.

It is agreed among the TM participants that the probability of successful retention of melted core in reactor vessels (RVs) is generally higher in lower power reactors, particularly due to the sufficient margins between the heat flux generated by the corium in the lower head and the CHF. It also highly depends on the specific designs as discussed later.

2.3. SESSION 1B: EXTERNAL REACTOR VESSEL COOLING

The second Technical Session addressed ERVC aspects through two (2) presentations that provided an introduction of CAP1400 in-vessel retention experiments, and an overview of IVMR strategy respectively.

2.3.1. Summary of the presentations

(a) Introduction of CAP1400 IVMR Experiments, K. Zhang, SNERDI, China

This presentation provided a detailed introduction of the R&D work performed for the CAP1400, focusing on IVMR as one of the key safety features in the design of CAP1400. The thermal success criterion of IVMR strategy is that the heat flux through the RPV lower head does not exceed the local CHF. This success depends on many factors such as decay heat of in-vessel molten debris, distribution of heat flux along RPV lower head, geometry of lower head, outer surface characteristics of RPV lower head, characteristics of natural circulation, geometry of flow path between RPV and insulation, water temperature at inlet structure, system pressure and composition of cooling fluid. There were several experimental facilities with different sizes (e.g. ULPU-I to ULPU-V, BETA, MIT, SBLB, CYBL, SULTAN) that investigated the influence of those factors on CHF. However, it was deemed necessary by SNERDI to further investigate them for the design of CAP1400.

The demonstration of the thermal success criterion of IVMR strategy for CAP1400 led to carry out three series of experiments as follows: (i) the first series investigated CHF distribution along outer surface of CAP1400 RPV lower head; (ii) the second series addressed the key factors of enhancing CHF in another test facility with forced water circulation; (iii) the third test series allowed to check the applicability of Globe-Dropkin equation at higher Rayleigh (Ra) numbers as it would be the case for the metal layer of melt core debris located in the RPV lower head of CAP1400.

The presentation focused on the design and use of IVMR-ERVC facility that was built to investigate CHF distribution along outer surface of CAP1400 RPV lower head using design principles such as full-height facility as prototype of CAP1400 with slice geometry, scaling of main flow area and modelling important structures, ensuring natural circulation flow, using prototype material for heating section, and same cooling fluid as prototype. The experimental conditions (e.g. thermal conditions, cooling fluid composition, heating surface characteristics and geometry of flow path) could be adjusted according to different accident scenarios. An appropriate measurement system was developed to monitor the following parameters: electric power in, and temperature distribution of the heater block, cooling fluid temperature, pressure, flow rate of natural circulation and water level in the reactor cavity. Flow pattern observation is performed with a high-speed camera. The CHF is measured by an indirect method as the ratio of the heating power and the effective heating area of group of rods where CHF occurs, corrected by a factor determined based on thermal equilibrium experiment and (CFD) analysis.

Regarding the impact of heating surface characteristics, it is possible to prepare samples by oxidizing them in air for a long time. Although changes in surface roughness after oxidation are negligible, experiments showed that oxidized heater block exhibited enhancement of CHF compared to those without pre-oxidation.

(b) Overview of the IVMR Strategy, J. Zdarek, UJV, Czech Republic

The speaker presented the status of the experimental and analytical work being performed at UJV for possible application of IVMR to WWER-1000. This work is part of the EC project HORIZON 2020 IVMR that includes several tasks. The UJV Rez a.s. is leading the Task 4 of this project with clear goal to provide data on external RPV surface cooling in case of SA scenarios, based on large scale experimental facility with WWER-1000 configuration.

The results of more than 100 small scale experiments performed at UJV with clean surface and with (cold spray) coating clearly show significant increase of CHF on coated heating surface compared to clean heating one. Also, the lessons learned from already performed tests and analytical work on relevant large-scale facilities are being used at UJV in the design and construction of the large scale experimental facility with WWER-1000 configuration. For the latter, cooling channel will exactly simulate the reactor cavity of WWER-1000/320 configuration. RPV steel explosively welded on copper heater bloc will simulate lower head and cylindrical part of the RPV. Thermocouples are located about 3 mm from the cooled outer surface of RPV test sample to provide indication on cooling crisis. Moreover, the configuration will allow to add a baffle and to change a gap between test sample and this baffle to optimize coolant mass flow conditions. Access to the outer surface of RPV test sample is eased to clean this surface and to have the same initial conditions as well as to perform (cold spray) porous coating. Cleaning and coating can be performed with hand held spray nozzle. The first tests are expected to be carried out at the large-scale facility in March 2017 with the first results to be presented in November 2017.

2.3.2. Summary of the session

One of the two success criteria of the IVMR strategy is ‘thermal criterion’ to make sure the heat flux from in-vessel molten pool is less than the CHF at the outer surface of the RV lower head that is determined by the conditions of external cooling with water poured in the reactor cavity.

Main factors affecting the CHF include: (i) stability of the natural circulation; (ii) geometry of the flow path, (iii) outer surface conditions of RV lower head, (iv) water sub-cooling at the inlet of the flow path.

Full height experimental facilities are necessary for validation data, and they need to be designed as closely as possible to the real conditions.

Based on the results from small-scale experiments, the most effective measures to increase CHF might be optimisation of the flow path and the outer surface conditions of the lower head.

The two presentations were given on new large experimental facilities, which are designed, based on the lessons learned from small- and large-scale facilities, to measure the CHF at the outer surface of the RV lower head under more realistic configurations and flow conditions and to comply with the recommendations provided above.

2.4. SESSION 1C: MOLTEN POOL BEHAVIOURS AND STRUCTURAL INTEGRITY OF REACTOR VESSEL

This third Technical Discussed Molten Pool Behaviors and Structural Integrity of Reactor Vessel based on seven (7) presentations.

2.4.1. Summary of the presentations

(a) Heat Transfer in Homogeneous and Stratified Melt Pools in the Lower Head of a Reactor Pressure Vessel, A. Miassoedov, KIT, Germany

The presenter provided a presentation summarizing the main experimental findings on heat transfer from the corium melt pool in homogeneous and stratified melt pool configurations and outlined the objectives of the future experiments planned in the European IVMR project.

As it was observed in the RASPLAV and MASCA OECD projects, a two-layer or even a three-layer melt pool can be formed depending on melt composition and boundary conditions with a metallic layer atop of an oxide layer. Such a melt pool configuration plays a very important role in the reactor case because the heat transferred to the overlying metallic layer could be then transferred to the vessel wall along the metallic layer. This phenomenon, known as the focusing effect, is one of the highest risks for the IVMR strategy.

The steady state behavior of debris and melt pools in the lower head has been investigated in several experimental studies in the past and is continued nowadays. The tests were performed in different geometries (2D or 3D), using different simulant materials (water, salts and prototypical corium melts) and boundary cooling conditions and simulation of the decay heat. The main objectives of the experiments were:

- to reduce uncertainties in the understanding of thermophysical phenomena influencing the melt pool configuration, composition and masses/thicknesses of melt layers and interfacial crusts, relative positions of the layers, heat and mass transfer between the layers and heat fluxes to the melt pool boundaries,
- to determine the conditions in the melt pool which are critical for the system behaviour, such us layer inversion, mixing and focusing of the heat flux,
- to develop correlations and validate calculation models for stratified fluid layers, and
- to predict the heat transfer loadings on the vessel wall for different configurations of the melt pool.

The results obtained in the previous experimental programmes indicated that:

- COPO II and BALI, experiments in large scale geometry using water as simulant and performed in 2D slice geometry, clearly showed higher heat transfer coefficients (20-30%) for the side and bottom boundaries than those in the 3D ACOPO experiments, while the upward heat transfer coefficients were only slightly higher.
- Large discrepancy exists between the experimental data and correlations used for heat generating layers for high Ra numbers typical for reactor conditions (10^{15} - 10^{16}).
- Experimental data for externally heated layer is still very limited.
- First confirmation of focusing effect in the melt layer atop of solid debris was observed in the LIVE experiment.
- Large discrepancy exists in the modelling of the heat flux in the two-layer melt pool and experimental data due to excessively simplified assumptions in models.
- No correlations exist for a heat transfer in a three-layer pool.

To complement the experimental data on behavior of two- and three-layer melt pools in the reactor vessel lower head, new experimental data are necessary to address the following issues:

- Higher Ra numbers up to 10^{16};
- Interfacial crust between top and bottom layer;
- Top cooling vs. adiabatic conditions;
- Layer mixing and inversion of stratification;
- High thermal conductivity of the top layer.

New experimental programmes are under preparation in the IVMR project of the EU H2020 programme; their objective is to improve the understanding of the heat transfer in homogeneous and stratified melt pools in the lower head of a reactor pressure vessel. They will include:

- Study of the heat transfer between a heat generating layer and an immiscible layer on top or at the bottom (including a three-layer configuration) under different boundary conditions;
- Effects of relative melt layer positions and of the top layer thickness on the heat partitioning and focusing;
- Study of inversion of layers caused by their density evolution and evaluation of corresponding transient heat fluxes, e.g. during formation of a new top layer;
- Influence of turbulence on the mixing and heat transfer between two layers having different densities;
- Influence of interfacial crust and partially solidified crust on the heat transfer between the heat generating layer and the layer atop of it;
- Analysis of direct cooling of the top metal layer after re-flood and resulting transient evolution of heat fluxes. The influence of oxide crust at the surface of the top metal layer on the resulting heat fluxes will be also considered;
- Measurements of heat flux between the vessel wall and the stratified melt pool providing the ratio of heat flux between the oxide pool and the metal layer. Various corium compositions will be studied to vary the thickness of the metal layer.

It is expected that an improved understanding of these processes will help to identify the most challenging situations for the vessel integrity and to define SAM measures and to design safety features that would minimize the risks in operating and in new reactors respectively.

(b) Research on Thermodynamic Interaction of Corium Material in Lower Head, P. Gu, SNERDI, China

R&D conducted at SNERDI on thermodynamic interaction of corium materials in RV lower head was presented by P. Gu. In case of IVMR application, the heat transfer imposed by in-vessel corium is a vital part for IVMR strategy success. For a given decay power, corium pool configuration determines the heat flux profile along the vessel wall, which might produce uncertainties associated with IVMR strategy.

SNERDI studied the corium pool configurations by analysing possible interactions between relocated corium and core internals using material thermodynamic tool to analyse the equilibrium state by minimizing Gibbs energy. Several assumptions have been made regarding the possibility of interaction between core lower support plate (LSP) and relocated corium through the crust, cladding oxidation fraction that impacts the content of unoxidized Zr for thermodynamic interaction, with all UO_2 and ZrO_2 assumed to relocate in the lower head where the temperature and pressure are assumed to be 3000 K and 2 atm. respectively.

As general conclusions, molten corium is found separated into two liquid phases under prototypical accident conditions. Oxidic and metallic phases are distinguished based on their oxygen content. Unoxidized zirconium has a strong capability to extract uranium from oxidic phase to metallic phase, while cladding oxidation fraction has impact on the amount of unoxidized zirconium for interaction.

In accident sequences where material infiltration through the crust is not considered, three-layer pool (respectively two-layer pool) is found if cladding oxidation fraction is low (respectively high). The absence of unoxidized Zr for interaction leads to two-layer pool. It was noted that CAP1400 has increased LSP mass. Therefore, it is expected that the thickened top metal layer could help preventing strong focusing effect if material infiltration through the crust is not considered.

In accident sequences where material infiltration through the crust is assumed, the safety margin provided by increased LSP thickness might be questioned; a reaction rate through the crust needs to be considered leading to the possibility of three-layer corium pool, with strong focusing effect, in transient process if cladding oxidation fraction is not very high. In these cases, in-vessel injection is needed to remove decay heat and decrease the focusing effect.

(c) Corium Propagation Modelling with the PROCOR Software – Application to Corium Vessel Lower Head, L. Saas, CEA, France

The speaker's presentation was on corium propagation modelling with the PROCOR (PROpagation of CORium) software and its application to corium vessel lower head. To simulate corium flow and propagation during a severe accident , CEA has been developing since 2013 the PROCOR software for statistical analysis of transients. This software is a platform for the development of simplified physical models (mostly, integral models) and for their assembly to build applications dedicated to each type of reactor and for different severe accident management strategies (for example IVMR and EVCC). Statistical analysis of the uncertain parameters of each application can be performed using a Monte Carlo method (URANIE platform).

To evaluate the IVMR strategy success on a PWR , an in-vessel PROCOR application has been developed. This application computes mass and thermal transients of a stratified pool, with several layers, in the lower head and the vessel ablation due to the heat fluxes transmitted by the corium pool. Transient mass transfers towards or through the pool layers, corresponding to corium draining from the core, ablation of the vessel and thermochemical phenomena are considered. The external cooling is computed using a critical heat flux correlation.

Some current simplified models that compose PROCOR application for IVMR and EVCC are presented. The PROCOR ex-vessel application is based on a chaining with the TOLBIAC-ICB code for the concrete ablation and the EVCC cooling. However, the presentation focuses on corium behavior in RPV lower head. The presented modelling corresponds to the current state of the art and is based on:

- Corium flowrates from the core as input data for the lower head;
- A 0-D energy and mass balance for each corium layer including (i) transient mass and energy equation formulation; (ii) stationary heat transfer correlations; (iii) the crust considered as boundary thermal conditions; (iv) transient added mass considered (corium flows from core and ablated steel);

- A corium pool stratification model based on species' diffusivities where stratification evolves during the transient and upper crust (between the light metallic layer and oxidic layer) is stable but totally permeable to mass transfer;
- The light metal layer is responsible for the focusing effect with 0D transient mass and energy equations formulation using stationary heat transfer correlations, the emissivity of the top metallic layer being a parameter;
- 1D ablation model for the vessel function of water level in the reactor cavity and CHF;
- Ablation model for internal structures; and
- A heating/ cooling and melting model for each debris bed.

Some results have been obtained using this simplified modelling in which conservative assumptions have been made regarding largely uncertain phenomena. In these results, two modes of vessel rupture by focusing effect have been identified that are associated with a thin metallic layer on the top of the stratified corium pool. The first one is an early mode that corresponds to the formation of the thin metallic layer with a few centimeters thickness. The second one is due to thermochemical effects and could occur lately with a few decimeters thickness if the first mode is avoided. They emphasize the need of a more accurate and finer modelling for the corium pool stratification and thermal evaluation. CFD computer code has to be able to provide a support to build such an improved simplified modelling.

The thin metallic layer thermal-hydraulic modelling is expected to gain improvement from participation to the Benchmark on Thin Metallic Layer with Trio CFD computer code for which a straightforward answer could be obtained from the literature before CFD simulation, regarding the top free surface of the thin metallic layer. As for corium pool stratification modelling, the completion of the CFD multiphase modelling and multi-component modelling is needed and requires a multiphase diffusion modelling in a liquid miscibility gap. It is understood that this fine CFD simulation is complementary to dedicated experiments in the overall process of improving the integral models of in-vessel corium behavior.

(d) Experimental and Numerical Study on the Heat Transfer Characteristics, Y. Zhang, X. Jiaotong University, China

The presentation was focused on a numerical study on the heat transfer characteristics of COPRA-L1 melt pool based on the application of the Large Eddy Simulation (LES). The COPRA facility is designed to simulate the lower plenum of reactor vessel at 1:1 scale for the ACP1000, which is a two-dimensional 1/4 circular slice structure with an inner radius of 2.2 meters. Water, and molten salt ($NaNO_3$ and KNO_3) were used as simulants in COPRA-L1 tests owing to their thermal properties and non-aggressive behaviour to the vessel and easiness of technical handling.

The tests with water and those with molten salt allowed to establish distributions of the dimensionless temperature (T_{local}/T_{mean}) and the heat flux expressed through (Nu_{local}/Nu_{mean}) as functions of the relative height (H/H_{max}) and the relative angle (Θ/Θ_{max}). Expressions of Nu as function of Ra were also established in the case of water and molten salts.

From the experimental results, it can be observed that compared to the water tests, the crust formation in the molten salt tests almost suppressed the thermal stratification. In the case of water tests, the heat flux increased quasi linearly to reach its maximum of 2 around $\Theta/\Theta_{max} =$ 0.9. In the molten salt tests, smaller heat fluxes were observed in the middle part and larger at the top leading to larger q_{max}/q_{min} in the range of 2.7. Comparison with previous experiments showed that COPRA experiments in general compare well with those tests, as far as molten tests are concerned.

Due to the full-scale geometry, the Ra numbers within the corium pools could be as high as 10^{16}, matching those in the prototypical situation during severe accident. Due to the high Ra numbers and high turbulence in the melt pool, LES method was chosen to perform the transient calculations. It is believed that the LES method can better capture the heat transfer characteristic in a high-Ra number volumetrically heated liquid pool. The numerical results of pool temperature and heat flux distribution obtained with FLUENT were compared with those from COPRA-L1 tests. Moreover, both the slice pool and hemispherical pool were calculated and compared to account for the geometry effect. These preliminary results revealed the need to modify FLUENT solidification model by using a user defined function. It is also planned to use a similar procedure to modify simulant component properties.

In the frame of ALISA (Access to Large Infrastructures for Severe Accidents between China and Europe), further research work proposed by KIT and EDF on melt heat transfer needs to focus on the influence of dimension and scaling effect to reduce uncertainty of the correlations mentioned above.

(e) Use of CFD for IVMR Studies, C. Le Guennic, EDF, France

The presentation showed CFD simulations of BALI tests for in-vessel corium pool behavior, and ULPU and SULTAN tests to assess NEPTUNE-CFD computer code capabilities in predicting ERVC, natural circulation flow.

IVMR evaluations are generally performed using integral codes, which rely on experimental correlations to assess the flux from the pool. Apart from the uncertainties regarding in-vessel corium behavior and that arise from the significant scattering of the various Nusselt number correlations currently in use, this modelling is problematic for several reasons: These correlations, which have been developed based on steady-state measurement, might not be suited for the transient state; additionally, due to differences in geometry and Rayleigh number ranges, they might be used out of their validity domain. The standard modelling for light metallic layers heated from below has been observed to overestimate the lateral heat flux for thin layers; the thinner the layer, the more important this deviation becomes. Similarly, very simplistic assumptions are used to assess the cooling loop's heat extraction capacity.

The maturation of CFD software, associated to the increase in computing power, opens new possibilities for the thermal-hydraulic study of corium pools and external vessel cooling. CFD simulations appear of interest to assess specific in-vessel corium thermal-hydraulic phenomena and can yield quantitative results that could be used to improve the modelling in integral codes as pointed out in the previous presentation. CFD studies related to IVMR have recently been launched at EDF; some of them are performed within the framework of the European H2020-IVMR project. Four different topics have been identified and are currently under study: ERVC, thin metallic layer thermal-hydraulics, homogeneous pool behavior and phase entrainment caused by interfacial shearing. NEPTUNE_CFD, a French 3D local CFD code developed in the frame of an EDF-CEA-AREVA-IRSN joint research and development programme, is the CFD computer code chosen for this purpose.

Since NEPTUNE_CFD has neither been developed nor validated for IVMR applications, suitable experiments must be identified, computed and added to the validation basis to test the code's capabilities. Such experiments have been chosen for ERVC, thin layer thermal-hydraulics and homogeneous pool convection, with generally good results.

Tests from the BALI-Métal and BALI campaigns have been simulated to assess the code capacity for thin metallic layers and homogeneous internally heated corium pools, respectively. Both tests use water as a simulant, and provide insight into the flow structure in the pool. The first results, using a higher order RANS (Reynolds Averaged Navier-Stockes) approach using RSM (Reynolds Stress Equation Model) turbulence modeling, are satisfying: the dimensionless numbers (e.g. Nu, Ra) are within the correct order of magnitude, the general flow structure is well reproduced, and the thermal behavior is in good agreement with the experimental data. However, no validation with prototypical materials could be performed yet. Exploratory computations for stratified pools were also performed, considering prototypical layer compositions. Additional data will be needed for future prototypical computations.

For ERVC, simulations with NEPTUNE_CFD of tests from the ULPU and SULTAN campaigns were performed, with positive results. However, those simulations showed that (i) a 3D simulation was necessary to account for the head losses caused by the configuration of the test facility (head losses caused by the front and back wall); (ii) there was a need to recalibrate the 4-flux wall heat transfer model for low pressures that led to significantly overestimate steam production; (iii) a thermal coupling with a solid conduction code improved the stability and avoided unphysical temperature discontinuities along the heated wall; (iv) there was significant impact of the outlet boundary conditions that could cause flow recirculation and disturb temperature and void fraction profiles. These encouraging first results, for in-vessel corium pools, need to be consolidated on a wider validation basis. Input parameters for prototypical computations are also under consideration, based on recent and upcoming experiments.

(f) Structural Integrity Research for Reactor Pressure Vessel under In-Vessel Melt Retention, Y. Gao, SNERDI, China

The speaker presented an RPV structural integrity study under IVMR. As stated earlier, the long-term integrity of the RPV considering ablation (i.e. thinning of the RPV wall) by the molten pool and survivability of penetrations and welds at the RPV LH needs to be ensured.

A Finite Element Model for an RPV considering lower head melting was established, the creep calculation was carried out after the temperature field analysis, and the stress-strain responses for different times were obtained with an ANSYS finite element model (2D axisymmetric model).

By means of choosing representative evaluation sections and applying the Accumulative Damage Theory based on Larson-Miller Parameter, the Creep Damage calculations and evaluations were conducted for the CAP1400 RPV. The results showed that the failure modes associated with creep rupture would happen only in local region through the RPV wall thickness under the considered IVMR conditions when a certain amount of internal pressure sustained. The approach, based on Larson-Miller Parameter method and total damage, used in this study could be extended in structural integrity evaluation of RPV under IVMR for other types of reactors.

(g) Original Core Catcher Design to Manage the Core Meltdown Severe Accident with the In-Vessel Retention Strategy, D. Aquaro, University of Pisa, Italy

The speaker's presentation was on a conceptual internal core catcher. The presentation addressed the design and the thermal analyses of this core catcher that were performed at the Department of Civil and Industrial Engineering (DICI) of the University of Pisa (Italy), allows to manage the core meltdown severe accident with the strategy known as In-vessel Retention.

The concept of the internal core catcher aims at retaining the corium inside the RPV by avoiding the contact between corium and the bottom head or at least to delay this contact in manner that the decay power decreases and means of cooling would be available and could be used.

The envisaged solution of core catcher is made of alumina pebble bed, located in alumina boxes; this pebble bed is expected to reduce by an order of magnitude the material conductivity. The boxes are assembled like bricks of masonry, supported by means of a CMC (SiC-SiC) framework. The internal and external liners of core catcher are made of high alloy steel. The alumina pebble beds have a high refractoriness and are expected to accommodate the thermal expansion without developing high thermal stresses.

The core catcher is designed for a PWR. It is 500 mm thick and externally lined by high alloy steel plates (internal liner 20 mm thick, external liner 50 mm thick). The internal cavity is filled of the pebble bed boxes (about 100 mm x100 mm x100 mm). The boxes are supported from four CMC frameworks. The purpose of these frameworks is to give a structural resistance of the core catcher until the alumina fusion temperature.

Based on Theophanous previous work and simple thermal assumptions in the oxidic pool and the metallic layer, numerical thermal analyses showed that owing to the thermal resistance of the core catcher (i) the transfer of all the power of the corium towards the upper surface of the metallic layer; and (ii) the core catcher is able to contain the corium for several days after the core melt down if the corium power decay is removed from its upper surface by means of various heat transfer mechanisms (e.g. convection with the water injected in the vessel).

A coupled thermo-mechanical model using a finite element method (FEM) code applied to the most severe thermal conditions showed that about half of the thickness of the core catcher is molten but that external temperature remains in the range of 500 K and 900 K. It is assumed that the pebble bed is molten, and the conductivity is set equal to that of alumina when the temperature of alumina reaches the fusion temperature (i.e. 2318 K).

While this theoretical model shows the feasibility of an internal core catcher with alumina pebble bed with low conductivity and high loading carrying capability, the used CMC are new materials whose applicability in the nuclear field needs further analysis. Also, the interaction between corium and alumina needs to be studied in detail to assess their chemical compatibility.

2.4.2. Summary of the session

Seven (7) presentations were given: five on molten pool behaviours; one on RV integrity; and one on an innovative concept of 'in-vessel core catcher'.

It is understood from the past research that molten pools can separate into either 2 layers (i.e. lower oxidic and upper light metal layers, or upper oxide and lower heavy metal layer) or 3 layers (heavy metal, oxidic and light metal layers), and there forms an oxidic crust between the oxidic and light metal pools.

During the past years a significant progress has been achieved in understanding and modelling of behaviour of molten pools in the RV, and new tests provide data that allow to be used at reactor-scale conditions, which was difficult to achieve.

Larger scale corium tests are desirable to provide more realistic data, and it is still open how to compare results obtained in facilities having different scales and what is the influence of geometry of test facilities. One possible solution is to compare the ratio of downward to upward heat flux at two different scales.

It appeared from the discussions that behaviour of stratified molten pools is still a key issue and requires additional information regarding experimental data and material properties. Behaviour of the upper metal layer and impact of its thickness on heat flux focusing (so called Focusing Effect) is a subject of research in many organisations. Visualisation of interfacial crusts and convection patterns in melt layers of different thickness will help in understanding and modelling the relevant phenomena. It was also pointed out that transient behaviour of molten pools is important to determine local heat flux values which could result in larger threat to the integrity of the RV than the fully-developed (steady) state.

Regarding behaviour of heterogeneous debris bed, limited penetration of liquid metal into oxide bed was observed in previous tests. Recent test results indicate that metal can penetrate through the oxidic crusts, which might change view of molten pool behaviour.

The other success criterion of the IVMR strategy is 'structural criterion' to ensure the long-term integrity of the RV, including survivability of penetrations and welds at the RV lower head. The structural integrity has to be evaluated based on detailed phenomenology and material properties measured at experiments representing realistic severe accident conditions.

2.5. SESSION 1D: APPLICATION OF IVMR TO SPECIFIC REACTOR DESIGNS

The fourth Technical Session was dedicated to the applications or the aim of application of IVMR strategy to specific reactor designs and consisted of six (6) presentations that addressed IVMR strategy for different reactor technologies and designs, including CAP1400, WWER-440, WWER-1000, BWR and PHWR.

2.5.1. Summary of the presentations

(a) CAP1400 IVR related Design Features and Analysis Methodology, G. Shi, SNERDI, China

This presentation highlighted the CAP1400 IVMR related design features and analysis methodology. He recalled the extensive studies of IVMR-relevant phenomena that have been performed or are underway and which include core melting and relocation, corium metal interaction, heat flux distributions to RPV from different corium configurations in lower plenum, RPV structural analysis, and CHF testing. For each topic, he referred to the corresponding presentation provided by SNERDI in the previous sessions.

Design features are implemented in CAP1400 design to increase IVMR strategy reliability. They include (i) RCS full depressurisation; (ii) absence of lower head penetrations to limit the RPV failure mechanism; (iii) designed vessel insulation to enhance RPV cooling by water and for steam venting; (iv) containment geometry and cavity flooding in order to easily flood the reactor cavity to sufficient level with in-containment refuelling water storage tank (IRWST) water, and to flood in-vessel core debris if RCS is fully depressurized and the break location is below water level; (v) containment passive cooling by condensing steam and return it to the cavity; (vi) lowering core support plate, thus increasing core internal mass and preventing focusing effect in transient; (vii) core shroud sitting on core support plate also increasing metal mass to mitigate focusing effect in the final state, and (viii) injection of water inside the RPV not later than the formation of corium pool in the lower plenum in order to cope with the uncertainties related to the formation of a conservative three layer (thin top metal layer, oxidic layer and bottom metal layer) configuration as a result of physico-chemical phenomena.

IVMR analysis methodology is applied to deduce an IVMR decomposition event tree (DET). Three steps (core heat up and melting, core relocation and corium pool in lower plenum) are considered with their corresponding in-vessel phenomena as well as the associated criteria (structural criterion or thermal criterion).

It is considered that metal control rods and un-oxidized zirconium in the core melt first and drain downward into the cooler regions of the core; in later phases of core melting, un-oxidized zirconium is expected to join the metal layer due to the absence of pathway to join the oxide layer and normally does not participate in thermodynamic interaction. Therefore, the possibility of two-layer corium pool is more likely. However, the situation of any un-oxidized zirconium existence in core pool cannot be ruled out. It is considered that if internals in lower plenum and partial (50%) un-oxidized zirconium participate in thermodynamic interaction, three-layer corium configuration is possible. The conservative three-layer configuration is unlikely and assumed when all un-oxidized zirconium and maximum steel is mixed in the lower plenum debris bed such that the density of heavy metal layer equals the oxide's one and that the upper metal pool thickness is minimized. The heat transfer analysis showed that heat flux in top metal layer can exceed CHF in case of conservative three-layer pool for accident sequences with large decay heat, and that the heat flux to RPV would decrease to values lower than CHF if injection of water inside the RPV is provided not later than the formation of corium pool in the lower plenum.

According to the frequency of each end state of the DET, the presentation concluded that IVMR failure contribution remains relatively small compared to the whole frequency even if the probability of the three-layer pool was conservatively set at 0.5 and that CAP1400 IVMR was efficient.

(b) R&D Activities to Resolve IVMR Strategy for VVER-1000 Reactor, J. Duspiva, UJV Rez, Czech Republic

This presentation discussed the R&D activities conducted to prepare decision making regarding possible IVMR strategy application to WWER-1000. The UJV support the NPPs in Czech Republic in the area of severe accident issues for 25 years. UJV activities strongly accelerated after Fukushima Daiichi accident NPP and their focus was reoriented from more academic proposals to final design solutions.

Some of SA issues have been already resolved and appropriate modifications have been implemented at Temelin NPP (WWER-1000/320). However, the issue of corium stabilization and cooling is more complex and needs much more R&D activities. Both IVMR and EVCC are under investigation for WWER-1000, the main goal of the solution being to maintain the containment integrity for the whole course of a severe accident or at least to significantly extend the time to the loss of containment integrity to enable performing immediate emergency response activities.

Activities to resolve the issues of IVMR applicability to Temelin NPP consist of many steps, some of them are completed, some of them are on-going and the remaining are on the list as they must be solved consequently. During the initial step it was necessary to identify what R&D activities were needed for the final resolution and what was already known.

The activities can be distinguished into three groups: (i) The first group covers identification of possibilities for pouring water in reactor cavity to enable reactor vessel cooling, and of sufficient paths for steam removal from the cavity, as well as ensuring reactor cavity leak-tightness; those activities are mostly already done; (ii) The second group is focused on analytical activities and covers initial identification of strategy boundary conditions (corium masses, decay power in corium, timing and so on), preliminary calculations of cooling capability in the cavity plant specific geometry, analytical support to the preparation of experimental facility (e.g. scaling, specific dimensions, heat flux distribution, to enable design of facility as well as the definition of the test matrix), evaluation of experimental results and their representativeness for the plant conditions, evaluation of safety margins (from thermal hydraulics as well as structural integrity point of view), evaluation of the containment loads and need and possibilities for the control of containment conditions (proposals for the heat removal strategies or retrofit); (iii) The third group supports the experimental programme based on two test facilities (small scale facility BESTH2, and large scale facility (THS-15) under construction).

BESTH2 facility was used to identify CHF at various conditions of cooled surface, cooling medium and inclination. This small facility was useful for gaining skills how to perform tests and to measure appropriate values (to be applied later in designing and using THS-15 facility) and for obtaining the first data on impact of various surface on CHF. More than 100 tests were performed at BESTH2. Tests at THS-15 are expected to be performed in middle of 2017. The last part of the experimental programme will focus on investigating material properties of vessel steel at high temperature.

Complementary to of the activities on IVMR applicability, investigations of the consequences of IVMR failure with release of corium into the flooded reactor cavity (e.g. steam explosion) started on plant specific basis and expected to last at least two years.

The R&D programme on the applicability of the IVMR strategy is very complex and international collaboration in such a case is very important and helpful. Therefore, UJV participated in the analytical benchmark on simulation of the IVMR strategy applied to a WWWER-1000 unit (finished in 2014), and is now taking part in the Horizon 2020 project IVMR (2015-2019), through contribution in analytical activities (WP2) and experimental activities (WP5) with small and large-scale tests.

(c) A case Study on WWER-440 for Severe Accidents with Core Meltdown and Reactor Vessel Destruction, T. Sargsyan, Armenian NPP, Armenia

The presenter discussed the results of a case study of severe accident with core melting and corium-concrete interaction for a WWER-440/270.This case consisted of a break of 100 mm on the primary circuit with loss of high pressure safety injection system and spray system, the only operator action being main coolant pumps switching off, 10 minutes after reactor scram.

The case study was aimed at determining the accident progression chronology up to containment failure. The study included also distribution of fission products in the core, primary side, secondary side, containment, reactor cavity and releases to the environment. The calculations were performed with MELCOR code while specific Armenian Nuclear Power

Plant (ANPP) containment concrete structure data was used. It is estimated that the RPV failure occurs, for this specific scenario, 3 hours and 47 minutes after reactor scram and that the loss of containment integrity occurs 40 hours after reactor scram. In addition, alternative ways of using permanent and mobile systems for corium cooling and localization were assessed. The results are expected to be used in PSA Level 2.

The presentation included also insights from a preliminary assessment of application of IVMR strategy to ANPP. The feasibility of IVMR strategy was concluded; however, confirmatory analytical calculations are necessary, regarding the flow rate of cooling water in the reactor cavity, its temperature, possibility of steam condensation on the surfaces of a specially developed passive heat removal system; the water reserve needed for at least 14 days and severe accident management to avoid early corium slumping in the RPV lower head following early core melting.

(d) Introduction of NPIC Work on In-Vessel Retention (IVMR) Strategy, D. Zhu, NPIC, China

This presentation showed an overview of the R&D activities related to IVMR as part of the design of ACP1000 and ACP100.

ACP1000 design includes active and passive systems such as emergency core cooling, core residual heat removal, containment heat removal and reactor cavity injection and cooling for IVMR. Injection lines are directly connected to insulation channels using fire water as source and IRWST water as the recycle water. In case of SBO, a dedicated passive water tank is used. Insulation design is optimized based on CFD analysis. A computer code, CISER code, was developed to assess IVMR for typical severe accident sequences. A two-layer corium pool configuration is considered with statistical evaluation of uncertainties of some parameters, e.g. metal layer thickness. This analytical effort is complemented by an experimental programme to obtain CHF and heat transfer coefficients for RPV cooling in different conditions. Impacts of parameters such as inlet sub-cooling and water flow rate on CHF are investigated.

ACP100 is characterized by an elliptical lower head and passive reactor cavity flooding and cooling. IVMR design for ACP100 is still ongoing. A simplified 2D model using SCDAP/RELAP5 showed an acceptable margin to CHF (more than 30%). However, CFD study of CHF showed the need for further analytical and experimental investigations.

(e) Application of IVMR in AREVA's Gen-3 Advanced BWR (KERENA), M. Fischer, AREVA GmbH, Germany

The presenter discussed the application of IVMR in AREVA's Gen-III advanced boiling water reactor KERENA (former SWR-1000). Its advanced safety concept is founded on the interplay of reliable active and passive systems for water supply into the RPV and heat removal out of the containment. Consequently, the predicted probability of a severe accident with core melting is extremely low.

Despite this low value, the KERENA reactor involves dedicated design measures for severe accident mitigation. The related measures which further reduce the frequency of large activity releases into the environment include application of IVMR strategy, with passive flooding of the reactor cavity and the cooling of the RPV from the outside. It is stated that reactor cavity flooding relies on internal water reservoirs only and diverse and independent components and signals.

For the validation of the KERENA IVMR function not only thermodynamic but also thermo-chemical phenomena have been considered. Among the latter are:

- the potential formation of a dense metallic phase within the oxidic pool;
- the relocation of this dense metal phase to the top and its impact on the focusing effect;
- the influence of Boron and Carbon on metal and oxide density.

In addition, the presence of an oxidic crust at the surface of the top metal layer was accounted for. The formation of such a crust which, under BWR conditions, mainly consists of zirconium dioxide and zirconium nitride, was confirmed in dedicated experiments. A related analysis demonstrated that already thin crusts are capable to insulate the surface and aggravate the focusing effect.

The relevance of the above listed phenomena on IVMR in the KERENA BWR was quantified in dedicated plant analyses. They confirm that local heat fluxes to the water on the outside of the RPV remain well below critical values, even when applying generally conservative assumptions. This is mainly due to the high mass of internal steel structures inside the KERENA RPV.

The required heat removal capacity of the KERENA lower head was experimentally confirmed using a 1:1 section of the lower head (incl. penetrations) which was electrically heated on the inside. To allow appropriate 2-phase flow and to keep the RPV intact under IVMR conditions, the design of the insulation, the RPV support and the instrumentation and control rods, including their penetrations, had to be specifically adapted. All penetrating elements are mechanically supported with consideration of RPV expansion; only small circumferential gaps exist inside the tubes which are not empty. Deformation and stability analyses have been performed using CFD methods.

According to the presentation, the performed deterministic analysis was complemented by PSA, which showed that owing to the high reliability of the KERENA safety systems, passive in-vessel water injection can re-arrest the melting core in an early state for most accident scenarios. Consequently, a large molten pool inside the lower head can form only after a preceding complete failure of the passive in-vessel flooding function. Even then, the large coolant reservoirs inside the KERENA containment help delaying RPV dry-out and melt relocation into the lower head over more than 48h, even in case of a postulated large break outside the containment. This long grace period effectively reduces the decay power level in the molten pool, the heat fluxes during IVMR and hence the risk of focusing effect.

However, it is recognized that uncertainties exist regarding heat flux from the melt to the vessel challenging IVMR robustness. IVMR application requires an acceptable level of robustness against RPV failure. Otherwise, risks related, steam explosion with the reactor cavity fully flooded, must be analysed in detail.

(f) In-Vessel Retention Analysis for Pressurized Heavy Water Reactors (PHWR) under Severe Core Damage Accident (SCDA), O. Gokhale, BARC, India

The speaker presented IVMR analysis for PHWR calandria under severe accident with core melting. The Severe Core Damage Accident (SCDA) scenario for PHWRs are defined as accidents involving loss of core configuration typically involving disassembly of fuel channels into horizontally placed cylindrical calandria. Unlike in case of LWRs, the debris are long pipe like structures at near saturation temperature. This debris configuration is widely different from debris configuration obtained from TMI-2 accident as well as different molten fuel coolant interaction (FCI) experiments. The boil-off of the moderator followed by heat-up of debris might lead to melting of debris forming molten magma layers. However, this is strongly governed by the dimensions of the calandria and fuel inventory and the reactor power. Light water present in the vault surrounding the calandria provides a large heat sink for decay heat removal from debris/magma. Addition of water into the vault is one of the prescribed SAMG action for Indian PHWRs to facilitate cooling of debris/magma.

Analyses have been carried out with ASTEC computer code to assess the capability of 220 MW(e) and 540 MW(e) calandria to retain debris/magma under such SAMG action. The PHWR specific module can model horizontal reactor core and heat-up of different components of reactor block like tube sheets of end shield (ES), carbon steel balls within the ES, baffle plate, octagonal flanges and liner tubes. The detailed modelling enables to account proper heat losses from debris bed to surroundings, which in turn, will predict a realistic heat-up of debris and calandria. The radiation heat transfer between debris/magma and the calandria and between the calandria and the vault wall has been exclusively modelled with radiosity approach. The heat exchanges to the ES that also acts as heat sink have been accounted for this analysis. A heat transfer package has been developed for ex-vessel cooling.

The analysis predicts that in case of an un-mitigated SBO the ex-vessel cooling of debris retained within the calandria is found to be successful for 220 MW(e) and 540 MW(e) Indian PHWRs. In case of 220 MW(e) PHWR (low power density and large calandria diameter ensuring sufficient heat transfer area), the heat-up of the debris is even limited to temperatures below the melting temperature till complete boil-off of the vault water. Time available for SAMG actions (injection of water in the calandria vault) is found longer than two days from SBO initiation. In case of 540 MW(e) PHWR, debris temperature rises after complete moderator boil-off and reaches melting temperature and there is conversion of solid debris in molten magma. Time available for SAMG actions (injection of water in the calandria vault) is longer than one day from SBO initiation. In both cases, there is time grace for addition of water in the calandria vault that helps maintaining calandria integrity.

2.5.2. Summary of the session

In general, the application of IVMR requires the following considerations, and various kinds of active and passive systems are designed:

- o Depressurization of the reactor coolant system (RCS);
- o Water source for EVRC using existing tanks or new dedicated tanks;
- o Initial flooding of the reactor cavity, followed by a long-term water supply;
- o Venting and condensation of steam generated in the reactor cavity; and
- o Potential negative impacts if the IVMR with external cooling fails (e.g. threat of steam explosion in the containment).

Although different types of reactors have different systems, internals, accident sequences and operational procedures, the following suggestions were made for possible common efforts:

- o Establishment of a PIRT to identify key phenomena;
- o Analysis codes benchmarking and validation;
- o Molten pool behaviour experiments, including transient and 3-layer behaviours, with higher fidelity and representativeness;
- o Large scale experiments with shared test matrices; and
- o Steam explosion phenomenology.

2.6. GENERAL DISCUSSION ON IVMR

Based on the presentations and discussion at Technical Sessions 1A-1D, the TM participants discussed general issues on IVMR, including:

1) Current Status and Recent Progress of IVMR R&D;
2) Remaining Challenges and Open Issues; and
3) Proposals for International Cooperation.

Regarding the current status and recent progress, it was discussed and agreed among the TM participants that:

- New approaches need to better consider the possibility of existence of a heavy metal layer and the consequences on stratification (2-layers or 3-layers configuration); and
- Design options to improve efficiency of IVMR strategy are well identified and could be implemented in new reactor designs such as: 1) large mass of molten steel to limit focusing effect; 2) no penetrations through the RV lower head; and 3) simultaneous in-vessel water injection.

It was also discussed that timely water injection into the RV can arrest core damage and prevent vessel failure. This action is particularly necessary for plants with lower head penetrations because penetrations and internal closure welds are generally more vulnerable than the rest of the lower head.

The participants discussed remaining challenges and open issues on IVMR, and pointed out the following challenges and issues:

- A lot of CHF data are available now, but some of them are contradictory (e.g. the effect of surface oxidation);
- Experimental results that need to be s and presented in a consistent way to be used easily by analysts, designers and regulators;
- Focusing effect and transient behaviour of molten pools is an important issue: the current models need to be further improved;
- Different models and codes produce quite different results partly due to the absence of a proper validation matrix for IVMR-related phenomena and due to user effect (or the lack of training of users for IVMR specific models);
- Quantification of the risk associated with IVMR failure: the energetic phenomena after melt release into a flooded cavity need to be adequately understood;
- There are not enough structural integrity analyses made for different shapes of vessel (considering ablation), different pressure and temperature conditions, and there are insufficient data for material behaviours of vessel steel.

Innovative ideas and concepts should be more encouraged. They might be difficult to be adopted directly in the design. However, new ideas could stimulate new perspectives on the relevant phenomena and methodology.

2.7. SESSION 2A: GENERAL CONSIDERATIONS ON EX-VESSEL CORIUM COOLING STRATEGY

The fifth Technical Session was dedicated to general considerations on EVCC strategy and containment integrity. Four (4) presentations were provided to illustrate CEA R&D activities on EVCC for a PWR, US DOE severe accident research on IVMR and EVCC, improvement of FCI models for ex-vessel debris coolability in Japan, simulation of Molten Corium Concrete

Interaction (MCCI) for CANDU accident late phase, and a summary of a very recent OECD/NEA report on State-of-the-Art Report on Molten-Corium-Concrete interaction and Ex-Vessel Molten-Core Coolability.

2.7.1. Summary of the presentations

(a) CEA R&D Activities related to Ex-Vessel Corium Behaviour in a PWR, B. Teisseire, CEA, France

This presentation provided an overview of CEA R&D activities related to ex-vessel corium behavior in a PWR carried out at CEA during the last twenty years.

The CEA severe accident programme aims at providing tools and expertise for severe accident phenomenology understanding and for their management. The scope includes: (i) for safety improvement of LWR power operating reactors in the frame of life time extension or for the design of severe accident features to mitigate them (IVMR or EVCC) for new power reactors; (ii) non-power reactors such as research reactors or reactors used for nuclear propulsion; (iii) post-Fukushima expertise to estimate the state of the reactors and/or support the decommissioning.

The corium behavior programme which is an important part of CEA severe accident programme consists of:

- Performing simulant and prototypical material experiment to improve the knowledge of the main relevant phenomena;
- Establishing qualified physico-chemical and thermodynamic databases;
- Developing and validating of both scenario and mechanistic computer codes for reactor calculations.

The main severe accident R&D corium issues are IVMR which has been extensively discussed in the previous sections, and EVCC which could involve ex-vessel corium retention in the reactor cavity (corium spreading on dry or under water conditions, debris bed coolability and Molten Core Concrete Interaction under dry or wet conditions), and FCI with corium fragmentation and possibly steam explosion.

Several provisions have been studied to quench and stabilize the corium into a solidified mass and prevent the basemat failure. One of these consists in top flooding the reactor cavity after corium relocation and spreading over the concrete basemat. Past experimental programmes have brought out that water ingress and corium eruption processes could increase heat transfer between corium and water and thus slow down the concrete melting. It appears that the cooling efficiency depends mainly on the concrete and corium compositions.

Moreover, in some scenarios, a corium jet could be ejected into a partially or totally flooded reactor cavity, leading to FCI. Then, fragmentation of the corium jet occurs potentially leading to steam explosion challenging the reactor containment integrity. Current and future experimental programmes planned at CEA aim at bringing additional data on spreading, MCCI and FCI phenomena to improve their understanding and their modeling in simulation tools to perform reactor calculations in support to safety assessment and to severe accident guidelines' optimization.

To lead these R&D programmes, CEA relies mainly on the experimental PLINIUS platform which has the peculiarity of using prototypic corium. PLINIUS platform includes KROTOS, VULCANO and VITI facilities. The KROTOS facility allows to study the interaction of a molten corium jet with water (FCI) in the reactor cavity whereas the existing VULCANO facility can perform dry MCCI and spreading experiments and the MERELAVA under construction facility is dedicated to simulating the corium pool top or bottom cooling. The VITI facility (using induction heating) and the ATTHILA facility (using laser heating device) enable to measure the thermo-physical and the thermo-chemical corium properties respectively. Those properties are fundamental data for corium behaviour analysis. Some of these R&D activities have been launched in the frame of Post-Fukushima research programme launched by the French government to strengthen nuclear safety and radiation protection. Improvements of the equipment and instrumentation (on-line visualization of the corium) and new experimental devices construction are under progress to enhance phenomenology understanding and knowledge and to increase the reliability of the data.

In parallel, computer codes such as THEMA (ex-vessel corium spreading under dry and wet conditions), MC3D (FCI and steam explosion) and TOLBIAC-ICB (ex-vessel corium coolability and MCCI) are being developed and/or are being validated against PLINIUS experiments but also against international programmes with prototypic corium (COMAS and FARO for THEMA, FARO for MC3D, CCI, SSWICS, SURC, ACE, MACE for TOLBIAC-ICB) or simulant materials (CORINE, KATS, ECOKATS for spreading, TREPANS, DISCO, BILLEAU, QUEOS for MC3D and BETA, COMET-L, ARTEMIS for TOLBIAC-ICB).

Currently, prototypic corium facilities are limited to masses in the range of 50-80 kg although needs exist for larger mass experiments: FCI experiments with stationary corium jet are needed to complement the data from the now dismantled FARO facility. Underwater spreading and MCCI experiments also require large masses of corium.

This has led CEA to launch the design phase for a new large-scale prototypic corium experimental platform, PLINIUS-2, to consolidate corium behaviour modelling, model development and code validation with realistic corium, as well as to support the development of new mitigation concepts for both operating and new reactors with conditions as close as possible to those in reactor case. The target for PLINIUS-2 is to melt and study behaviour of several hundreds of kilograms (up to 500 kg) of prototypic corium. PLINIUS-2 platform will allow investigating issues such as: (i) IVMR oxide/ metal stratification and focusing effect (with on-line instrumentation to catch kinetics); (ii) FCI with water fragmentation, debris bed formation and steam explosion; (iii) corium spreading with several 10 cms of water, and (iv) MCCI top flooding and other mitigation concepts. PLINIUS-2 benefit from national and international collaborations; first test is planned in 2021.

(b) US Department of Energy Severe Accident Research in the Area of In-Vessel Core Melt Progression and Ex-Vessel Debris Coolability, M. Farmer, Argonne National Laboratories, USA

The presenter discussed an overview of US Department of Energy (DOE) severe accident research in the area of in-vessel core melt progression and ex-vessel debris coolability as part of the Light Water Reactor Sustainability (LWRS) Programme, and more precisely in the Reactor Safety Technologies (RST) pathway that provides insights, data, analyses and methods that are intended to support industry efforts to enhance nuclear safety in addressing severe accident conditions.

The DOE has played a major role in the US response to the reactor accidents at Fukushima Daiichi. In the months that followed these events, a coordinated analysis activity aimed at gaining a more thorough understanding of the accident sequence was completed using laboratory-developed system-level (i.e. MELCOR) and best-estimate (i.e. MELTSPREAD and CORQUENCH) codes, while a parallel analysis was conducted by US industry with their system-level accident analysis code (MAAP).

Large variations in plant accident response predictions with MELCOR and MAAP included in-core hydrogen production, melt relocation behavior to the RPV lower head and melt pour conditions to the reactor cavity following RPV failure. Those variations were due to the assumptions regarding the extent that in-core blockages were permeable to gas flow. They led to the need to perform a technology gap evaluation on accident tolerant components and severe accident analysis methodologies with the goal of identifying any data and/or knowledge gaps that could exist, given the current state of LWR severe accident research in the light of insights from Fukushima Daiichi accident. This effort identified and ranked a total of 13 knowledge gaps (e.g. in-vessel assembly/core-level degradation, wet reactor cavity melt relocation and corium concrete interaction) that were not being addressed by US industry, DOE, or the NRC. On this basis, DOE launched various R&D activities within the RST pathway of the LWRS programme, several of which are related to in-vessel core melt progression that defines the initial and boundary conditions for core melt behavior in the reactor vessel lower head, as well as ex-vessel core debris relocation (i.e. core melt spreading) and long-term debris cooling behavior.

The presentation highlighted the efforts aimed at developing tools for quantifying the extent of core melt spreading and debris coolability in reactor containments. Those efforts are motivated by the Severe Accident Water Management (SAWM) approach that the US industry is pursuing as an alternative to installing filters on Mark I and Mark II containment vents requested by Order EA-13-109 to provide containment protection and risk reduction. The objective is to upgrade existing ex-vessel severe accident analysis tools (MELTSPREAD and CORQUENCH) to support the optimization of SAWM strategies that preserve the wetwell vent path while maintaining core debris covered with water. The following tasks were or are being performed:

- Upgrade, validate and document MELTSPREAD for industry use to make realistic estimates of core debris location following the spreading phase: a melt pour jet fragmentation model in water and a realistic water inventory for SAWM analysis has been added. Melt interaction model with below vessel structure was in progress at the time of the TM.
- Upgrade, validate and document CORQUENCH for industry use to model MCCI with realistic debris coolability and cavity flooding models: CORQUENCH was made multi-nodal to address realistic containment geometries and core debris distributions and a realistic water inventory for SAWM analysis has been added.
- Provide analysis with the upgraded models to support industry in optimizing SAWM strategy: the analysis of Fukushima Daiichi Unit 1 accident (long term phase) performed with the enhanced MELTSPREAD and CORQUENCH tools provided valuable insights regarding the extent of MCCI, importance of water injection location in accident management in case of low water injection flowrates, and extensive particle bed formation below the RPV.
- Conduct core debris coolability experiments to provide further data for upgraded coolability models' validation: large scale reactor material experiments, sponsored by EDF, IRSN, CEA, USNRC and DOE and using conclusions from previous OECD/MCCI Project, are being carried out. These experiments are investigating core debris coolability under different cavity flooding conditions and concrete compositions.

(c) Improvement of Fuel-Coolant Interaction Models for Ex-Vessel Debris Coolability Evaluation, T. Matsumoto, JAEA, Japan

The presentation discussed the improvements of fuel-coolant interaction models implemented in JASMINE computer code, developed by JAEA, for ex-vessel debris coolability evaluation.

As one of the accident managements to prevent the MCCI in a severe accident, the pedestal or cavity in the containment vessel would be flooded with water. The current regulation in Japan requires some measures to suppress the MCCI. In response, Japanese utilities are introducing facilities to inject water prior to the melt release from the RPV. The problem is the determination of a sufficient depth of water to avoid significant damages caused by possible steam explosion on, and to protect the containment boundary. After penetration at the surface of water pool, melt jet falls in the water with releasing liquid melt particles due to effects of the fluid dynamic instability at the melt-water interface (so-called jet breakup). The jet breakup particles are quenched and solidified during the fall in the water. Some of liquid particle would attach with adjoining particles and become the agglomerated debris with a relatively low coolability. If the depth of water pool is not sufficient for the entire jet breakup, the melt jet can reach the bottom of the pedestal. Then, a melt pool would be formed and would spread in the horizontal direction. The top and bottom sides of the melt pool are quenched by the surrounding water and floor materials. The crust layers grow at the both sides, which slows the melt pool spreading and finally stops it.

JASMINE main target was to simulate short time phenomena in several seconds while long time phenomena were ignored. However, evaluation of debris coolability requires treatment of longer time phenomena in several minutes to even hours. Therefore, the capability of the JASMINE code needs to be extended, regarding jet breakup model and melt spreading model. A more realistic approach considering uncertainties (on mass, temperature and composition of molten core, water level outside the vessel and physical models in the computer code), will be developed to assess the consequence of melt release in the reactor cavity and the probability of the cavity failure under various water depth conditions.

As for the jet breakup model, the procedure to generate diameter distribution of the jet breakup particles was improved to be consistent with the various experimental results. This distribution F is based on Rosin-Rammler cumulative distribution function which provides the cumulative mas fraction of particles whose diameter is lower than a given value. In JASMINE code, the mass is determined from jet breakup length and particle diameter values are given when breakup particles discharged in the water phase; particle solidification is due to heat removal. Uniform random numbers are generated using Monte-Carlo method and are used for the inverse function F^{-1} that is implemented in JASMINE code. The DEFOR-A jet breakup tests using $Bi_2O_3 + WO_3$ as corium simulant material and 4 catchers to collect falling molten/ solidified particles were analysed with the improved JASMINE code. The analysis results qualitatively correspond to the experimental observations. At the same time, the results suggested that the consideration of agglomeration of neighbouring melt particles could improve the simulation.

Concerning the melt spreading, the model of melt pool consists of three layers: top and bottom crust layers and the molten pool in between. Crust layer at the top and at the bottom of the melt was implemented in the JASMINE code and applied to the analysis of the PULiMS tests carried out by the KTH and using $Bi_2O_3 + WO_3$ as corium simulant material. The growth of the crust layer successfully stopped the melt spreading as intended. However, the analysis results showed a tendency to overestimate the test results (spreading area) by a factor between 1.7 and 2.1. Further improvement will take in consideration gas generated inside melt phase and thermal conduction with the reactor cavity floor.

In the future, JASMINE will be used as deterministic code in the evaluation procedure, to assess MCCI starting and the debris coolability given a reference probability.

(d) Simulation of Molten Core Concrete Interaction for CANDU accident late phase by MEDICIS module, M. Apostol, Institute for Nuclear Research Pitesti, Romania

The presenter discussed a simulation of molten core concrete interaction for CANDU accident late phase with MEDICIS module from ASTEC computer code

After Fukushima Daiichi accident, research and development (R&D) activities in SA area, particularly simulation of ex-vessel corium progression sequences and corium interaction with the structures, became one of the most important issues addressed at the Institute for Nuclear Research (INR) Pitesti. INR devoted important efforts to SAs research activities by including and studying them in national R&D programme and by participating in international Euratom projects (PHEBEN2, SARNET, SARNET2) and in the on-going H2020 FASTNET project.

The peculiarities of CANDU type reactors in initiation and progression of the severe accidents and the evolution of core degradation during severe accident progression were presented. For some initiating events (such as large LOCA, station blackout – SBO), the simulations performed with MAAP4-CANDU, indicated the possibility of CV failure, followed by the discharging and spreading of the corium into the rectangular cavity of CV. When the corium reacts with water from CV, strong energetic reactions appear, leading to the decreasing of the water level or even completely loss of water. If an insufficient cooling of the melt released from the CV occurs, the molten corium will enter into reaction with the concrete floor. In case of the penetration of the CV floor, the corium is relocated into the basemat of containment and can interact with the water sump. If the water is lost, the molten corium can produce the erosion of the basemat, with potential radiological consequences on people and the environment.

Accident Source Term Evaluation Code (ASTEC), is an integral code, jointly developed by IRSN (France) and GRS (Germany), which simulates a complete severe accident scenario in light water reactors, by coupling phenomena relevant for source term evaluation, PSA level 2 studies, evaluation of SAs management and support of experimental programmes. Applicability of different ASTEC modules to CANDU SA was investigated by the INR Pitesti as partner in Severe Accident Research Network of excellence (SARNET) and the specificities and needs for the development of the code were identified. Some improvements in models and algorithms were directly devoted to take into consideration CANDU peculiarities.

Model of Erosion Due to Interaction of Corium with basemat Substrate (MEDICIS), simulates molten core concrete interaction (MCCI) to investigate whether and when the basemat concrete could be completely ablated opening a pathway for fission products getting into soil and water, with radiological consequences on people and environment. MEDICIS includes (i) a model of the structure of the corium-concrete interface; (ii) models of corium coolability in case of water injection on the top corium pool surface; (iii) models of evolution of corium pool configuration; (iv) models to evaluate the release of concrete aerosols and fission products during MCCI; and (v) interface with the physico-chemistry Material Data Bank (MDB) to evaluate the corium layers' properties.

The application of MEDICIS module from ASTEC V2, in stand-alone mode to CANDU 6 reactor, supposed using of specific data (available from literature or calculated) and generic data. As specific data, uranium dioxide mass in the core, uranium mass, total mass of ZrO_2, Zircaloy mass in the core, thickness of the floor calandria vault, height and radius of the cavity, thermal reactor power, and masses of Fe, Cr and Ni in the core were used. The initial temperatures of oxide and metallic layers, the properties of calandria vault concrete and the gas pressure above the corium, which represent important data necessary as input in MEDICIS module, are not CANDU 6 specific. For simulation of MCCI for the late phase of accident, an initial stratified configuration of corium composed by metal layer, oxide layer and crust, configuration with progression in time was assumed.

To see the influence of some uncertain parameter (such as the starting time of MCCI, the mass of UO_2 and mass of ZrO_2, the solid concrete density, the initial temperature of metal layer and oxide layer) on the results on simulation of MCCI for CANDU accident late phase by MEDICIS module, sensitivity calculations have been performed. The results in terms of complete erosion of the CV floor using 1D model, suggest a significant influence of these parameters within the considered assumptions. The influence of the radial erosion was outlined, this significantly delaying the penetration of CV floor. It was shown that for SA initiated by LOCA, approximately 63 h (using 1D model), 93 h respectively (using 2D model), after the MCCI starting, are necessary to penetrate the CV floor. Some recommendations were formulated to improve the quality of the input data and the present model of MCCI for CANDU late phase accident, by considering the multiple cavities.

(e) Recent Progress in Phenomenology, Analysis, and Accident Management Strategies related to Molten Core-Concrete Interactions and Coolability, M. Farmer, Argonne National Laboratories, USA

The speaker presented an overview of the recent State-Of-the-Art Report (SOAR) on molten corium concrete interaction and debris coolability that was completed in the framework of the Working Group on Accident Management and Analysis (WGAMA) of the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency. The report documented the current knowledge base of the MCCI phenomena with reference to the role of water addition as a severe accident management strategy, and the state of maturity of modeling the phenomena in severe accident analysis codes at the time of the SOAR preparation. It highlighted that significant progress has been achieved in understanding MCCI phenomena with a view to assessing the efficacy of water addition to promote ex-vessel debris coolability, and identified a few issues that could warrant further investigation to reduce residual uncertainties. These issues include specific realistic reactor conditions from short term to long duration transients, improvement of plant simulations with more realistic input data and boundary conditions, and potential to improve the efficiency of melt coolability under top flooding conditions.

Lessons have been learned regarding nature and extent of core-concrete interaction for dry conditions (e.g. influence of melt temperature and ablation shape by concrete composition, existence of ablation burst, ablation seems more pronounced in the areas where metal is in direct contact with concrete), as well as effectiveness of water addition for melt stabilization for wet reactor cavity (e.g. identification of bulk mechanisms such as bulk cooling, water ingression, melt ejection and suspended crust breaching, and potential enhancement by early containment flooding).

The SOAR concluded that current computer codes can analyze ideal cases where corium is assumed to be instantaneously spread over the entire floor of a dry reactor cavity. Moreover, empirical correlations are needed to be implemented to consider ablation anisotropy (anisotropy of the lateral and vertical ablation of specific siliceous concrete) that cannot be explained by existing phenomenological models. In some computer codes, the effect of metal is described; however, considering metal effect introduces additional uncertainties as this effect is not yet well understood. Other computer code limitations are related to the consideration of local corium accumulations in the case where the reactor cavity is initially flooded, or in situations involving multiple melt pours (issues of debris bed re-melting and stability of corium accumulations). The SOAR also highlighted that very few codes can model the impact of top flooding; melt ejection assessment appears difficult due to melt crust anchoring phenomena observed at scaled experiments and not expected at reactor scale, and lack of data for top flooding of metal-oxide melt.

Extrapolation of scaled experiments' results to MCCI at plant scale through computer codes needs some idealization of plant geometry and configuration. Moreover, in plant safety assessment, a pragmatic approach is followed based on conservative assumptions regarding the weaknesses of the containment design.

To extrapolate to reactor situations with higher degree of confidence, there is a need to obtain longer term experimental data with the aim to:

- Confirm long term ablation as a function of concrete composition;
- Obtain data on long term behavior with heat flux low enough and concrete fraction in the melt is very high;
- Confirm the reproducibility of intermittent phenomena, e.g. melt eruptions;
- Investigate if the crust formed by water ingression remains sable.

Improvement of plant simulations need to consider more realistic input data, and initial and boundary conditions, metal presence within the melt or within the concrete (rebars) has to be considered in terms of influence on the ablation profile, metal cooling mechanism and ablation mechanism. In addition, the initial conditions for MCCI on melt pour conditions need to be improved by considering initial composition and temperature of the melt, as well as the risk of corium accumulation(s) in already flooded reactor cavity or in the case of successive pours after reactor cavity flooding. The presence of impurities in cooling water has to be considered also as they might impact water ingression mechanism and debris bed coolability; similarly, presence of fission products impacting source term in sump water, and modification of sump water chemistry that might have impact on sump filter clogging risk in case a cooling system is dedicated to avoiding containment venting.

Improvement of melt coolability under top flooding could be achieved by:

- Benefitting of a larger initial corium spreading area to reduce downward heat flux to the concrete and hence reducing its ablation;
- Avoiding as far as practicable wet reactor cavity because large spreading is more effective, and risk of steam explosion is eliminated in dry cavity;
- Taking benefit of early flooding after initial corium spreading; however, there is a time window as subsequent melt pour after top flooding cannot be excluded due to the uncertainties related to in-vessel melt progression;
- Benefitting of high carbonate or hydrate concrete for efficient melt ejection; this is particularly useful in the design of new reactors or back-fitting measure for siliceous basemat.

There is a need to obtain additional experimental results; this can be achieved by a balanced use of integral and separate effect tests (e.g. separate tests to address effect of water impurity). Most significant experimental challenges are related to:

- improve facility capabilities to avoid crust anchoring;
- heat meat-oxide melt;
- consider rebars in the concrete;
- heat solid particle debris under water up to the re-melting point and the onset of MCCI.

Due to the cost and complexity of such experiments, international collaboration is essential to foster know-how and cost sharing.

2.7.2. Summary of the session

This session provided overviews of R&D efforts related to EVCC, including a summary of a very recent OECD/NEA report on State-of-the-Art Report on Molten-Corium-Concrete interaction and Ex-Vessel Molten-Core Coolability. These presentations provided a good overview of the recent progress in experimentation, in gaining insights in phenomenology, analysis and accident management strategies related to molten core-concrete interaction (MCCI) and coolability, and revealed the challenges and remaining issues still to be addressed.

It is recognized that the EVCC strategy remains the last means to limit MCCI consequences on WCR containment integrity in case that IVMR is not applied or fails. Several NPPs under operation or under construction made the choice of EVCC strategy.

The insights from OECD/NEA's state-of-the-art report and national R&D programmes suggest that significant progress has been made in terms of understanding phenomenological behaviour, but also led to the conclusion that some issues may need further investigation to reduce current uncertainties.

These issues include those related to debris bed formation and coolability, e.g. melt jet fragmentation in water, formation and behaviour of mixed oxide/metal debris bed, post-dry out heat transfer and debris re-melting, oxidation effects, and molten core and concrete interaction (MCCI), e.g. lack of data for long duration transients, anisotropy of the lateral and vertical ablation of specific siliceous concretes, effects of molten metal layer in corium pool and reinforcement bars in concrete. It was agreed among the TM participants that depending on the design, the layout of the containment and the strategy steps, issues related to EVCC (e.g. steam explosion, MCCI) could be significant.

2.8. SESSION 2B: APPLICATION OF EX-VESSEL CORIUM COOLING TO SPECIFIC REACTOR DESIGNS

The sixth Technical discussed applications of EVCC strategy to specific reactor designs based on six (6) presentations that addressed EVCC strategy for different reactor technologies and designs, including PWRs in Belgium and France, WWER in Czech Republic and Ukraine, BWR Mark II in Mexico, and the evolution of the ex-vessel melt stabilization concept for the EPR™.

2.8.1. Summary of the presentations

(a) Status and the Strategies of Ex-Vessel Corium Cooling in Belgium, A. Malkhasyan, Bel V, Belgium

The speaker presented and reviewed the evolution of the strategies of EVCC in the Belgium NPPs considering plant features (e.g. design of reactor cavity and concrete composition) and progress in R&D which was used by the utility and carefully followed by the regulator body.

The first period of the review of the EVCC in SAMGs starts in the late nineties when the SAMGs were developed and implemented in Belgium. The IVMR strategy has been considered not applicable to Belgian units, based on the latest available knowledge at that time and the reactor design, therefore an ex-vessel melt coolability strategy has been chosen. The SAMGs included the experience of Westinghouse Owners Group, as well as benefitted from experimental data available at that time. Based on the results of the OECD MCCI programme and the results from PSA level 2 recommendations, the ex-vessel coolability strategies have been optimised for each unit. Different strategies (i.e. cavity flooding before or after vessel failure) have been selected depending on the unit specific layout. Early cavity flooding before RPV failure was preferred for units where jet breakup is possible and where reactor cavity flooding is ensured. Cavity flooding after RPV failure was preferred for units with no jet breakup and cavity is always wet or can be isolated. For units with no jet breakup and where the cavity cannot be isolated no clear decision could be taken. At the end of 90s and beginning of 2000s, the development of SAMGs was completed for all the units and summary of the strategies for each unit was available to be used in severe accident guidance.

The second period of the review starts in January 2008 when the Reactor Harmonization Working Group (RHWG) published the first version of the WENRA Reference Levels (2008), which has been subsequently transposed into the Belgian Legislation (Royal Decree of 30 November 2011). In the framework of the action plan elaborated following the requirements of WENRA Reference Levels (and the Royal Decree), RL F.4.7 requiring that "containment degradation by molten fuel shall be prevented or mitigated as far as reasonably practicable" (see Ref. [2]) was added to the discussions between the utility and the Belgian regulatory body. The related assessment by the regulatory body led to request re-assessing the strategy of reactor cavity flooding, accounting also for ex-vessel steam explosion for some plants relying on early reactor cavity flooding. After Fukushima Daiichi accident, the stress test has been performed for all Belgian units, leading to two additional actions regarding the ex-vessel corium coolability strategies. This included:

- revisiting the available R&D (e.g. CCI-7 and CCI-8, and SERENA programme) about ex-reactor vessel cooling and basemat melt-through;
- the feasibility study for the installation of additional means for top flooding for some units (modifications needed to guarantee a dry cavity before RPV failure, and alternative injection means in the reactor cavity after RPV failure).

The status of the strategies is hence based on the reactor cavity concrete type (siliceous or limestone/common sand (LCS) concrete) and the possibility to ensure dry cavity before RPV failure. Therefore, early cavity flooding strategy is considered for siliceous concrete reactor cavity for which flooding can be ensured, while late reactor cavity flooding is considered LCS concrete reactor cavity for which dry cavity can be ensured if RPV does not fail.

(b) Strategy for the Corium Stabilisation in Case of a Severe Accident for the French PWRs, F. Fichot., IRSN, France

This presentation provided an overview of corium stabilization strategies for the French PWRs in the context of the programme proposed by Electricité de France (EDF) for long term operation (LTO) of its nuclear power plants. In the context of French Periodic Safety Reviews (PSR) including the recent 3rd PSR of PWR1300MW(e), the on-going 4th PSR of PWR 900MW(e) and the impact of Post-Fukushima Complementary Safety Evaluation (CSE), the assessment of the measures for the prevention and the mitigation of severe accidents included a specific focus on the risks related to basemat penetration. One of the French Nuclear Safety Authority (ASN) Orders dated on 21 January 2014 requires "…the setting up (by EDF) of a system…. To prevent basemat melt-through by the corium…" (see Ref. [3]).

Two strategies have been assessed by EDF for its NPPs, namely:

- the reactor cavity is filled with water and can help for IVMR; in case the RPV fails, the corium is poured in the water o the reactor cavity;
- the reactor cavity remains dry until he RPV failure and water is injected in the reactor cavity after corium spreading.

The first strategy was not chosen by EDF because the demonstration that the RPV will keep its integrity in the worst cases is difficult, for high power; in addition, in case of RPV failure, a consequent steam explosion in the already flooded reactor cavity, with challenging effects on containment integrity, cannot be excluded. Hence, based on R&D results and plant design, the second strategy has been selected by EDF with the following main principles:

- dry reactor cavity before corium spreading;
- dry additional corium spreading zone in some cases where dry cavity area is not enough;
- containment sump filled with water during early phase;
- corium spreading in the reactor cavity and additional area if any;
- gravity flooding above the corium after its spreading using the water of the sumps.

The solution is under discussion between EDF and ASN for the French reactors concerned by the LTO. It requires equipment modification to maintain initial dry reactor cavity, additional spreading room and basemat thickness increase with high gas content concrete (depending on the containment design in terms of geometry and concrete composition), and passive top flooding with sump water. Remaining issues are related to uncertainties on corium pouring conditions and effect of metal on cooling mechanisms.

The adaptation of the strategy to the specific case of Fessenheim 900MW(e) plants has been implemented as follows:

- reactor cavity thickened by self-sustained LCS-type concrete layer;
- fusible plug opens a transfer channel to a surrounding additional zone which is also thickened;
- possible spreading in this additional zone;
- passive early flooding of the spread corium using sump water.

The operationality of the modification procedure was demonstrated for two plants.

To address remaining issues related to MCCI under water and to improve knowledge on corium stabilization by top flooding during early MCCI phases, EDF launched, within an international partnership, an important and costly experimental programme carried out by ANL.

Some pending issues needing further R&D to support their resolution concern: (i) the initial ex-vessel conditions, in particular localized faster ablation due to localized corium pouring and accumulation and the stability of this accumulation; (ii) MCCI and corium coolability as highlighted in the OECD/NEA report on State-of-the-Art Report on Molten-Corium-Concrete interaction and Ex-Vessel Molten-Core Coolability; (iii) early containment failure risk due to steam explosion in deep water pool or shallow water layer, and consequences on MCCI of corium fragmentation process or corium accumulation.

(c) Technical Evolution of the EVR Melt Stabilization Concept of the EPR, G. Urzua Torres, AREVA GmbH, Germany

The speaker discussed the technical evolution of the ex-vessel melt stabilisation concept of the EPR™ whose design targets were to also address severe accidents with core melting. This evolution process has been supported by many experimental programmes performed in the frame of national and international R&D projects and by detailed discussions with the related researchers and experts.

Preserving containment integrity throughout the severe accident requires a balanced approach which adequately addresses all related challenges. For stabilizing the molten core, the first concept considered was naturally IVMR. The main question related to IVMR was: can sufficient margins be established considering all related uncertainties? At the time of its consideration, this question could not be answered positively. Consequently, RPV failure under outside flooded conditions and, in consequence, energetic steam explosions would have to be addressed in licensing, which appeared problematic as they represent an ongoing R&D issue. Therefore, IVMR was abandoned in favour of EVCC. In addition, the general design requirement to keep the space below the RPV dry before and during melt release was established.

The next Line of Defence for stabilizing the melt is a crucible-below- the-RPV. Though a corresponding conceptual design was developed, and a basic validation provided, the "crucible-below-the-RPV" concept was – in the end - not pursued for the EPR™, because of the close distance between RPV and reactor cavity bottom and the expected high loads (impact and melt jet) during RPV failure in case of elevated internal pressure.

To avoid these loads, a spatial separation was achieved by placing the core catcher into a separate, large compartment lateral and below of the reactor cavity. As a detrimental effect, this separation required additional provisions to ensure adequate melt spreading, including a dry surface to spread on. Another positive effect is that melt spreading increases the surface-to-volume ratio, which results in lower heat fluxes and faster quenching and freezing after flooding.

The spread melt can in principle be stabilized by applying various cooling means. The simplest one is top flooding during MCCI. Different from the narrow EPR™-reactor cavity, the spreading compartment provides enough surface area to easily fulfil the EPRI-criterion of 0.02 m^2/MW_{th} which was widely followed at that time. During the EPR conceptual design the related experimental projects, (i.e. MACE at ANL, U.S.A.) did, however, not demonstrate the efficiency of self-fragmentation during MCCI. Hence, the integrity of the concrete basemat after flooding could not be proven relying on this mechanism alone. Therefore, two other concepts were also followed. The first was the COMET-concept. Its function is based on the passive injection of water into the melt from below at a self-adjusting rate, driven by the hydrostatic pressure of a water column in an adjacent reservoir. This injection mode enhances water ingress by providing an additional, more effective flow path for the water to enter the melt. While the water progresses towards the top of the melt, decay power is removed via water vaporization and steam release, causing fast fragmentation and effective volumetric cooling even for deeper melt pools. However, the achieved fast quenching also results in high rates of steam and hydrogen production and, in consequence, in high values of pressure and temperature in the containment. Therefore, the concept was finally not applied in the EPR™.

The second concept followed was the stabilization of the spread melt by top flooding on a melt-resistant, protective bottom layer. This variant has been extensively studied by dedicated experiments ranging from laboratory to industrial scale. The observed main problem is related to the thermochemical stability of the material. Under certain conditions, chemical dissolution and mechanical abrasion take place at the interface in contact with liquid melt. With the purpose of limiting and ultimately stopping this process, the original concept was modified, first by adding a sacrificial steel layer and then a passive bottom cooling. However, due to the reached complexity level, the protective layer was eliminated, resulting in the current design of the EPR™ core melt stabilization system (CMSS). The CMSS function is based on a two-phase approach. During the 1st phase the melt is accumulated and conditioned in the reactor cavity by the addition of sacrificial material. Then, in the 2nd phase, all collected melt spreads into the core catcher in a single pour, where it is cooled by passive water overflow from the IRWST. After this, the melt is surrounded by water, so all decay power in the melt is removed by water heat-up and boiling.

(d) R&D Activities to Resolve ExVC Strategy for VVER-1000 Reactor, J. Duspiva, UJV Rez, Czech Republic

This presentation provided an overview of the R&D activities to resolve EVCC strategy for WWER-1000 reactors which complements his presentation summarized in section 2.5.1(b) of this publication. The main goal of the solution of both IVMR and EVC strategies is to maintain the containment integrity for the whole course of the severe accident, or at least to significantly extend time to the loss of containment integrity to enable perform immediate emergency response activities.

Activities related to the resolution of the question of the applicability of the EVCC strategy to WWER-1000/320 (Temelin NPP) consist of many steps, some of them are finished and closed, some of them are on-going and remaining are on the list as they must be solved consequently. The initial idea on the solution of the corium issue appeared in middle of 90'last century with expectation that corium spreading from cavity to neighbouring room with top cooling will be sufficient to terminate MCCI or at least significantly extend time to containment basemat melt-through. As the knowledge about the processes and phenomena at that time was insufficient, the UJV actively participated in the OECD MCCI and subsequent MCCI2 project, and in the additional test CCI7 that was specific for the French NPPs, but with conditions very similar to the Temelin NPP, concerning siliceous concrete type.

Extensive UJV activities were focused on the code validation against OECD CCI tests using the ASTEC/MEDICIS and CORQUENCH codes. The code MELCOR/CORCON was validated on code to code comparison with those two codes mentioned above on plant conditions. Direct validation of MELCOR/CORCON was impossible due to cavity geometry assumptions. The codes were then extensively used for the evaluation of the MCCI under various conditions including spreading and spreading with top cooling. The analyses confirmed the conclusion that it is impossible to successfully arrest MCCI in case of siliceous concrete, consistently with the experimental conclusions. Additional solution had to be investigated.

The latest solution of the corium localization after WWER-1000 lower head failure is based on the idea to apply a refractory material as a liner with the function of temporary protective layer against MCCI in the reactor cavity and the additional spreading area. This liner would have several positive features such as (i) easier and faster spreading of corium; (ii) absence of steam release into corium and hence elimination of the exothermic oxidation of remaining zirconium in corium; and (iii) more efficient top cooling which could result in corium solidification or at least in significant reduction of corium concrete interaction intensity resulting in significant extension of tie to base mate melt-through. This extension of core basemat is very important because the design of the WWER-1000 containment includes three levels of non-hermetic room below the basemat. The critical step is the selection of the refractory material whose installation needs to have any impact on the reactor operation (e.g. on the venting system, thermal insulation and biological shielding in the reactor cavity), nor on activities during outage. The solution in the reactor cavity is strongly influenced by high radiation. The experimental programme on the investigation of the appropriate material is under preparation. Selection of candidates (e.g. ZrO_2 and MgO) is on-going and experimental testing is expected in 2017. Assessment is also ongoing to identify the needs for new equipment necessary to implement the design modifications of the reactor cavity for much easier corium spreading from the reactor cavity to spreading room.

The final decision on the choice of the strategy must be taken at the end of 2017; therefore, it is very important to perform as many studies as possible to support a sound decision making. The international collaboration and experience exchange are therefore very important.

(e) Corium Cooling Challenges for Ukrainian NPPs, D. Gumenyuk, SNRCU, Ukraine

The presentation from the State Scientific and Technical Center for Nuclear and Radiation Safety, Kyiv, Ukraine discussed corium cooling challenges for the Ukrainian NPPs in case of severe accident with core melting. There are currently 15 power units operated at four NPP sites in Ukraine with a total installed electrical power of 13,835 MW, which constitutes approx. 50% of the total installed electrical power of all power plants in Ukraine.

There are three types of WWER units operating in Ukraine: WWER-440/213 (Rivne NPP Units 1 and 2), WWER-1000/320 (Zaporizhzhya NPP Units 1-6, South-Ukraine NPP Unit 3, Khmelnitsky NPP Units 1, 2 and Rivne NPP Units 3, 4) and WWER-1000/302 (South-Ukraine NPP Units 1, 2).

After the accident at Fukushima Daiichi NPP (Japan), the Ukrainian regulatory body (SNRIU) Board approved an Action Plan for a targeted safety reassessment and further safety improvement of the Ukrainian NPPs in the light of the Fukushima Daiichi accident and an Action Plan for a targeted safety reassessment. One of the actions defined in the Action Plan was a targeted safety reassessment of operating nuclear facilities at NPP sites (stress tests). Based on stress-test results a package of measures has been developed aiming at severe accident management. In this respect, containment filtered venting is being installed at South Ukraine NPP units 1 and 2, and at Zaporojie NPP units 1 and 2, and passive autocatalytic recombiners (PARs) to mitigate hydrogen risk have been or will be installed at the Ukrainian NPPs. SAMGs were developed and completed for all the plants at power operation and are being completed for shutdown modes. Results of analytical justifications conducted under SAMG development showed that current design of WWER-440/213 and WWER-1000 is very vulnerable to reactor vessel failure and melt corium spreading.

For basic WWER-440/213 design, ejected corium enters to A004/V1 hermetic compartment which is separated from the non-hermetic compartment A0065 with two metal doors (thickness ~150 mm each). No corium cooling is possible in the reactor cavity due to very small square of A004/V1 floor (12.8 m^2). Therefore, the most probable end state of ex-vessel phases of severe accident of WWER-440/213 is containment failure due to hermetic door melting. IVMR appears the best solution for prevention of WWER-440/213 containment failure at ex-vessel phase of severe accidents. Some countries operating WWER-440/213 already implemented this measure (e.g. Czech Republic, Finland, Hungary, Slovak Republic) already implemented this strategy. Ukrainian Utility started to perform analytical justifications for Rivne-1, 2 NPPs. Preliminary results of these justifications were reviewed by SSTC NRS. The main questions were related to verification and validation of used codes and models and scope of analyses.

WWER-1000/320 NPP units have the structural disadvantage concerning the location of the 54 channels for movement of ionization chambers of the RCP system. These chambers are located inside concrete wall around reactor cavity compartment (GA301) and connected with non-hermetic compartment A336.

The only scenario for an early degradation of the containment and large early radioactive releases is reached when the RPV bottom is melted and the molten core penetrates the reactor shaft. Further, as a result of the MCCI, a breakthrough of the reactor cavity wall to the locations for movement of ionization chambers, where the molten core is discharged outside the containment. Therefore, the ex-vessel cooling of spreading corium is the main possibility to prevent containment failure for existing WWER-1000/320 design. In the framework of Comprehensive Safety Improvement Programme for the Ukrainian NPPs, two main actions are foreseen to overcome the disadvantage linked to the location of the 54 channels for movement of ionization chambers of the RCP system; they consist of ionization chambers' sealing and investigations of possibilities for EVCC. Concerning the latter, preliminary investigations showed many critical aspects such as used code and model validation, corium viscosity, concrete properties. For resolving these issues, the special investigation programme entitled: Programme of Severe Accident Phenomena Investigation has been launched and is being conducted. The use of coupled MELCOR-LAVA for modelling corium spreading and coupling and the evaluation of original properties of each NPP concrete are foreseen.

(f) Analysis of Strategies for Corium Cooling and Retention for a BWR Mark II, T. Garcia, CNSNS, Mexico

The presenter discussed the analysis of strategies for corium cooling and retention for Laguna Verde NPP (BWR Mark II). IVMR and EVCC strategies are the main challenges to minimize releases of radioactive material to environment; in those strategies the priority is ultimately to keep the integrity of the containment.

This work presents different strategies analyzed of reflooding for corium cooling (in vessel) at different level of core damage in the case of an SBO with the RCIC (Reactor Core Isolation Cooling) available for 4 hours. Strategies were analyzed for different values on timing and total flow value to flooding the damaged core. Those accident scenarios were simulated using the fully integrated, engineering-level computer code MELCOR modeling the progression of accident phenomena in the light water reactor model of a Boiling Water Reactor 5 - Mark II containment developed in the CNSNS, México.

Two illustrative scenarios have been considered based on an SBO as initiating event. They are different in terms of power supply/ batteries and high and low-pressure system recovery, and the possibility of reflooding the core after a certain time. In the case of injection of water 4 hours after core uncover starts (which corresponds to 25-30% of damaged core), the results showed that RPV failure is delayed by 10 hours compared to the case with no core reflooding.

The RPV failure is initiated by penetration failure whose time is function of the quantity and temperature of material in contact with the walls as well as melted structures coolability.

This analytical work is expected to be pursued and will support future evaluation the SAMG of Laguna Verde NPP and analysis of the hardened containment venting system.

2.8.2. Summary of the session

The EVCC implementation or potential application provided in this Technical Session underscored the R&D experimental and analytical effort in the area that served as technical basis for decision making for severe accident management planning; they also referred to specific reactor design and layout, concrete type as well as to the applicable safety regulations.

While for the IVMR strategy, both new reactors and operating reactors must fulfil the same safety objective (i.e. retention of the corium in the reactor vessel), the EVCC strategies are currently subject to different criteria concerning structural integrity of containment vessels.

It is recognized that the implementation of EVCC back-fitting measures could be more complicated for operating reactors, due to already existing design and layout and radiation protection issues induced by possible modifications.

Integrated solutions were described for operating reactors back-fitting. They depend on e.g. reactor design, containment layout, accident scenario, water presence in the reactor cavity at lower head failure, type of concrete, and possibility of corium spreading. For example, concrete likely to be reached by corium could be thickened with an additional layer to delay MCCI, and/or achieve debris coolability.

2.9. GENERAL DISCUSSION ON EX-VESSEL CORIUM COOLING

The TM participants noted that during the last three decades, there have been many tests (both integral and separate effect) completed and significant progress has been made towards understanding phenomenology related to MCCI and corium coolability. The following conclusions were discussed with, and agreed by the meeting participants:

Lessons learned from these tests and associated analyses are being applied in severe accident management planning for operating plants and in the design of new plants.

With respect to modelling, dedicated MCCI computer codes can currently analyse idealized cases where corium is spread uniformly over a dry reactor pit. A few physics-based models for debris cooling mechanisms and debris beds have been developed and implemented in system codes to support accident management planning and plant safety analyses.

However, there is a lack of data to support analysis of long duration transients. Indeed, tests are typically limited to a few hours while Fukushima Daiichi accident indicated that plant accident can last for several days.

There is a need to perform more realistic plant simulations. Most codes are unable to address non-uniform core debris distributions and/or particle beds as initial conditions in real plant configurations. Also, realistic containment features such as deep sumps, sump drains, and cable penetrations need to be considered in analysis and testing activities.

There are data gaps in the experiment database related to EVCC concerning: (i) high metal content in the melt, (ii) rebar in the concrete, (iii) non-uniform melt accumulations, crust formation/failure mechanisms and their effect on MCCI, and (iv) the effect of raw water on coolability.

Depending on reactor design and accident scenario, debris coolability might not be ensured for all situations. Therefore, engineered features could be needed to ensure coolability in some plants.

Recriticality in debris beds formed for MOX fuel needs to be considered.

2.10. INTERNATIONAL COOPERATION

International cooperation was discussed at the Discussion Sessions, and the TM participants summarized the status of international cooperation on IVMR and suggested possible activities as follows:

- Several national and regional R&D programmes both on IVMR and EVCC are ongoing or planned to start soon;
- Information exchange in existing bilateral and multilateral cooperation is limited among contracting parties, and it is difficult to disseminate the information beyond them;
- Code benchmarking against well-defined experiments, including blind test calculations, will be useful for code/model development and validation;
- In the case of IVMR, RV integrity including characterization of material properties at severe accident conditions and improvement of the mechanical modelling could be an interesting topic for cooperation;
- A R&D activity on top cooling could be interesting for both IVMR and EVCC; and
- Education and training of young nuclear professionals is a key element of future planned activities to ensure knowledge transfer in the area of severe accidents.

The value of IAEA Technical Meetings and Workshops was well recognized in fostering the exchange of information on the latest advances and challenges in the addressed topic and its dissemination, and in contributing to building consensus on a common approach. IAEA means such as Technical Meetings, Workshops, Coordinated Research Projects (CRPs), International Collaborative Standard Problem (ICSP) for code benchmarking or other activities leading to the preparation of TECDOCs or Safety Reports are open to all interested Member States and need to be encouraged as appropriate.

3. CONCLUSIONS AND RECOMMENDATIONS

3.1. CONCLUSIONS

A lot of R&D activities have been carried out and are still on-going to develop IVMR strategy and technologies at national, regional and international level, and the Fukushima Daiichi accident revitalized it. Most of the efforts have been made in understanding of key phenomena, both inside and outside of the RV, with experiments and numerical analyses, code improvement/validation, and application of IVMR strategy to specific reactors and its optimization.

In a similar way and during the last three decades, there have been many tests (both integral and separate effect) and analyses completed; significant progress has been achieved towards understanding phenomenology related to EVCC (MCCI and corium coolability), as well as code development and validation.

The meeting participants have agreed that international scientific collaboration remains essential in having better and common understanding of IVMR and EVCC phenomenology and technologies, and hence in increasing the safety level of operating and new nuclear power plants. In fact, the research on IVMR and EVCC requires significant human and financial resources whose use has to be optimized and shared to the extent possible.

3.1.1. IN-VESSEL MELT RETENTION

During the past years a significant progress has been achieved in understanding and modelling the behaviour of molten pools in the RV, and new tests (e.g. IVMR project of the EU H2020 programme) are expected to provide data for the validation of computer codes allowing them to be used at reactor-scale conditions.

Main factors affecting the maximum heat flux that can be removed by external water flow (i.e. CHF) include: stability of the natural circulation; outer surface conditions of RV lower head; geometry of the flow path; and water sub cooling at the inlet of the flow path. Recent R&D results suggest that the most effective measures to increase CHF might be optimization of the flow path and outer surface conditions (i.e. porosity).

The application of IVMR strategy requires design considerations such as: depressurization of the reactor coolant system (RCS); water source for ERVC and/or in-vessel flooding; initial flooding, followed by a long-term water supply; venting and condensation of generated steam; and management of potential negative impacts if the IVMR with external cooling fails (e.g. hydrogen risk, threat of steam explosion in the containment). Various active and passive systems have been designed. It is agreed among the TM participants that the probability of success of IVMR is generally higher in lower power reactors, due to the sufficient margins between the heat flux generated by the corium in the lower head and the CHF. However, it also highly depends on the specific designs (e.g. designs with additional steel mass to avoid focusing effect, or those where corium relocation in the lower plenum is delayed leading to low residual power relocated corium).

There are still uncertainties in phenomenology inside the RV, and they mainly come from insufficient knowledge in key phenomena related to accident progression and limitations of experimental facilities and instrumentation.

Regarding phenomenology inside the RV, behaviour of stratified molten pools is still a key issue and requires additional information in terms of experimental data and material properties. Especially, transient behaviour of molten pools is important to determine local heat flux values which might result in larger threat to the integrity of the RV than the fully-developed (steady) state. Larger scale corium pool tests are desirable to provide more realistic data.

Phenomenology outside the RV is much clearer than that inside the RV: a lot of CHF data is available now. However, some of it is contradictory (e.g. the effect of surface oxidation), and experimental results need to be sorted and presented in a consistent way to be used easily by analysts, designers and regulators. Full height experimental facilities to measure CHF are necessary for validation data, and they are to be designed as closely as possible to the real conditions. Two new large experimental facilities are designed to measure CHF at the outer surface of the RV lower head under more realistic configurations and flow conditions.

Code improvement and validation is important to simulate transient behaviors of molten pools inside the RV, water flow outside the RV and their interaction, which will give thermal-hydraulic assessment of the IVMR strategy under the given conditions. At present, different models and codes produce quite different results partly due to the absence of a proper validation matrix for IVMR-related phenomena. The current models for focusing effect and transient behaviour of molten pools need to be improved.

Concerning application of IVMR strategy and technologies to reactor designs, there is a consensus among TM participants that there are not enough analyses of structural integrity made for different shapes of RVs and different pressure and temperature conditions. Hence, the structural integrity analysis is an important part of the IVMR strategy assessment that must complement the thermal-hydraulic assessment. Structural integrity needs to be evaluated based on detailed phenomenology and material properties at realistic severe accident conditions. Probabilistic approaches or assumptions are considered necessary as complement for deterministic approach in the analysis of IVMR strategy effectiveness.

3.1.2. EX-VESSEL CORIUM COOLING

It is recognized that EVCC strategy, combined with other measures, remains the ultimate means to limit molten corium-concrete interaction (MCCI) in case IVMR is not applied or fails. Several NPPs under operation or under construction apply or aim to apply EVCC strategy.

Ex-vessel progression of severe accidents depends on the RCS pressure at the moment of RV failure, the possibility of containment over-pressurization, presence of water in the reactor cavity, corium cooling (e.g. by flooding), thickness of the concrete, and accident management measures to address non-condensable gases generation. There can be several paths which end by early containment failure, delayed containment failure or a manageable situation. An ex-vessel manageable situation is a situation in which the melt/debris has been cooled and quenched and kept in this state for a long time. Since early or delayed containment failure can lead to early releases or large releases, it is assumed that those situations are excluded. Hence, the focus is on the phenomena related to EVCC to stop MCCI and reach a manageable situation after a significant amount of the in-vessel corium inventory would have poured by gravity in the reactor cavity.

When presence of water in the reactor cavity is allowed, the following melt fuel-coolant interaction can occur: (i) if an energetic steam explosion occurs outside the RV, the melt involved in the explosion might be dispersed out of the reactor cavity while the one not directly involved in the explosion can still form a particle debris bed; (ii) otherwise, the melt jet can break up in the water pool and a portion of the initial melt could form a particle debris bed and the remaining part might form a cake.

In case of dry reactor cavity conditions, the poured debris accumulates at the bottom, transfers heat to the atmosphere by radiation and convection and ablates the concrete substrate, possibly leading to containment melt-through if no measures are taken to stop MCCI. Reactor cavity flooding and/or corium spreading to reduce the heat flux are necessary to reach a manageable situation. Melt coolability could be improved by top flooding.

Lessons learned from three decades of tests and associated analyses have been or are being applied in severe accident management planning for operating plants and in the design of new plants. They highlighted the following:

- For dry cavity conditions, melt temperature evolution and concrete ablation shape are influenced by concrete composition (radial ablation rate is faster than the axial ablation rate for siliceous concrete while a more isotropic ablation is observed for limestone-rich concrete). In a few tests, ablation appears to be more pronounced and faster oxidation kinetics in the areas where metal is in direct contact with concrete are observed.
- Under wet cavity conditions, several corium cooling mechanisms (e.g. bulk cooling, water ingression, melt ejection and crust breaching) were identified, and potential enhancement of coolability was observed.

For current plant applications, MCCI phenomena are analysed based on conservative assumptions with respect to the weakness of the containment design. In general, extrapolating the results of experiment results at plant scale requires some idealization of plant geometry and configuration; in fact, dedicated MCCI computer codes can currently analyse idealized cases where corium is spread uniformly over a dry reactor cavity. In addition, empirical correlations are needed as ablation anisotropy cannot be explained by existing phenomenological code models.

A few physics-based models for debris cooling mechanisms and debris beds have been developed and implemented in system codes to support accident management planning and plant safety analyses. However, there are limitations in addressing local corium accumulations in the case of an initially flooded reactor cavity, and very few codes model the impact of top flooding, due to the lack of data for the top flooding of metal-oxide melt.

Despite the significant progress mentioned above, the following challenges and open issues are still to be addressed:

- the lack of data to support analysis of long duration transients. Indeed, tests are typically limited to a few hours while Fukushima Daiichi accident indicated that plant accident conditions can last for several days before reaching a controlled state;
- the need to perform more realistic plant simulation. Most codes are unable to address non-uniform core debris distributions and/or particle beds as initial conditions in real plant configurations. Also, realistic containment features such as deep sumps, sump drains, and cable penetrations need to be considered in analysis and testing activities;
- Data gaps in the experiment database related to EVCC concerning: (i) high metal content in the melt, (ii) rebar in the concrete, (iii) non-uniform melt accumulations, crust formation/failure mechanisms and their effect on MCCI, and (iv) the effect of raw water on coolability;
- Depending on reactor design and accident scenario, debris coolability might not be ensured for all situations. Therefore, additional engineered features might be needed to ensure coolability in some plants;
- Recriticality in debris beds formed for MOX fuel needs to be considered.

The presentations and subsequent discussions highlighted that the implementation of EVCC back-fitting measures could be more complicated for operating reactors, due to already existing design and layout and radiation protection issues induced by possible modifications.

3.1.3. International collaboration

Several national and regional R&D programmes both on IVMR and EVCC are ongoing or planned to start soon. However, information exchange in existing bilateral and multilateral cooperation is limited among contracting parties for confidentiality reasons and funds involved, and it is difficult to disseminate the information beyond them.

The following activities could be carried out, as a complement, in the frame of international cooperation:

- Code benchmarking against well-defined experiments, including blind test calculations, will be useful for code/model development and validation;
- In the case of IVMR, RV integrity including characterization of material properties at severe accident conditions and improvement of the mechanical modelling could be an interesting topic for cooperation;
- A R&D activity on top cooling could be interesting for both IVMR and EVCC; and
- Education and training of young nuclear professionals is a key element of future planned activities to ensure knowledge transfer in the area of severe accidents from senior to younger generations.

3.2. RECOMMENDATIONS

During the TM, the following practical suggestions were made by the meeting participants for possible common efforts:

- o Establishment of a PIRT to identify key IVMR phenomena;
- o Analysis codes benchmarking and validation;
- o Molten pool behaviour experiments, including transient and 3-layer behaviours, with higher fidelity and representativeness;
- o Large scale experiments with shared test matrices; and
- o Steam explosion phenomenology.

For follow-up activities, it is recommended to focus on the first four suggestions which can be grouped in one structured activity in which the IAEA could play a major role by providing an appropriate platform for information exchange, discussion and building consensus on a technically sound approach to IVMR analysis. This activity could start with a PIRT to identify key IVMR phenomena and corresponding suitable tests and test facilities of different scales to validate analysis computer codes in the frame of code benchmarking against separate effect tests. To complete this code benchmarking, an analytical “benchmark” case could be defined, simulating an accident transient in a specific reactor design for which evaluations are already available. In the analytical benchmark case, all the models, previously validated, would be activated at the same time.

ABBREVIATIONS

ANPP: Armenian Nuclear Power Plant
CFD: Computational Fluid Dynamics
CHF: Critical Heat Flux
CMSS: Core Melt Stabilization System
CSE: Complementary Safety Review
CPRR: Containment Protection and Release Reduction
CV: Calandria Vessel
DET: Decomposition Event Tree
ERVC: External Reactor Vessel Cooling
ES: End Shield
EVCC: Ex-Vessel Corium Cooling
FCI: Fuel Coolant Interaction
FEM: Finite Element Method
ICI: In-Core Instrumentation
IEM: International Expert Meeting
IRWST: In-Containment Refuelling Water Storage Tank
IVMR: In-Vessel Melt Retention
LBLOCA: Large Break Loss of Coolant Accident
LCS: Limestone/Common Sand
LES: Large Eddy Simulation
LOCA: Loss of Coolant Accident
LSP: Lower Support Plate
LTO: Long Term Operation
LWR: Light Water Reactor
LWRS: Light Water Reactor Sustainability
MCCI: Molten Corium Concrete Interaction
Nu: Nusselt Number
PAR: Passive Autocatalytic Recombiner
PHWR: Pressurized Heavy Water Reactor
PIRT: Phenomena Identification and Ranking Table
PSR: Periodic Safety Reassessment
Ra: Rayleigh Number
RANS: Reynolds Averaged Navier-Stockes
RCIC: Reactor Core Isolation Cooling
RHWG: Reactor Harmonization Working Group
RPV: Reactor Pressure Vessel
RSM: Reynolds Stress Equation Model
RV: Reactor Vessel
SA: Severe Accident
SAM: Severe Accident Management
SAMG: Severe Accident Management Guidelines
SAWM: Severe Accident Water Management
SBLB: Subscale Boundary Layer Boiling
SBO: Station Blackout
SOAR: State-of-the-Art-Report
WCR: Water Cooled Reactor
WP: Work Package
WWER: Water Cooled Water Moderated Power Reactor (sometimes noted VVER)

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ANNEX I. MEETING PROGRAMME

INTERNATIONAL ATOMIC ENERGY AGENCY

Technical Meeting on
Phenomenology and Technologies Relevant to
In-Vessel Melt Retention and Ex-Vessel Corium Cooling

17–21 October 2016
Lecture Hall, D Building, SNERDI
Shanghai, China

AGENDA
Monday, 17 October 2016

TOPIC	Time	Presenter
Registration Start	09:00	All
Opening Session		
Welcome Remarks by Host Organization	09:30	M. Zheng, President of SNERDI, China
Logistics Information	09:40	T. Liu, SNERDI, China
Opening Remarks by IAEA	09:50	K. Yamada, Scientific Secretary, IAEA
Self-Introduction of Participants	10:00	All
Goals of Technical Meeting	10:30	K. Yamada, IAEA
Adoption of Agenda	10:50	All
GROUP PHOTO	11:00	All
BREAK	11:10	
Technical Session-1A: General Considerations on IVMR Strategy Session Chair: J. Q. Yan, SNERDI, China (30 minutes/presentation including 5-minute Q&A)		
Summary and Conclusions from the International Seminar on In-Vessel Retention Strategy	11:40	F. Fichot, IRSN, France
Modeling and Analysis of In-Vessel Melt Retention and Ex-Vessel Corium Cooling in the U.S.	12:10	E. Fuller, USNRC, USA
LUNCH BREAK	12:40	
Application of IVMR-ERVC Strategy in Korea	14:00	K.H. Lim, KINS, Korea
Corium Retention Strategy on VVER under Severe Accident Conditions	14:30	Y. Zvonarev, Kurchatov Institute, Russian Federation

In-Vessel Melt Retention at Loviisa NPP: Past and Present Activities	15:00	T. Laato, Fortum, Finland
BREAK	15:30	
R&D Activities for the Evaluation of In-Vessel Corium Cooling Performance at KAERI	16:00	H.Y. Kim, KAERI, Korea
In-Vessel Retention (IVMR) Design Features, Phenomenology and Technical Basis in the CANDU Reactor	16:30	R. McLean, COG, Canada
Discussion on IVMR Strategy	17:00	All
Adjourn 1st day	17:30	
Social Event	18:00	Hosted by SNERDI

Tuesday, 18 October 2016

TOPIC	**Time**	**Presenter**
Technical Session-1B: External Reactor Vessel Cooling Session Chair: N. Kozlova, SEC NRS, Russian Federation (30 minutes/presentation including 5-minute Q&A)		
Introduction of CAP1400 IVMR Experiments	09:00	K. Zhang, SNERDI, China
Overview on the IVMR Strategy	09:30	J. Zdarek, ÚJV Řež, Czech Republic
Discussion on External Reactor Vessel Cooling	10:00	
BREAK	10:30	
Technical Session-1C: Molten Pool Behaviours and Structural Integrity of Reactor Vessel Session Chair: A. Miassoedov, KIT, Germany (30 minutes/presentation including 5-minute Q&A)		
Heat Transfer in Homogeneous and Stratified Melt Pools in the Lower Head of a Reactor Pressure Vessel	11:00	A. Miassoedov, KIT, Germany
Research on Thermodynamic Interaction of Corium Material in Lower Head	11:30	P. Gu, SNERDI, China
Corium Propagation Modelling with the PROCOR Software – Application to Corium Vessel Lower Head	12:00	L. Saas, CEA, France
LUNCH BREAK	12:30	
Experimental and Numerical Study on the Heat Transfer Characteristics of Melt Pool	14:00	Y. Zhang, XJTU, China
Use of CFD for IVMR Studies	14:30	C. Le Guennic, EDF, France
Structural Integrity Research for Reactor Pressure Vessel under In-Vessel Melt Retention	15:00	Y. Gao, SNERDI, China

BREAK	15:30	
Original Core Catcher Design in order to Manage the Core Meltdown Severe Accident with the 'In-Vessel Retention Strategy'	16:00	D. Aquaro, University of Pisa, Italy
Discussion on Molten Pool Behaviours and Structural Integrity of Reactor Vessel	16:30	All
Adjourn 2nd day	17:00	

Wednesday, 19 October 2016

TOPIC	**Time**	**Presenter**
Technical Session-1D: Application of IVMR Strategy to Specific Reactor Designs Session Chair: R. McLean, COG, Canada (30 minutes/presentation including 5-minute Q&A)		
CAP1400 IVMR related Design Features and Analysis Methodology	09:00	G. Shi, SNERDI, China
R&D Activities to Resolve IVMR Strategy for VVER-1000 Reactor	09:30	J. Duspiva, ÚJV Řež, Czech Republic
A Case Study on WWER-440 for Severe Accidents with Core Meltdown and Reactor Vessel Destruction	10:00	T. Sargsyan, Armenian NPP, Armenia
BREAK	10:30	
Introduction of NPIC Work on In-Vessel Retention (IVMR) Strategy	11:00	D. Zhu, NPIC, China
Application of IVMR in AREVA's Gen-3 Advanced BWR (KERENA)	11:30	M. Fischer, AREVA, Germany
In-Vessel Retention Analysis for Pressurised Heavy Water Reactors (PHWR) under Severe Core Damage Accident (SCDA)	12:00	O. Gokhale, BARC, India
LUNCH BREAK	12:30	
Discussion on Application of IVMR Strategy	13:30	All
Discussion Session-1: General Discussion on IVMR Session Chair: F. Fichot, IRSN, France (60 minutes)		
Discussion on IVMR - Current Status and Recent Progress of IVMR R&D - Remaining Challenges and Open Issues - Proposals for International Cooperation	14:00	All
BREAK	15:00	
Technical Session-2A: General Considerations on EVCC Strategy and Containment Integrity Session Chair: S. Bechta, KTH, Sweden (30 minutes/presentation including 5-minute Q&A)		

CEA R&D Activities related to Ex-Vessel Corium Behavior in a PWR	15:30	B. Teisseire, CEA, France
US Department of Energy Severe Accident Research in the Area of In-Vessel Core Melt Progression and Ex-Vessel Debris Coolability	16:00	M. Farmer, ANL, USA
Improvement of Fuel-Coolant Interaction Models for Ex-Vessel Debris Coolability Evaluation	16:30	T. Matsumoto, JAEA, Japan
Simulation of Molten Core Concrete Interaction for CANDU accident late phase by MEDICIS module	17:00	M. Apostol, RATEN ICN, Romania
Adjourn 3rd day	17:30	

Thursday, 20 October 2016

TOPIC	**Time**	**Presenter**
Technical Session-2A: General Considerations on EVCC Strategy and Containment Integrity [cont'd] Session Chair: S. Bechta, KTH, Sweden (30 minutes/presentation including 5-minute Q&A)		
Recent Progress in Phenomenology, Analysis, and Accident Management Strategies related to Molten Core-Concrete Interactions and Coolability	09:00	M. Farmer, ANL, USA
Discussion on EVCC Strategy and Containment Integrity	09:30	All
Technical Session-2B: Application of EVCC Strategy to Specific Reactor Designs Session Chair: J. Duspiva, ÚJV Řež, Czech Republic (30 minutes/presentation including 5-minute Q&A)		
Status and the Strategies of Ex-vessel Corium Cooling in Belgium	10:00	A. Malkhasyan, Bel V, Belgium
BREAK	10:30	
Strategy for the Corium Stabilisation in Case of a Severe Accident for the French PWRs	11:00	F. Fichot, IRSN, France
Technical Evolution of the EVR Melt Stabilization Concept for the EPR	11:30	G. Urzua Torres, AREVA, Germany
R&D Activities to Resolve ExVC Strategy for VVER-1000 Reactor	12:00	J. Duspiva, ÚJV Řež, Czech Republic
LUNCH BREAK	12:30	
Corium Cooling Challenges for Ukrainian NPPs	14:00	D. Gumenyuk, SNRIU, Ukraine
Analysis of Strategies for Corium Cooling and Retention for a BWR Mark II	14:30	T. Garcia, CNSNS, Mexico
Discussion on Application of EVCC Strategy	15:00	All
BREAK	15:30	

Discussion Session-2: General Discussion on EVCC Session Chair: M. Farmer, ANL, USA (60 minutes)		
Discussion on EVCC - Current Status and Recent Progress of EVCC R&D - Remaining Challenges and Open Issues - Proposals for International Cooperation	16:00	All
Adjourn 4th day	17:00	

Friday, 21 October 2016

TOPIC	**Time**	**Presenter**
Summary Session 1: Meeting Summary		
Summary of Technical and Discussion Sessions on IVMR	09:00	All
Summary of Technical and Discussion Sessions on EVCC	10:00	All
BREAK	11:00	
Summary Session 2: International Cooperation on IVMR&EVCC		
Summary of Proposals for International Cooperation	11:30	All
Closing Session		
Closing Remarks by IAEA	12:00	A. Amri, Scientific Secretary, IAEA
Adjourn Meeting	12:10	

ANNEX II. LIST OF PARTICIPANTS

Name	Organization	Country
Sargsyan, T.	Armenian NPP	Armenia
Malkhasyan, A.	Bel V	Belgium
McLean, R.	CANDU Owner's Group (COG) /Bruce Power	Canada
Guo, Q.	China Nuclear Power Engineering Co. (CNPEC)	China
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