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Advances in Severe Accident Simulation and Modelling for Pressurized Heavy Water Reactors

ADVANCES IN SEVERE ACCIDENT SIMULATION AND MODELLING FOR PRESSURIZED HEAVY WATER REACTORS

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IAEA-TECDOC-2077

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INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The IAEA organized a series of activities after the accident at the Fukushima Daiichi nuclear power plant to advance the understanding of severe accident phenomenology and to develop, improve and benchmark severe accident analysis codes. Although technical meetings and related publications have addressed the topic for water cooled reactors, the most recent IAEA status report for heavy water reactor severe accident codes and test facilities for validation dates from 2008.

To share knowledge and experience on the recent advancements in heavy water reactor severe accident codes, the IAEA organized the Technical Meeting on Pressurized Heavy Water Reactor Severe Accident Simulation and Modelling, which was held in September 2022. Seventy-five participants from 18 Member States registered for the meeting, which included updates and a discussion organized into four main sessions: (i) updates in licensing requirements and their implementation by the industry; (ii) code developments and validation of heavy water reactor severe accidents; (iii) experiments for heavy water reactor severe accidents; (iv) general discussions and next steps.

The meeting highlighted the progress made in understanding phenomena relevant to severe accident progression in heavy water reactors and the corresponding improvement in simulation and modelling tools. These improvements are documented in this publication. Analyses however still rely on assumptions where knowledge gaps remain, and the contributing Member States suggested further improving the simulation tools and the phenomenological understanding through experiments and cooperation.

The meeting highlighted specific areas expected to have an important impact on the accident progression and for which additional investigations are needed. These include the evaluation of the critical heat flux on the calandria vessel, corium flow through calandria vessel penetrations, pressure loads caused by the core collapse, generation of hydrogen and fission products during the event, the impact of iodine chemistry models in the containment on the release of fission products, reflux condensation in the steam generator and flow distribution in the reactor headers and among feeder pipes.

The IAEA is grateful to the experts listed at the end of this publication for their contributions to drafting and review. The IAEA officer responsible for this publication was E.-L. Pelletier of the Division of Nuclear Power.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The demand for severe accidents simulations and the verification of mitigation measures has risen in operating nuclear power plants. Severe accidents simulations have also become an integral aspect of designing advanced nuclear power plants, including pressurized heavy water reactors (PHWR). Over the past decade since the Fukushima Daiichi Nuclear Power Plant (NPP) accident , the scope and application of severe accident analysis have expanded within deterministic and probabilistic safety analysis frameworks. This expansion includes informing Level 2 Probabilistic Safety Assessment (PSA) and providing support for emergency planning. In parallel, plant simulators have enhanced their capabilities to cover the severe accident domain for training and visualization purposes.

Analysing severe accidents involves modelling complex and interrelated physical phenomena occurring throughout the different stages of accident progression. Consequently, sophisticated computer codes are crucial for predicting the reactor core and containment responses during the accident. Despite notable progress in understanding and modelling severe accident phenomena, the implementation into integral codes remains predominantly empirical. Additionally, challenging experimental conditions pose difficulties in obtaining accurate and comprehensive measurements, leading to validation data characterized by significant uncertainties.

The most recent comprehensive analysis and status report of severe accident codes and test facilities specific to PHWRs dates to 2008, as detailed in the IAEA-TECDOC-1594 "Analysis of Severe Accidents in Pressurized Heavy Water Reactors" [1]. The status and R&D gaps in modelling and simulation of severe accidents in water cooled reactors were further addressed through several IAEA activities, including a coordinated research project (CRP) that was initiated in 2007 and competed in 2012 on Benchmarking of severe accident computer codes for heavy water reactor applications documented in the IAEA-TECDOC-1727 [2]. It was determined that additional research is needed to enhance the understanding and simulation of various phases in the progression of severe accidents in PHWRs, specifically core disassembly, debris oxidation, and corium-vessel interaction.

In 2022, the IAEA organized a Technical Meeting on PHWRs severe accident simulation and modelling. The aim was to share experience and knowledge pertaining to recent initiatives in advancing, validating, and implementing codes for severe accidents in PHWRs, along with the associated research and development activities. The insights shared during the meeting are documented in this TECDOC that describes advances in PHWR severe accident and modelling over the last decade.

1.2. OBJECTIVE

The objective of this publication is to capture the recent efforts in the development, improvement, validation, and application of PHWR severe accident codes and related R&D by summarizing the information from the Technical Meeting on PHWR severe accident simulation and modelling that was held on 20–23 September 2022. The information is detailed in terms of needs and application of severe accident simulation and modelling, regulatory aspects, updates in experimental programmes and in code development, specifically for PHWRs. This publication builds on the information presented in the IAEA-TECDOC-1594 [1] and the IAEA-TECDOC-1727 [2].

1.3. SCOPE

This publication summarizes the recent updates in severe accident simulation and modelling for PHWRs based on the materials and discussions presented at the Technical Meeting on PHWR severe accident Simulation and Modelling.

The following topics, relevant for simulation and modelling of PHWR severe accidents, were discussed:

- Updates in licensing requirements;
- PHWR severe accident simulators in the most recent years;
- Experiments in past decade;
- Developments in computer codes for simulation, modelling, and validation in the past decade.

1.4. STRUCTURE

Section 2 of this publication provides background on the need for and application of severe accident simulation and modelling for the safety case, as a basis for the accident management and for emergency response, training and drills, with additional information provided in Appendix I. Section 3 provides an overview of some regulatory aspects of severe accidents simulation and modelling in PHWR Member States (MS). Section 4 reviews the updates since the last IAEA publication on the topic in experimental programmes for PHWRs. Updates in the experimental programme on limited core damage in India is given in Appendix II as a complement to Section 4 Section 5 provides some advances in integrated simulation codes in PHWR MS as well as describes the adaption of some Light Water Reactor (LWR) codes for applicability to PHWR severe accident analysis, with additional details for MAAP given in Appendix III. Section 6 presents developments in supporting models and tools such as Computational Fluid Dynamics (CFD) and Finite Element Analysis (FEA) to allow more detailed modelling. A short overview of common CFD models adopted in for hydrogen distribution and mitigation system is given in Appendix IV for completeness. Finally, Section 7 summarizes the key conclusions. The Technical Meeting presentation abstracts are provided in Annex.

2. NEED AND APPLICATION FOR SEVERE ACCIDENT SIMULATION AND MODELLING

Severe accidents consist of accidents more severe than design basis accidents and that involve significant fuel degradation, either in the reactor core or in fuel storage. This section describes the need for and application of severe accident simulation and modelling in the safety case, in support of accident management and for emergency response, training and drills. Some background and definitions are presented first.

2.1. BACKGROUND AND DEFINITIONS

The IAEA-TECDOC-1594 [1], dated from 2008, described the important phenomena and accident progression for accidents that could result in damage to the reactor core in PHWRs. The reference, specific to PHWRs, was needed since the phenomena and accident progression in PHWRs differs significantly compared to LWRs, at least in the early phases.

Such accidents were classified as follows [1]:

- Limited Core Damage Accidents (LCDAs): events for which core geometry is preserved and can involve single channels events or entire core events. In those entire core events, the moderator acts as a secondary heat sink to prevent failure of the fuel channels and core degradation. A portion of LCDAs was considered part of the design basis for PHWRs;
- Severe Core Damage Accidents (SCDAs): events for which the core geometry is lost. They correspond to multiple fuel channel failures resulting in significant core damage, which is possible only if the secondary heat sink (moderator) is lost. It was considered that SCDAs in PHWRs were equivalent to the severe accident definition.

The 2018 edition of the IAEA Glossary introduced the concept of Design Extension Conditions (DECs) defined as “Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits” [3]. Design Extension Conditions include events without significant fuel degradation (DEC-A) and events with core melting (DEC-B). The definition of severe accident (“Accident more severe than a design basis accident and involving significant core degradation”) remain unchanged.

The LCDAs and SCDAs definitions almost naturally align with the DECs without significant fuel degradation (DEC-As) and with core melting (or with significant core damage, DEC-Bs). Figure 1 shows the alignment between LCDAs SCDAs and DECs. The usage of those terms however varies between the different MS (see Section 3, Regulatory Updates in heavy water reactor countries).

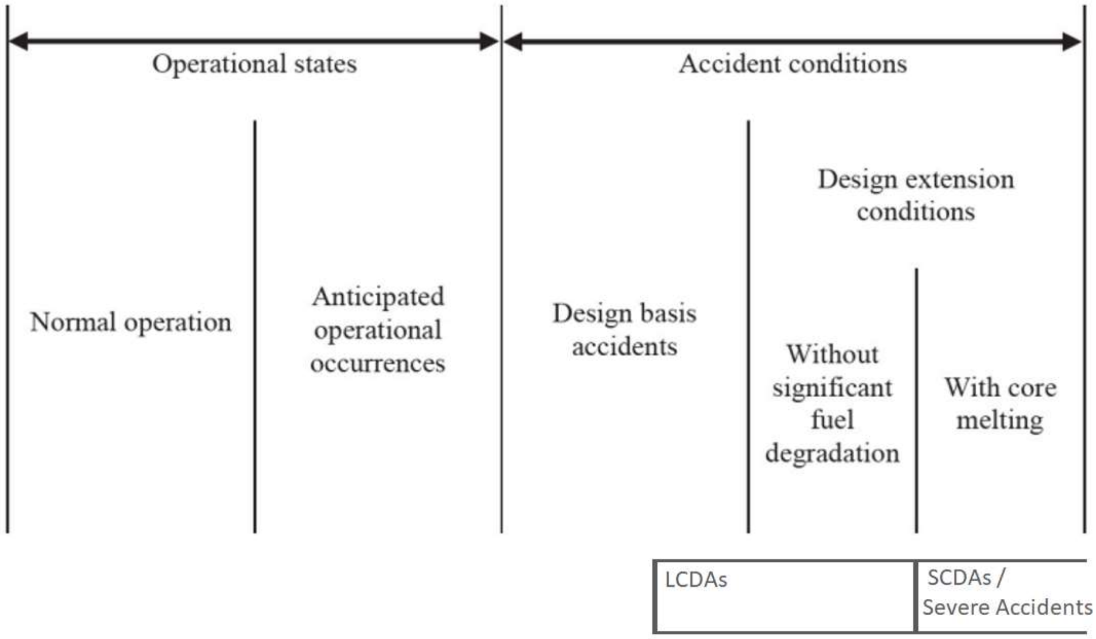


FIG. 1. Plant operational states and accident conditions (adapted from [4]).

It is usually assumed that LCDAs are part Design Basis Accidents (DBAs) (e.g., single channel events with limited core damage) and extend into the DEC-As. SCDAs are almost equivalent to DEC-Bs however it is usually assumed that SCDAs involve significant core degradation and are more in line with the definition of severe accident. As an approximation, the terms SCDAs, DEC-B and severe accidents are used interchangeably in this document. The modelling approach used in safety analysis of LCDAs and SCDAs is based on their classification as DBAs or DECs.

Following are the typical examples of events without significant fuel degradation analysed for PHWRs [5]:

(a) Steam generator tube rupture with failure of isolation of affected Steam Generator (SG);
(b) Station blackout;
(c) Main steam line break along with SG tube rupture;
(d) Main steam line break along with containment spray system failure;
(e) Loss of Coolant Accident (LOCA) along with containment spray system failure;
(f) LOCA along with Emergency Core Cooling System (ECCS) failure.

Following are the typical examples of events with core melting (or with significant core damage) analysed for PHWRs without and with various mitigation measures such as water injection to Calandria Vessel (CV), calandria vault, steam generator and primary heat transport system:

(a) Large Break LOCA with loss of ECCS and moderator circulation system failure;
(b) In-core LOCA along with loss of ECCS and moderator circulation system failure;
(c) Station Black-Out (SBO) with additional failures of fire water system injection to SG;
(d) SBO with additional failures of fire water system injection to SG along with small leak in primary heat transport system (rupture of one instrument tube).

2.2.SAFETY CASE

The safety case of an NPP evaluates the potential hazards associated with its operation to prove design adequacy and ensure that all the relevant safety requirements are met by means of analysis of various postulated initiating events. A set of conditions are also analysed to assess the capability of the plant to prevent and mitigate accidents that are more severe than design basis accidents. These conditions include design extension conditions. For accident conditions that could lead to early radioactive releases, or large radioactive releases, the safety case needs to provide a demonstration that such conditions have been practically eliminated.

The safety analyses presented in safety cases use deterministic and complementary probabilistic approaches to provide a comprehensive view of the overall safety of the plant.

2.2.1. Deterministic safety analyses

Deterministic analyses of severe accidents are carried out to provide assurance that the plant safety systems, or additional safety features developed for design extension conditions, can prevent or mitigate accident conditions that are more severe than the design basis, or maintain the integrity of the containment under those conditions.

These activities include reviewing the outcomes of severe-accident computer code analytical simulations to demonstrate the effectiveness of the safety feature or mitigating strategy. Alternatively, simulations can provide the anticipated design extension (e.g., environmental) conditions to which equipment or humans would be exposed during severe accident conditions as an input to equipment qualification, plant habitability assessments, defining design objectives for portable emergency equipment deployment, and as an input to design modification processes for emergency mitigating equipment or other safety features.

2.2.1.1. Methodology

Best estimate/realistic approach is generally used for the safety analysis of severe accidents with appropriate recognition of uncertainty existing in the timing and severity of the phenomena. Plant specific data including plant operational parameters, set-points and plant systems configuration and performance characteristics are preferably used in the analysis. Typically, for long-term sequences, the non-permanent equipment is considered operational and is assumed available in accordance with the emergency operating procedures or accident management guidelines. Consideration is given to the fact that severe accident analyses are usually performed for a bounding case using a best estimate, and not conservative, approach and do not provide margins [6-7].

In addition, a systematic process involving expert engineering judgment is typically employed to pinpoint potential cliff edge effects. This includes but is not limited to, fuel dry-out, inventory depletion and pressure boundary failure. The aim is to identify the dominant parameters by evaluating their impact on analysis results. For scenario of high likelihood and potential large impact, sensitivity analyses are used to verify to the extent practicable that, when more conservative assumptions are used, there are still margins in regards to cliff edge effects [6-7].

2.2.2. Probabilistic safety assessment

Probabilistic Safety Assessments involve a thorough and integrated evaluation of a NPP's vulnerabilities. This assessment takes into account the initial plant state, the likelihood, progression, and consequence of equipment failures, as well as operator responses. PSA produces numerical estimates corresponding to the relative level of safety of the design.

Simulation and modelling of severe accident in the context of PSA refers to the deterministic safety analyses of severe accidents performed in support of PSA to identify, for different combinations of equipment failures and human errors, a minimum set of safety features that can prevent nuclear fuel degradation, containment failure and risk to the public. Those analyses are performed with the same methodology as described in Section 2.2.1. More specifically, the role of severe accident analysis in PSAs is to:

(a) Level-1 PSA:

The results of the Level-1 PSA are key input into the start of level 2 PSA analysis. It determines and estimates the sequences of events that may lead to the loss of core geometry and massive fuel failures including the determination of core damage states.

(b) Level-2 PSA:

Severe accident analyses are performed in support of Level-2 PSA to:

- Determine the success criteria of containment engineered safety features and mitigation measures;
- Identify integrity challenges to physical barriers (containment);
- Identify a barrier's failure mode when challenged;
- Evaluate whether an accident scenario may pose challenges to multiple barriers;
- Determine the time available for operator actions in accident scenarios;
- Establish mission time of the integrity of containment systems;
- Help in estimation of nodal question probabilities for the phenomenological aspects (e.g., hydrogen deflagration/detonation, Molten Corium Concrete Interaction (MCCI), steam explosion, fuel coolant interaction);
- Estimate the source term (release from the containment);
- Provide inputs for identification of large and/or early releases depending on the accident progression.

(c) Level 3 PSA:

Level 3 PSA is not widely performed. In the Level 3 analyses, more uncertainty is added by including atmospheric dispersion and low-dose-response relationships in the evaluation of health effects in large populations [8], but can present an advantage for communicating potential deterministic and stochastic health risks to the public and stakeholders. Level 3 PSA may also be utilized to assist with the development of off-site emergency plans or to confirm the effectiveness of existing emergency plans, zone sizing and response actions.

Severe accident analyses performed in support of level 2 PSA calculate the radioactive source term that would be released from containment into the atmosphere. The source term is then used in level 3 PSA for environment dispersion analysis and estimation of health effects during a postulated severe accident.

2.2.3. Selection of scenarios or accident sequences

A diverse range of accident scenarios is carefully chosen to effectively encompass the possible developments of initiating events into design extension conditions, along with a comprehensive set of plant damage states. Probabilistic Safety Assessments Level 1 and 2 (see Section 2.2.2) in combination with engineering judgement are used for selection of the scenarios [9].

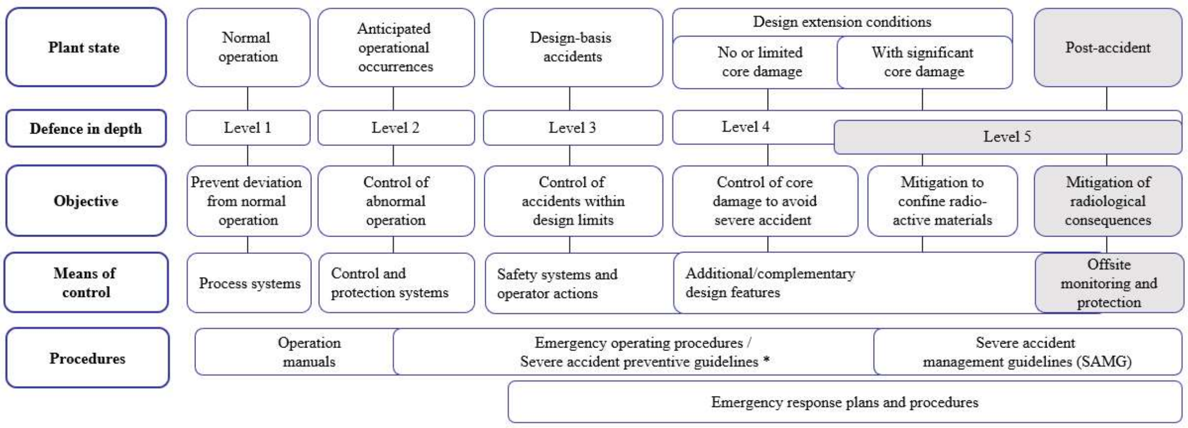
The following aspects of accident scenarios that would lead to core damage and subsequent potential challenge to Fission Product (FP) barriers are to be taken into account:

(a) Sequences with no operator action or inappropriate operator actions leading to core damage;
(b) Availability and functionality of equipment, including instrumentation, and the habitability of working places under anticipated environmental conditions;
(c) Potential cliff-edge effects.

Guidance on selection of accident sequences can be found in IAEA SSG-3 [10] and SSG-4 [11].

2.3. BASIS FOR ACCIDENT MANAGEMENT

Accident management is an integral component of Defence in Depth (DiD) levels 3 and 4, and the emergency response plan of level 5. Figure 2 shows an overlap of accident management (level 4) and emergency response plan (level 5) during the management of a severe accident.



**Some states have emergency preventing and mitigating equipment guidelines*

FIG. 2. Overview of accident management including severe accidents.

2.3.1. Severe accident management guidelines

Accident Management includes actions or measures that can be divided in preventive measures, to prevent core damage, and mitigatory measures, for managing severe accidents and mitigate the consequences of significant core degradation. The corresponding actions are typically documented in Emergency Operating Procedures (EOPs)/severe accident preventive guidelines and Severe Accident Management Guidelines (SAMGs), respectively. SAMGs are symptom-based actions that are an essential part of accident management to minimize radiological risk to the public.

The Diagnostic Flow Chart (DFC) and Severe Challenge Status Tree (SCST) serve as the primary foundations for proficient SAMGs diagnostic tools. Figure 3 illustrates the controlling platform for both the DFC and SCST.

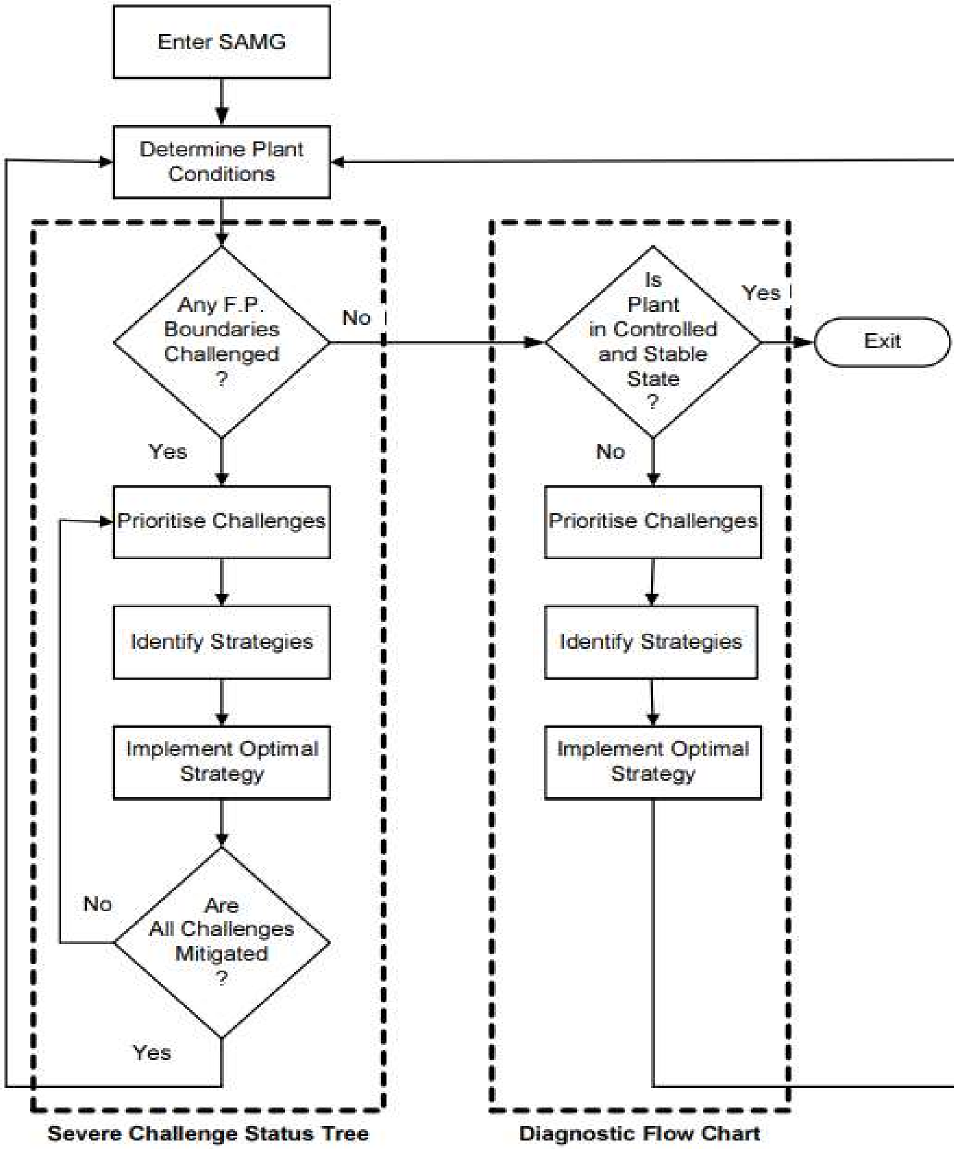


FIG. 3. Decision logic for DFC and SCST [1].

The actions for the DFC and SCST are obtained from plant parameters and are linked to the Severe Challenge Guidelines (SCGs) and Severe Accident Guidelines (SAGs):

— SCGs which are entered to prevent an imminent failure of containment with action needed to be taken immediately with negative consequences not considered. This is the highest priority;

— SAGs which are entered to prevent a future problem with a safety barrier, and which considers negative consequences of actions which have to be considered, so it is possible that no action is taken.

Figure 4 shows the linkages between the SAMG suite of documents and the DFC. While the objective is to protect the public by preserving containment integrity, critical parameters for both the SCST and DFC are typically monitored in parallel during emergency response. Various SCGs may also be implemented in parallel depending on plant conditions. A similar approach can be applied to various SAGs.

The specific objectives of severe accident management are outlined in IAEA-TECDOC-1594 [1]. Since the Fukushima Daiichi NPP accident , the SCG 5, Control Irradiated Fuel Bay conditions, has been added to the list of SCGs presented in [1].

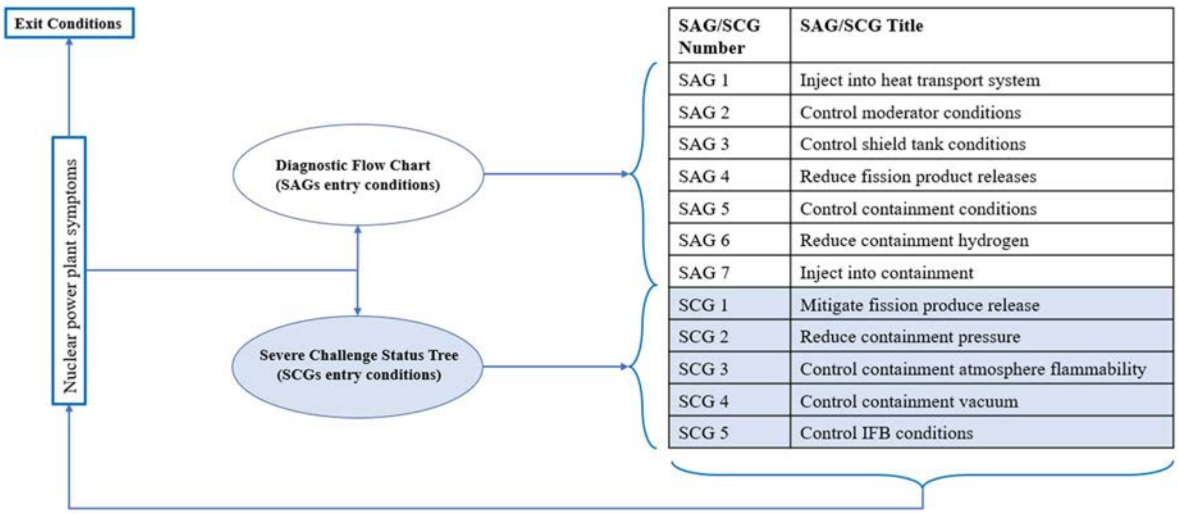


FIG. 4. Linkage of SAMG documents to DFC.

2.3.2. Analytical simulations for effective mitigation measures

The analysis generally begins with the accident sequences/scenarios selection, without operator intervention, leading to core damage. Grouping of the accident sequences with similar characteristics may be carried out to limit the number of sequences that are analysed to reduce the computational efforts. Such a categorization can be based on several indicators of the plant state such as the postulated initiating event, the shutdown status, the coolant pressure boundary, the secondary heat sink, the system for the removal of containment heat and the containment boundary among others.

Analytical simulations are carried out to assist the development of strategies that an operator has to follow if the Emergency Operating Procedures (EOPs) fail to prevent the severe accident. The analyses performed by using one or more computer codes to model the relevant physical phenomena.

The analyses are used to identify what are the challenges to be expected during the progression of accidents, and what phenomena would occur. They can be used to provide the basis for identifying plant capabilities and strategies that can be transferred into SAMGs where severe accident progression may be mitigated or terminated and, thus, these analyses are useful in developing a set of guidelines for managing accidents and mitigating their consequences.

Severe accident analysis provides the input that is necessary to specify the operator actions needed to terminate accident progression or mitigate consequences of the event. It also provides indicative timelines for the operator to take actions in response to some accidents, and the analysis is an important element of the verification of the accident management strategies. In the development of the response strategies, sensitivity analyses are generally performed on the timing of the necessary operator actions, and these calculations can be used to optimize the procedures.

Preventive measures are analysed to investigate what actions are capable of preventing or delaying the onset of core damage and to derive conditions for initiation/termination of mitigatory actions. Severe accident analysis also provides input for the timelines required for the availability of non-permanent equipment [12].

2.3.2.1. Objective of the analysis for accident management programmes & severe accident management guidelines

Analyses are performed, for the postulated accident scenarios expected in all relevant normal operating and shutdown states, considering all sources of radioactive material (e.g., reactor core and spent fuel pools), and considering multi-unit accidents. These analyses are performed with following objectives as part of the development of accident management programme, as applicable:

(a) For identification of challenges to integrity of barriers and capabilities and to demonstrate the acceptability of the identified solutions to support the accident management strategies and measures. This also calls for the analysis without crediting the mitigatory measures;
(b) For formulation of the technical basis for development of strategies, procedures, and guidelines;
(c) For verification and validation of procedures and guidelines (with other safety analysis tools, if available);
(d) For source term assessment;
(e) To support the decision-making regarding plant upgrades;
(f) For arriving at the conditions required for determining survivability of equipment/instrumentation with reasonable assurance;
(g) To arrive at working conditions/habitability of working places for personnel involved in the execution of the accident management actions;
(h) For identifying the accident scenarios for personnel training and exercise purposes.

2.3.2.2.Inputs for development of guidelines and procedures

Severe accident analyses provide the technical basis for the development of guidelines, procedures and strategies. Guidance for those is provided in SSG-54 [9], while Appendix I provides some examples applicable to PHWRs.

2.4. EMERGENCY RESPONSE, TRAINING AND DRILLS

Severe accident analyses can be performed by the emergency operations and technical support centre personnel to support their duties during severe accident management, which are to diagnose the plant conditions, select and recommend the viable mitigating actions to implement while taking into account both positive and negative impacts, and evaluate the effectiveness of the already implemented actions. Severe accident analyses also serve as the technical basis for training and drills.

2.4.1. Fast running codes and tools for emergency response

Following declaration of an emergency, the technical team of subject matter experts within the activated emergency response organization (both on-site and off-site), can use fast-running tools to:

- Assess the onsite situation by evaluating the accident scenario and determine appropriate actions;
- Quantify the risk by determining if a radiological release would occur and what would be its magnitude;
- Determine the offsite consequences and provide support to external stakeholders to ensure the appropriate actions are taken.

The following section provides an example of fast-running tools usage by the Canadian regulator under emergency conditions.

2.4.1.1. Example of a fast-running tools in emergency operations centre in Canada

Per the Canadian Nuclear Safety Commission (CNSC) Nuclear Emergency Response Plan [13], the role of Emergency Operations Centre (EOC) of the Canadian regulator is to gather information of onsite conditions from the licensees, assess the actions of the licensees, and evaluate the potential consequences of such actions. This includes informing key stakeholders in a timely manner and provide technical support and advice. The EOC also supports communication activities related to the event and its response.

The Reactor Safety Group is activated within the EOC and is responsible for using fast-running codes and tools to assess the safety significance of the emergency and provide appropriate technical advice and support. EOC fast running codes and tools are used to diagnose and prognose severe accidents and used as inputs to verify actions and decisions in the event of a nuclear emergency at an NPP.

The suite of EOC technical tools is based on four fission product barriers with four diagnosis (4D) and four prognoses (4P) therefore named the 4D/4P tool. Figure 5 shows the 4D/4P tool.

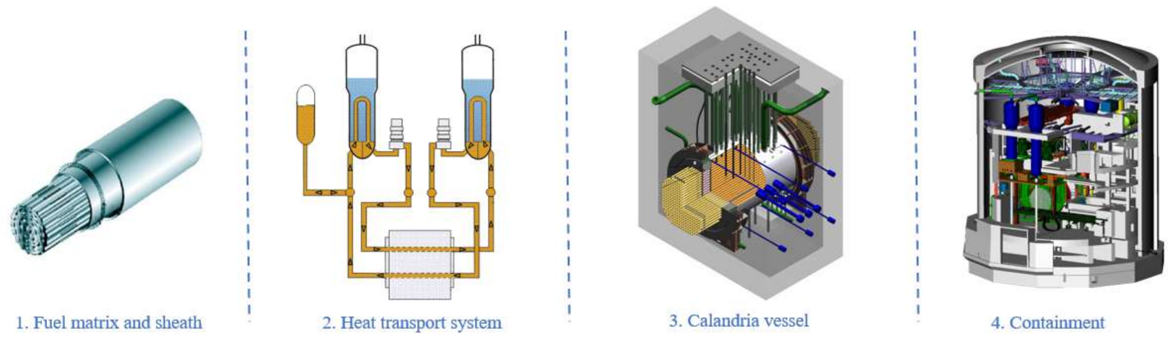


FIG. 5. CANDU 4D/4P-fission product barriers.

Figure 6 illustrates that this tool is used to perform an assessment of the integrity of the CANada Deuterium Uranium (CANDU) reactor's 4 fission product release barriers in a systematic approach.

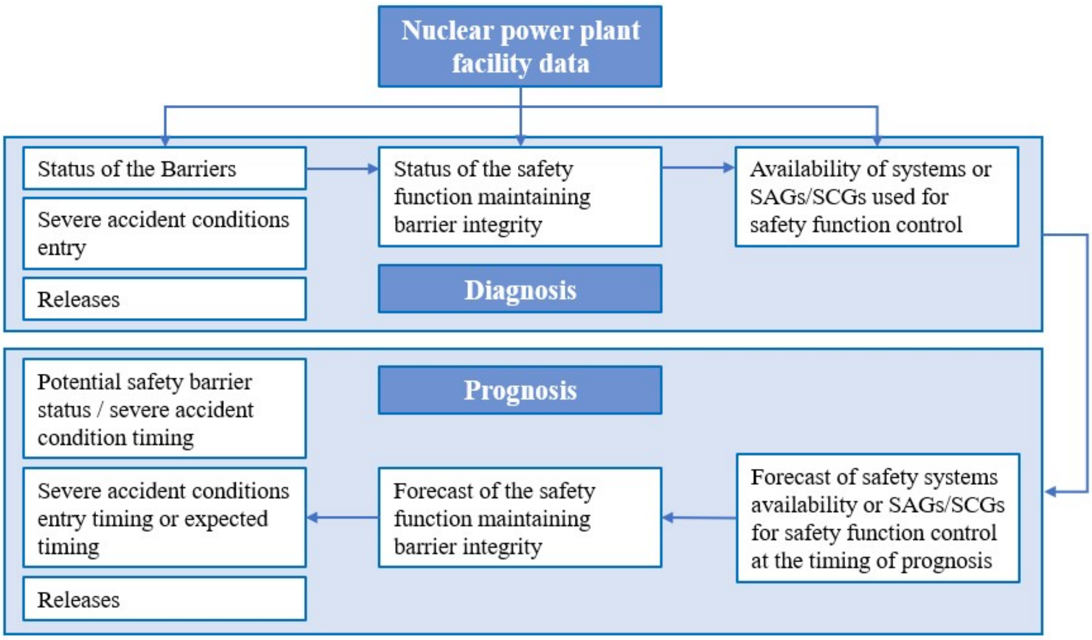


FIG. 6. Illustration of the CANDU 4D/4P systematic process.

The 4D-4P methodology aligns with the IAEA accident assessment and prognosis methodology [14]. The tool is coupled with the Reactor Assessment Tool (RAT) of the IAEA, and plant data trending is received from the impacted NPP in case of an accident.

The 4D-4P tool interfaces with other in-house tools to help determining the hypothetical source term in case of an accident. The source term tools include:

— The severe accident handbook which is a compilation of various simulated runs from the MAAP-CANDU/ GRAPE (GRaphical Animation Package Extension) and licensee documentations;
— The EOC source term database which consists of MAAP-CANDU/GRAPE runs of various accident scenarios and initiating events;
— VETA (Visual EOC Source Term Assessment), Unified Rascal Interface (URI),

CHAT (Combustion Hydrogen Assessment Tool) and REPRESS (Re-pressurization of containment). VETA and URI deal with source term modelling, while CHAT addresses hydrogen computation and assessment tool and RREPRESS deals with re-pressurization of multi-unit containment systems with vacuum building units;
- BAYLOCA (Spent fuel bay loss of coolant accident) for spent fuel bay accidents.

2.4.2. Simulators, training and drills

The objective of emergency drills and exercises is to provide an opportunity for the plant staff responsible for the emergency response and accident management to demonstrate the use of the plant procedures (Abnormal Incident Manual or EOP) and guidelines (SAMG and Emergency Mitigating Equipment Guides) to respond to a hypothetical event leading to different stages of core damage or in the progression of an accident from a routine postulated initiating event to a severe accident. This activity also enables an emergency response organization to evaluate personnel and understand the plant-specific practices with respect to accident management and emergency response and have the capability to provide insightful feedback into the overall Severe Accident Management (SAM) programme. The IAEA-TECDOC-1411 [15] and IAEA-TECDOC-1963 [16] provide guidance for control room stimulator training as well as the maintenance and upgrades to the simulators.

Emergency drills and exercises are traditionally conducted based on predefined scenarios. The drills evolution is therefore pre-established, and participants are notified about the conditions in the field by means of verbal notifications or written notes. Alternatively, a severe accident code can be integrated into the full-scope simulator. In this case, the operating crew and the emergency response team use the same information tools as they would be using during a real situation such as the parameters shown on the control rooms panel instrumentation. Although the simulator allows for real-time accident progression, severe accidents scenarios can expand up to a few days while emergency drills typically span over a few hours only. The scenario therefore needs to be previously ran on the simulator, and intermediary initial conditions stored to allow the advancement throughout different stages of the exercise within the time allowed.

In addition to their usage in emergency drills, full-scope simulators with severe accident modelling capability can also be used to train the operating personnel in severe accident management, to demonstrate to technical personnel the phenomenon taking place during a severe accident and for SAMG validation. Models for spent fuel bay or facilities used in common by multiple reactors on the same site can also be simulated. This can increase the domain of the severe accident scenarios to be performed on the simulator.

A few considerations need to be accounted for while integrating severe accident capability within a full-scope simulator. For instance, communication and live data transmission from the main control room of the plant to the emergency control centre has to be developed and introduced in the simulator to provide the support team with the same information by using the same communication method as in a real situation. Additional water or power sources have to be integrated in the plant model to allow practicing the enabling instructions that implements the mitigations actions for severe accidents. Also, a smooth transition mechanism needs to be implemented when switching from the regular simulation code to the severe accident code to prevent mismatches between the parameters that could result in unrealistically discontinues trends in simulation.

Visualisation tools can be integrated in the simulator to better demonstrate to operations and technical personnel the phenomenon taking place during a severe accident such as fuel degradation and core melting. Training of operations and emergency response personnel can however only rely on information available on the control room and on applying the emergency response procedures.

2.4.2.1. Example of simulator upgrades to include severe accident modelling capabilities

Cernavoda NPP, in Romania, completed a simulator upgrade in 2021, and a severe accident code was introduced in the simulator together with a hardware upgrade and some model improvements. MAAP5-CANDU was integrated in the operator training simulator and severe accident scenarios such as large LOCAs, steam line breaks or SBOs were developed and tested in comparison to the results obtained with standalone MAAP5-CANDU. Facilities such as the containment filtered venting system, additional water sources to cool the reactor and additional power sources to be used during a severe accident were introduced in the simulator model to perform the mitigation actions described in the emergency procedures. A smooth transition mechanism from the classical simulation code (i.e., the code running during all design basis conditions) to severe accident code was implemented to prevent sudden modifications on parameters and trends. A communication software was introduced in the simulator to provide the offsite control centre with the same information in the same format as received from the main control room in case of a real event. Figure 7 shows an example of the dynamic and interactive 2D and 3D visualisations that are provided to better illustrate the evolution of the event. The upgraded simulator is now used in the initial and continuous training of the operating personnel and also in emergency drills with the operating crew and the response team.

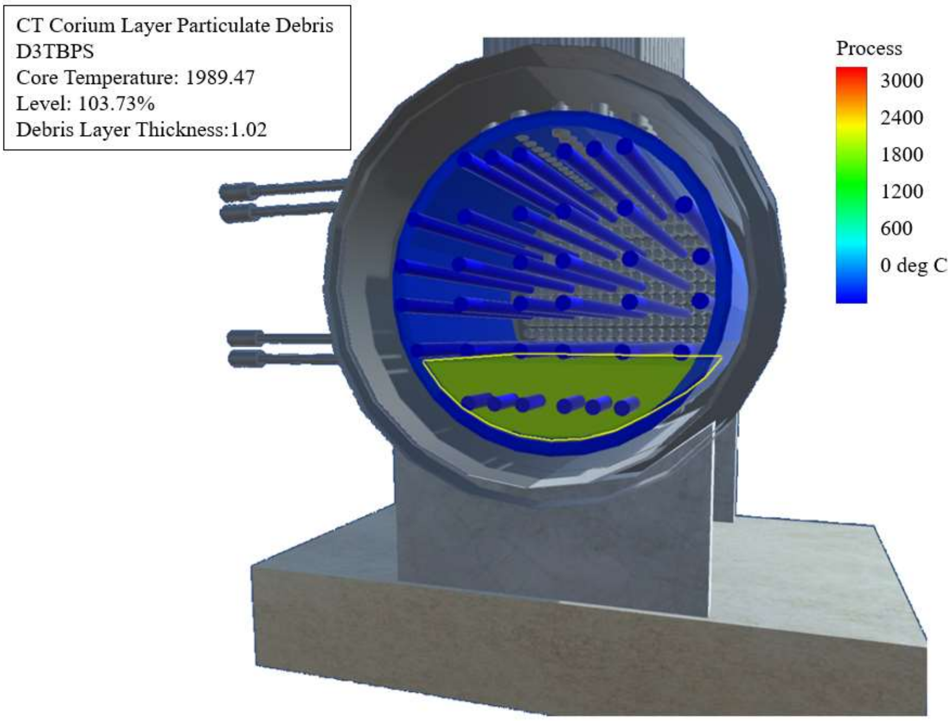


FIG. 7. A 3D view of the reactor showing corium at the bottom of the CV on the Cernavoda simulator (courtesy of L3Harris MAPPS, Canada).

3. REGULATORY UPDATES IN HEAVY WATER REACTOR COUNTRIES

The Fukushima Daiichi NPP accident demonstrated the importance of considering severe accidents in regulatory influence and emergency planning and response. One of the main actions from lessons learned from the accident was an improvement of the regulatory framework and processes including off site response of nuclear regulatory agencies. This section provides an overview of those regulatory framework for severe accidents in the PHWR MS that provided a presentation at the IAEA Technical Meeting, held in 2022.

3.1. CANADA

The role of the CNSC is to “regulate the use of nuclear energy and materials to protect health, safety, security and the environment” [1]. CNSC also “implements Canada’s international commitments on the peaceful use of nuclear energy, and disseminate objective scientific, technical and regulatory information to the public” [1].

3.1.1. Regulatory framework

The Nuclear Safety and Control Act (2000) is the enabling legislation that mandates the Canadian regulator to enforce its regulations through a transparent process [17].

Regulatory requirements are continuously updated based on a systematic and transparent, and risk-informed process that aligns with national interests, and international standards including that of IAEA’s safety standards. Regulatory documents are a key part of the CNSC's regulatory framework for nuclear activities in Canada. They generally present both requirements and guidance in a single document and distinguish between them by the use of mandatory (e.g., shall, must) and non-mandatory (e.g., should, may) language. Guidance provides direction to licensees on how to meet the requirements. If guidance to requirements has not been followed, the licensee has to justify how the alternate approach meets the regulatory requirements. Figure 8 shows the Canadian regulatory framework and the licensing process.

The CNSC uses performance of programmes divided in 14 safety and control areas (SCAs), grouped into the functional areas of management, facility and equipment and core control processes, to ascertain how licensees comply with the regulatory requirements.

The CNSC uses IAEA safety standards as the basis for many of its regulatory documents. Hence there is good correspondence with the fundamental safety approach described in the IAEA Safety Fundamentals [18] and the suite of safety standards and guides derived from them.

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[1] Quotation from the CNSC webpage: https://nuclearsafety.gc.ca/eng/.

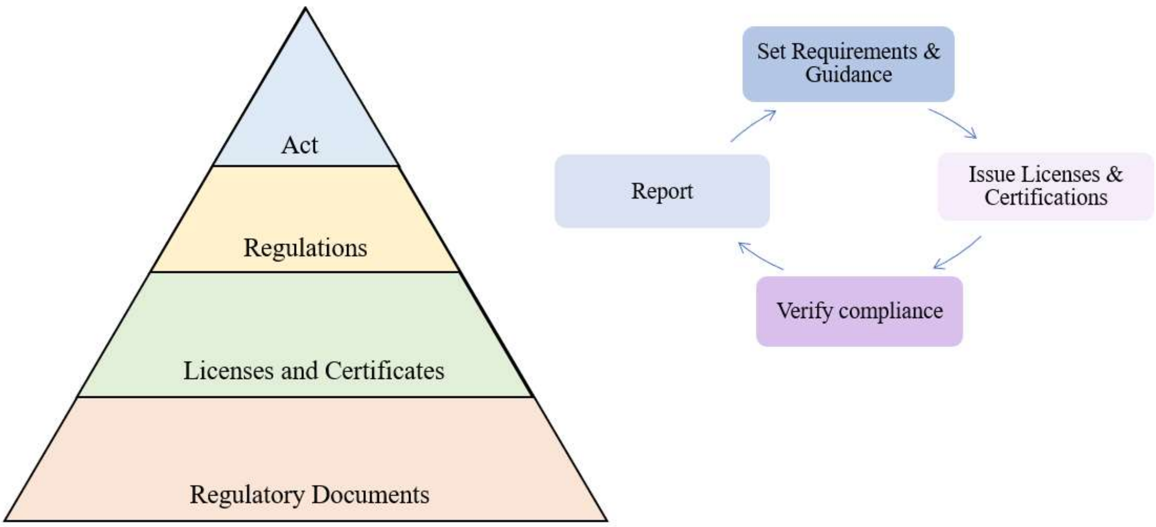

FIG. 8. CNSC's regulatory framework and licensing process.

3.1.2. Plant states, events and dose criteria

In Canada, the plant states and dose criteria as outlined in the REGDOC 2.5.2 [19] can be summarized in Table 1.

TABLE 1. DEFINITION OF PLANT STATES AND DOSE CRITERIA IN CANADA

	Plant Design Envelope					
Plant States	Normal Operation	Anticipated Operational Occurrences	Design Basis Accidents	Beyond Design Basis Accidents		
				Design Extension Conditions		Practically eliminated conditions
				No severe fuel degradation	Severe Accidents	
Public Radiological Acceptance Criteria	As low as reasonably achievable (Typically < 10µSv)	Dose Acceptance Criteria: 0.5 mSv	Dose Acceptance Criteria: 20 mSv	Safety goals and deterministic requirements: core damage frequency (CDF) < 10^{-5} small release frequency < 10^{-5}, large release frequency < 10^{-6}		

3.1.3. Requirements for severe accidents

Prior to the Fukushima Daiichi NPP accident , the G-306, Severe Accident Management Programmes for Nuclear Reactors, provided guidance on how to meet requirements for severe accidents [20]. Requirements and guidance for severe accidents have been amended to reflect lessons learned from the Fukushima Daiichi NPP accident, address findings from the CNSC's Fukushima Task Force Report [21], and modernize the regulatory framework for emergency management. This resulted in the publication of the version one of the REGDOC 2.3.2 in September 2013 as a guidance document [22]. Two years later, the version 2 of the REGDOC 2.3.2 was published as a regulatory requirement [23].

Within the Canadian regulatory framework, requirements for severe accident as enshrined in REGDOC 2.3.2 are included in a series of supporting regulatory documents including Canadian Standards Association (CSA) norms:

- REGDOC-2.4.1, Deterministic Safety Analysis [24];
- REGDOC-2.4.2, PSA for Nuclear Power Plants [25];
- REGDOC-2.5.2, Design of Reactor Facilities [19];
- DIS-14-01, Design Extension Conditions [26]: Promotes common understanding of the concept of design extension conditions;
- REGDOC-2.10.1, Emergency Preparedness and Response [27];
- REGDOC-2.3.3, Periodic Safety Review [28] ;
- CSA N290.3-16, Requirements for the containment system of nuclear power plants [29];
- CSA N290.16, Requirements for beyond design basis accidents [30];
- CSA N286.7-2016, Quality Assurance of Analytical, Scientific and Design Computer Programs [31].

The REGDOC-2.3.2 gives the overall objectives of accident management, to achieve the overarching nuclear safety objective of protecting individuals, society, and the environment from harm against radiological hazards and hazardous substances by fulfilling the following fundamental safety functions:

- "Control of reactivity;
- Removal of heat from the fuel;
- Confinement of radioactive material;
- Shielding against radiation;
- Control of operational discharges and hazards substances, as well as limitation of accidental releases;
- Monitoring of safety-critical parameters to guide operator actions." [23].

In section 4.2.4, *Supporting analysis*, the REGDOC 2.3.2 outlines the requirements for accident management and the analysis to support its compliance in Canada. It states that:

"Safety analysis should be used to assist in developing accident management measures by:

- Formulating the technical basis for identification of reactor challenges and capabilities and development of strategies, measures, procedures, and guidelines;
- Demonstrating the acceptability of the identified solutions to support the selected strategies, measures, procedures, and guidelines against the established criteria;
- Determining the reference source terms and accident conditions for environmental qualification of equipment for DBAs and survivability/operability assessments of equipment for BDBAs, including severe accidents." [23].

Per the REGDOC 2.3.2, safety analysis for SAM should be performed using a best estimate approach while accounting for uncertainties if the knowledge level of the phenomena modelled is low and the supporting experimental data is insufficient.

3.1.4. Regulatory oversight of severe accident management

Figure 9 shows the physical and procedural barriers and outlines the interfacing of REGDOC 2.3.2 (Accident Management Program) and REGDOC 2.10.1 (Emergency Preparedness). The physical barriers to fission product release, with the use of procedural barriers are used by the CNSC to provide oversight of SAM.

Oversight of SAM by the CNSC includes, in addition to inspections, staff interviews and observations of the SAM drills/exercises, a review of the stations specific SAMG documentation, including:

- Generic SAMG technical basis;
- Special topics such as in-vessel retention, hydrogen mitigation, containment venting strategy and performance, time constraints for implementing SAMG actions;
- Control room guides, diagnostic flow chart, severe challenge status tree, severe accident guides, etc;
- SAMG Verification and Validation (V&V);
- Emergency Mitigating Equipment and their integration with SAMG and Abnormal Incident Manual (AIMs);
- Responsibilities of personnel involved in SAM.

Details of the plant specific SAMG evaluation process are provided in Reference [32]. Upon need basis, CNSC's oversight also includes performing independent analyses with licensees' tools.

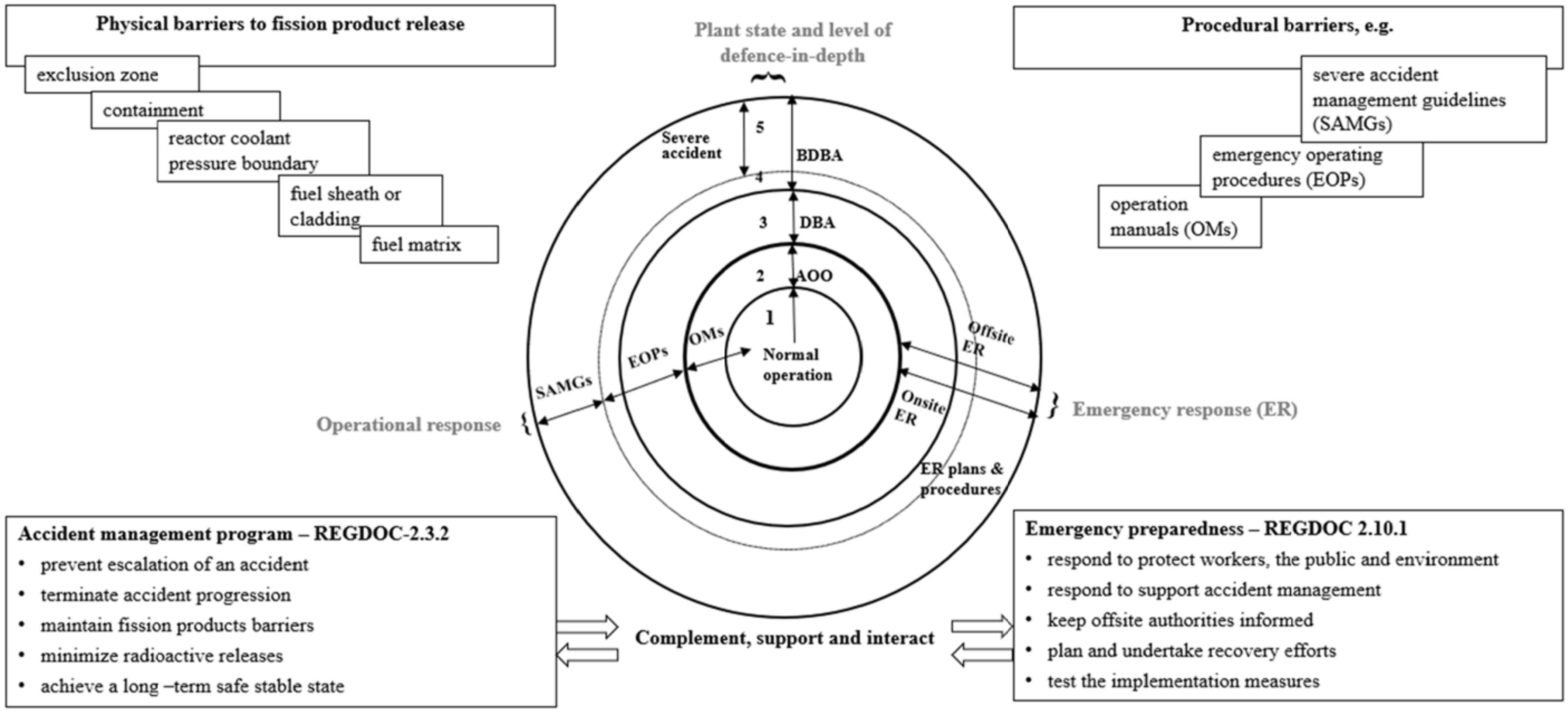

FIG. 9. Overview of accident management and plant states, physical and procedural barriers, and their linkages to REGDOC 2.3.2 and REGDOC 2.10.1.

3.2. INDIA

This section provides an overview of the regulatory framework for PHWR severe accidents in India.

3.2.1. Regulatory framework

The Atomic Energy Act [33], and the Rules framed thereunder provide the main legislative and regulatory framework pertaining to atomic energy in India. Figure 10 shows the hierarchy of legal and regulatory framework in India. The National Policy and Strategy for Safety paves the way forward for establishing national legal framework for nuclear and radiation safety. The national policy on safety is indicated at conceptually higher level in the hierarchy since such policies and strategies are legally not enforceable. The hierarchy has five levels starting from the legal statutes followed by regulatory requirements and guidance documents developed by Atomic Energy Regulatory Board (AERB), the national regulatory agency for regulation of nuclear and radiation facilities and activities in India.

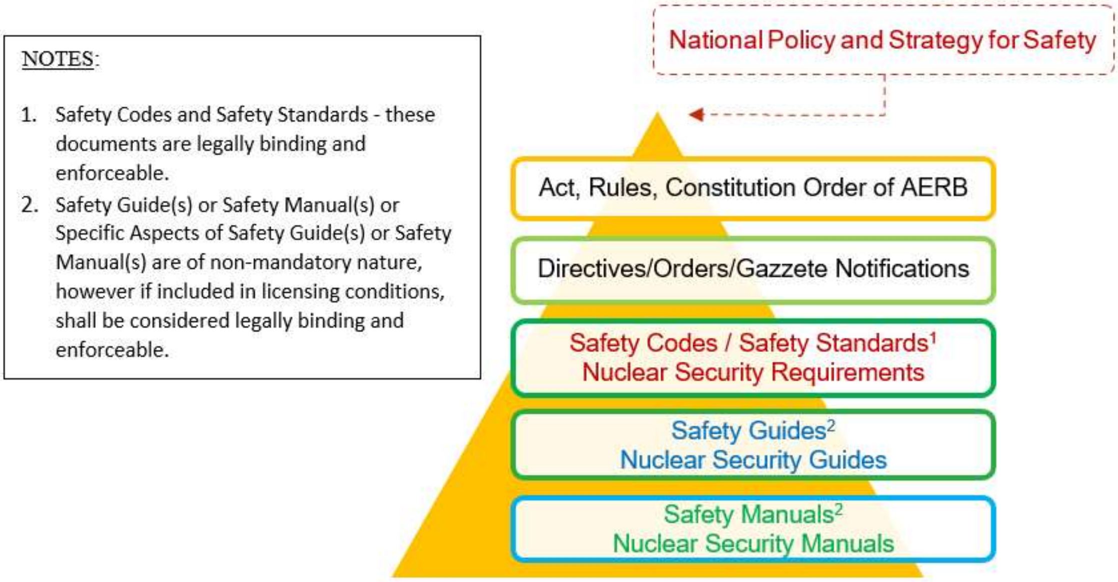

FIG. 10. AERB regulatory framework.

Under the Atomic Energy Act, the Central Government promulgated the following Rules:

- Atomic Energy (Radiation Protection) Rules [34];
- Atomic Energy (Safe Disposal of Radioactive Wastes) Rules, [35].;
- Atomic Energy (Working of the Mines, Minerals and Handling of Prescribed Substances) Rules, [36];
- Atomic Energy (Arbitration Procedure) Rules, [37];
- In addition, the central government has also framed the Atomic Energy (Factories) Rules [38], to administer the requirement of Factories Act in the nuclear establishments to ensure industrial safety.

These acts and rules are legally binding to all entities within India including Regulatory Body and the regulated parties and hence included in the topmost level of the hierarchy. The second level consists of legal instruments issued under a statute/legislative enactment or in discharge of executive functions. The next three levels of hierarchy are the regulatory documents on safety and security aspects, which are developed by AERB.

The legal and regulatory framework in India did not require any substantial changes in the aftermath of the Fukushima Daiichi NPP accident. However, regulatory requirements and guidance documents were updated.

One of the mandates of AERB is to formulate safety requirements for nuclear and radiation facilities. The legal power of issuing safety codes and standards, and to enforce requirements is provided by the Atomic Energy Rules. For NPPs, AERB has issued safety codes, site evaluation, design, operation, radiation protection, emergency preparedness and response, radioactive waste management & quality assurance and several safety guides & manuals under these codes.

Safety Codes establish objectives and specify minimum requirements that have to be fulfilled to provide adequate assurance for safety in nuclear and radiation facilities. Safety Guides provide guidance and indicate methods for implementing specific requirements prescribed in the Safety Codes/Standards. In addition to these, AERB also issues Safety Manuals which elaborate specific aspects and contain detailed technical information and procedures.

During the development and revision of regulatory documents, AERB also considered IAEA standards and the regulatory documents of other countries' regulatory bodies. The regulatory documents are periodically reviewed and updated based on experience and scientific developments and for harmonization with current IAEA standards and regulatory documents developed by other regulatory agencies.

AERB established a working group to identify existing regulatory or safety documents that would need review or revision. The group also identified additional provisions or requirements that would need consideration for addressing the lessons learned from Fukushima Daiichi NPP accident. Based on the recommendation of the working group, number of requirements and guidance documents are updated. Those are described in the following subsections.

3.2.2. Plant states, events and dose criteria

The plant states definitions have been updated by changing the description of the reactor states and introducing additional states into the design envelope, i.e. DECs. The concept of practical elimination is also introduced in the design process, which is depicted in the definition of the plant states. Table 2 summarizes the plant states definitions post Fukushima Daiichi NPP accident per Reference [39].

TABLE 2. DEFINITION OF PLANT STATES AND DOSE CRITERIA IN INDIA

<table>
<tr><td colspan="5">Plant Design Envelope</td><td></td></tr>
<tr><td colspan="2">Operational States</td><td colspan="3">Accident Conditions</td><td rowspan="3">Practically Eliminated Conditions</td></tr>
<tr><td rowspan="2">Normal Operation</td><td rowspan="2">Anticipated Operational Occurrences</td><td rowspan="2">Design Basis Accidents</td><td colspan="2">Design Extension Conditions</td></tr>
<tr><td>DEC-A - Without core damage /Without significant fuel[2] degradation</td><td>DEC-B[3]
With core damage/With core melt</td></tr>
<tr><td></td><td></td><td></td><td></td><td colspan="2">Severe Accidents</td></tr>
</table>

Post Fukushima Daiichi NPP Accident, a frequency informed DiD based approach has been adopted for categorization of design basis events in AERB/NPP-WCR/SG/D-5 (Rev.1) (Reference [5]) as given in Table 3.

TABLE 3. EVENT CATEGORIES TO BE ANALYZED

Plant State	**DiD Levels**	**Frequency (/year)**	**Category**	**Remarks**
Normal Operation	I	-	1	Normal Operation includes operational transients
Anticipated Operational Occurrences	II	$10^{-2} \leq f$	2	Single Failure Initiating Events
Design Basis Accidents	III	$10^{-6} \leq f \leq 10^{-2}$	3	Single Failure Initiating Events
Design Extension Conditions (without core melt / core damage)	IV	$f < 10^{-4}$	4A	Multiple Failures and Rare External Events
Design Extension Conditions (with core melt / core damage)	IV	$f < 10^{-6}$	4B	Event Sequences leading to core melt

The limit of public exposure during normal operation and during anticipated operational occurrence is prescribed as an effective dose of 1 mSv in a year. This limit is unchanged and valid as on date. The acceptable limit for dose for a member of the public was prescribed as an effective dose of 100 mSv for the whole body. This is updated after the Fukushima Daiichi NPP accident as an effective dose of 20 mSv/year following the event. Acceptance criteria for DECs are defined as safety goals: CDF< 10^{-5}, Large Early Release Frequency (LERF) < 10^{-7}.

3.2.3. Requirements for severe accidents

The requirements on severe accidents were updated and included in the AERB safety code, AERB/NPP-LWR/SC/D [40] published after the Fukushima Daiichi NPP accident. The Safety code, AERB/NPP-PHWR/SC/D [41], published in 2009, is currently under revision. The updated requirements on severe accidents included in LWR safety code are being incorporated in the current revision of PHWR safety code. The important requirements, which are updated are as given below:

Section 4.3.3 of the safety code states that:

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[2] Fuel stored in fuel pool as well as fuel within reactor core.

[3] The 'Core melt' for LWRs/fast breeder reactors and 'Core Damage' for PHWRs. Further, in case of PHWRs, as an exception, single channel events resulting in fuel failure/melt in the affected channel should not cause failure/melt of other channel and it comes under DBAs.

“Defence in depth level four shall include consideration of design extension conditions (DEC). The DEC are accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. DEC include severe accident conditions involving significant core degradation or core melt. The severe accident sequences which may lead to early or large radioactive releases shall be practically eliminated. A clear distinction shall be introduced in level four between means and conditions for DEC without core melt and DEC with core melt” [40].

Section 4.5.4 (iii) of the safety code states that:

“In case of severe accident e.g., accidents with core melt within design extension conditions, the release of radioactive materials should cause no permanent relocation of population. The need for off-site interventions should be limited in area and time” [40].

Section 5.18.5 of the safety code states that:

“Severe accident management guidelines shall be prepared, taking into account the plant design features and the understanding of accident progression and associated phenomena” [40].

Section 5.20.4 of the safety code describes severe accident safe state as given below:

“Severe accident safe state is a state which shall be achieved subsequent to a design extension condition with significant core damage or core melt phenomena. Severe accident safe state shall be reached at the earliest after an accident initiation. It should be possible to maintain this state indefinitely […]” [40]

“As the plant state is in design extension condition with core melt (severe accident), the severe accident safe state should be preferably reached within about one week from accident initiation.” [40]

Section 5.1.11 of Safety Code on Nuclear Power Plant Operation states that:

“The authorised persons shall be covered by periodic retraining programme including the simulator retraining to maintain knowledge proficiency in coping with anticipated operational occurrences and accident conditions and, to minimise human errors” [42].

The simulators of the existing NPPs have features that simulates few DEC-A scenarios such as SBO, total loss of SG feed water and end shield circulation failure followed by process water failure to shutdown cooling heat exchangers. The modelling of DEC-B scenarios involving core degradation, hydrogen generation are not in the scope of simulators as the progression of these scenarios is slow and mitigating actions involve field actions.

3.2.3.1. Accident Management

Section 6.27 of the draft revision 2 of the AERB/NPP-PHWR/SC/D describes requirements for severe accident monitoring as given below:

“For the purpose of severe accident monitoring and management, appropriate means shall be considered for the plant, by which the operating personnel obtain information for event assessment, and for the planning and implementation of mitigating actions […]” [41].

“The measurement systems/instrument shall be capable of measuring over the entire range within which the measured parameters are expected to vary during accident conditions” [41].

In addition to the above, additional requirements are being incorporated based on IAEA SSR-2/1, Rev. 1 [43] and latest guidance included in SSG-88 [44]. These includes additional provisions to ensure diverse and flexible accident response capabilities are in place to provide a backup to permanent equipment and to the equipment already available for management of severe accidents in Section 6 K of the safety code, AERB/NPP-PHWR/SC/D (Rev. 2) (draft).

Per Section 6.49 of the safety code, AERB/NPP-PHWR/SC/D (Rev. 2) (draft) features to enable the use of non-permanent equipment should also be included in the design. This non-permanent equipment can however not be credited in the safety assessment of DBAs and DECs.

3.2.4. Regulatory oversight of severe accident management

The requirements for severe accident management are covered in Safety Codes on Design of NPPs. AERB has prepared Safety Guides on ‘Accident management Programme’, AERB/NPP/SG/D-26 [45]. There is some overlap in accident management at site and Emergency Preparedness and Response actions following accident. As a part of regulatory oversight of SAM by the AERB, the following activities are carried out:

- Review of Generic Technical Document on SAMG (by AERB);
- Preparation of site-specific SAMG document (by licensee);
- Issues considered are: strategies for in-vessel retention, hydrogen mitigation, containment venting, time constraints for implementing SAMG actions, entry and exit criteria, operator aids, technical support centre etc;
- Validation and Verification of SAMG;
- Diverse and flexible response approach including emergency mitigating equipment and their integration with SAMG;
- Responsibilities of personnel involved in accident management and On-Site Response Organization.

This is achieved by considering the physical barriers to fission product release, with the use of procedural barriers.

AERB also carries out independent analyses for identified safety cases submitted by the licensee based on the safety significance. Section 6.1.3 of the safety guide, AERB/NPP-PHWR/SG/D-19 provides guidance for independent analyses as given below:

“Additional independent analyses of selected aspects may also be carried out by regulatory body itself or on behalf of the regulatory body by technical support organization. The responsible organization should provide the necessary inputs and details for such analyses to the regulatory body as per agreed terms and conditions.”

3.3. REPUBLIC OF KOREA

This section provides an overview of the regulatory framework for PHWR severe accidents in the Republic of Korea.

3.3.1. Regulatory framework

Figure 11 shows the five levels of the regulatory hierarchy for nuclear safety: Nuclear Safety Act (NSA) [46], the Enforcement Decree of the NSA (Presidential Decree) [47], the Enforcement Regulations of the NSA (Ordinance of Prime Minister) [48], Regulations on Technical Standards for Nuclear Reactor Facilities, Etc. [49], and the Nuclear Safety and Security Commission (NSSC) Notice [50].

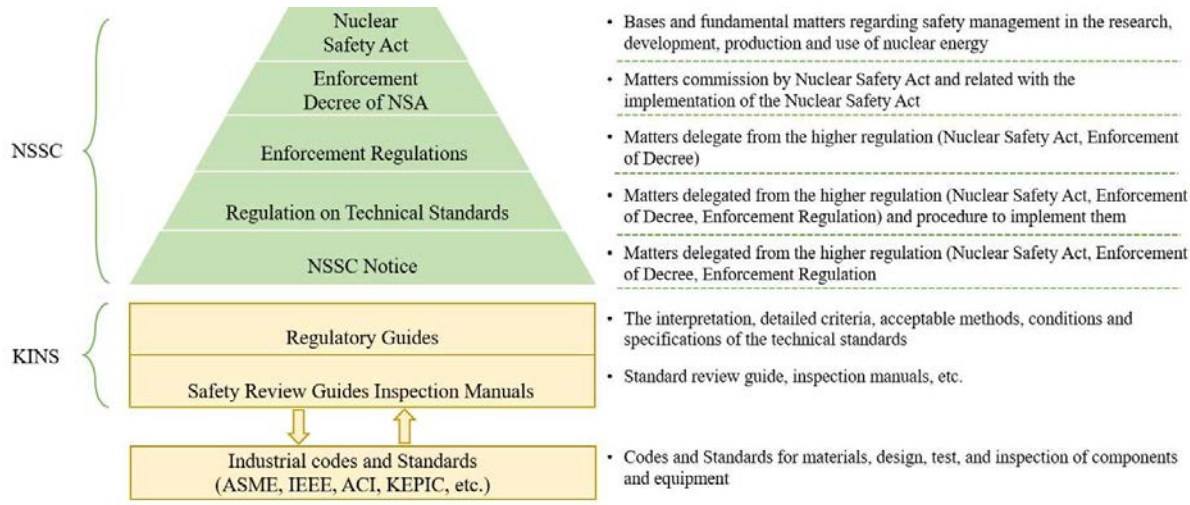


FIG. 11. Regulatory hierarchy of nuclear safety in the Republic of Korea.

The NSA is the highest law that governs the safety regulations for nuclear installations. It addresses key aspects pertaining to the management of safety in the research, development, production, and usage of nuclear energy. The NSA stipulates the matters related to nuclear safety regulation, comprehensive plans for nuclear safety, and the construction permit and operating license of nuclear installations.

The Enforcement Decree of the NSA states the matters commissioned by the NSA, and administrative matters related to the implementation of the NSA. The Enforcement Regulations of the NSA specify the matters delegated from higher statutes such as the NSA and the Enforcement Decree. The regulation on the Technical Standards gives details on the matters delegated from higher statutes (the NSA, the Enforcement Decree, and the Enforcement Regulations) and procedures to implement them, and specific administrative procedures. The NSSC Notice outlines the delegated matters from higher statutes (such as the NSA, the Enforcement Decree, the Enforcement Regulations, and the Regulation on Technical Standards) and delineates specific regulatory requirements and standards necessary for the enforcement of these laws. Furthermore, the regulatory body endorses and applies industrial standards relevant to the nuclear industry in the design and operation of nuclear installations.

As regulatory authority, the NSSC has the responsibilities for safety regulation of nuclear installations including the licensing of nuclear installations under the NSA. Korea Institute of Nuclear Safety (KINS) as the technical support organizations of the NSSC has developed and utilized the regulatory guides and Standard Review Guides (SRG). The regulatory guide and SRG interpret the specific matters of technical standards and give detailed descriptions on allowable methods, conditions and specifications to meet the technical standards.

3.3.2. Plant states, events and dose criteria

In the Republic of Korea, the plant states and dose criteria are based on References [46], [49], [51], [52], and can be summarized in Table 4.

TABLE 4. DEFINITION OF PLANT STATES AND DOSE CRITERIA IN THE REPUBLIC OF KOREA

	Plant Design Envelope				
Plant States	Normal Operation	Anticipated Operational Occurrences	Design Basis Accidents	Multiple Failures & Natural and artificial disasters exceeding the external causes in DBA	Severe Accidents
Frequency	-	$< \sim 10^{-2}$	$< \sim 10^{-4}$	$< \sim 10^{-6}$	Performance goals*
Dose Criteria	The annual dose at the exclusion area boundary due to gaseous effluents (1) Absorbed dose of air by gamma ray: 0.1 mG; (2) Absorbed dose of air by beta ray: 0.2 mGy; (3) Effective dose by external exposure: 0.05 mSv; (4) Skin equivalent dose by external exposure: 0.15 mSv; (5) Human body organ equivalent dose by particulate radioactive material, 3H, 14C, and radioiodine: 0.15 mSv. Annual dose at the exclusion area boundary due to liquid effluent (6) Effective dose: 0.03 mSv; (7) Human body organ equivalent dose: 0.1 mSv.		Equivalent to Reactor Site Criteria (250mSv)	Equivalent to Reactor Site Criteria (250mSv)	Equivalent to Reactor Site Criteria (250mSv)

* Performance goals:

Performance goals for new plants:

— CDF < 10^{-5}; LERF < 10^{-6}.

Performance goals for existing plants

— CDF < 10^{-4}; LERF < 10^{-5};

— Cs-137 over 100TBq < 10^{-6}/year.

3.3.3. Requirements for severe accidents

Republic of Korea started regulation on severe accidents by issuing severe accident policy in 2001. The key elements of severe accident policy are to require the licensee of the commercial nuclear reactors to establish safety goal, and perform probabilistic safety analyses, secure severe accident mitigation capability, and setup accident management plan to minimize radiological risk from the severe accidents. After severe accidents at Fukushima Daiichi NPPs in Japan, the Korean government conducted a comprehensive Special Safety Inspection. The Inspection was performed from March 23 through April 30 2011 to confirm the mitigating capabilities against severe accident resulting from huge earthquakes and tsunami which were beyond the design basis as those occurred in Fukushima. The government identified 50 post Fukushima action items to be taken for enhancing nuclear safety and ordered the licensee of commercial the nuclear reactors to implement all of them.

Public concern about the safety of domestic NPPs has greatly increased in the Republic of Korea. Therefore, the National Assembly amended the NSA to clearly stipulate the regulatory control on severe accidents. The amended NSA requires the licensee of the commercial nuclear reactor to establish and implement the Accident Management Programmes (AMP) which is to cope with not only design basis accidents, but also severe accidents by June 23rd in 2019. To support the amendment of NSA, subsequent rulemaking was made to amend the Enforcement Decree, the Enforcement Regulations, the Regulation on Technical Standards, and NSSC notices.

3.3.3.1. Accident management

The regulation on Technical Standards [53] specifies the objectives on the assessment of accident management plan capability as follows:

(1) "The accident management plan shall prevent the discharge of large quantities of radioactive materials that may threaten the health of residents in the surrounding areas or cause long-term contamination outside the site in the event that an accident takes place.
(2) It shall minimize the increased rate of risk that the operation of nuclear reactor and related facilities is likely to have on the health and the environment of the residents in the surrounding areas." [53].

There are deterministic and probabilistic methods to achieve the objective. Focusing on the deterministic methods, the NSSC notice no. 2017-34 [51] describes seven threats to the containment integrity which were required to be assessed as following in article 5:

— "Combustible gas burning or explosion
— Containment high temperature or over-pressurization
— Corium and concrete interaction
— High pressure melt ejection
— Containment direct heating
— Fuel coolant interaction
— Containment bypass including steam generation tube rupture"

Article 7 of the NSSC notice no. 2017-34 describes the evaluation of Severe Accident Mitigation Capability as follow;

— “The nuclear power reactor facility shall be capable of preventing the loss of the protective barrier function in its reactor containment buildings to avoid the mass release of radioactive materials due to threat factors that may occur after severe core damage.”

The SRG for the AMP [54] describes the key criteria suited for the nature of seven threats due to severe accident in NSSC notice no. 2017-34 and be summarised as follows;

- Selection of severe accident sequences:
  To ensure that the threat to the containment integrity is mitigated, it should be demonstrated that the containment maintains its integrity under the severe accident challenges. The severe accident challenge should be selected based on engineering judgement and PSA.
- Mitigative measures and strategy:
  Reliable mitigative measures and appropriate strategies should be prepared. Support facilities such as electric power and water source should be prepared to maintain the operation of mitigative measures. The simulation of the mitigative measures should be consistent with the SAMGs.
- Assessment methodology:
  Accident analysis code should have models based on up-to-date technology and knowledge with regard to threats to the containment integrity. The accident analysis should consider phenomena which are expected in the progress of severe accidents. Assumptions and user inputs should be based on reasonable ground. The significant uncertain factors should be identified and their influence on the containment integrity estimated. If the influence is as significant as it can threaten the containment integrity, alternative measures such as design changes should be provided.
- Integrity of containment:
  The containment should maintain its role as a reliable, leak-tight barrier during the entire severe accident progress by ensuring that the containments stresses do not exceed ASME service Level C limits for metal containments, or Factored Load Category for concrete containments. The containment steel liner plate in the reactor cavity should not be damaged.

The criteria were initially developed for LWRs. The specific criteria for heavy water reactor were also added up in the SRG to consider the different design features with regards to mitigative measures and strategy, and integrity of containment.

3.3.4. Regulatory oversight of severe accident management

Regulatory oversight of SAM includes a review of the SAMG, including:

- Plant specific background documents and severe accident management guidance;
- Writers’ guidance;
- Validation programme;
- Training programme;
- Maintenance programme.

As the AMP has been legislated, the SAMG has become part of the AMP and been managed as a licensing document. After receiving approval for the AMP, the adequacy is confirmed on regular inspection and review of periodic safety report.

3.4. ROMANIA

This section provides an overview of the regulatory framework for PHWR severe accidents in Romania.

3.4.1. Regulatory framework

The main Romanian Laws governing the nuclear facilities and activities are:

- Law no. 111/1996 on the Safe Deployment, Regulation, Licensing and Control of Nuclear Activities, republished, with subsequent modifications and completions [55];
- Law no. 703/2001 on the Civil Liability for Nuclear Damage [56];
- The Laws ratifying the Convention on Nuclear Safety and the other major conventions in the nuclear field.

Nuclear activities in Romania, as a member of the European Union, are subject also to the applicable EU legislation.

Romania has an adequate legal infrastructure to fulfil its commitments to all relevant international nuclear safety conventions and obligations. The updating and modernization of the legal and regulatory framework takes into account the development of the international legislation, the current standards, the regulatory and industry feedback, the results of research and development.

Figure 12 shows the regulatory framework in Romania that has been continually developed to reflect the evolution of the international standards and the operating experience available at international level, including lessons learned from major accidents.

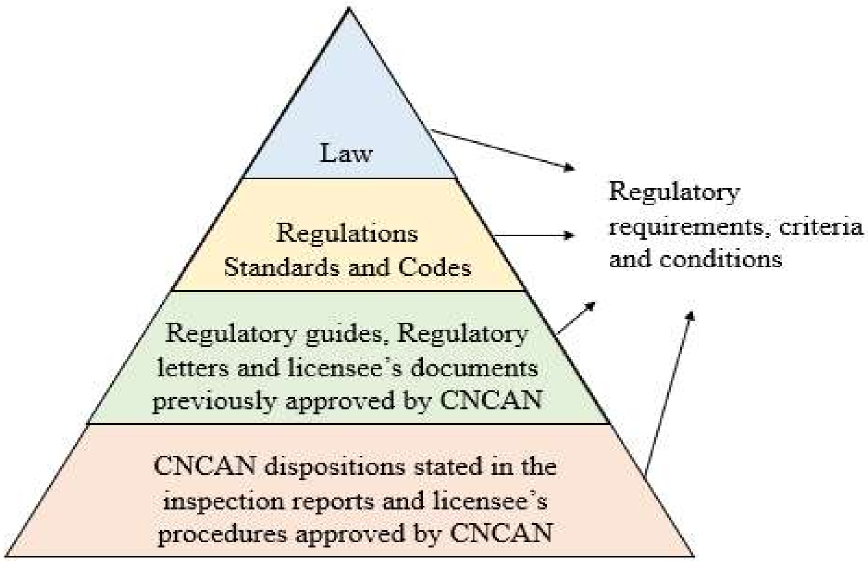


FIG. 12. Regulatory framework in Romania.

3.4.2. Plant states, events and dose criteria

The plant states addressed in the regulatory framework in Romania are in accordance with the Romanian regulation NSN-24 [57] for deterministic safety analyses and reflected in Table 5.

TABLE 5. DEFINITION OF PLANT STATES AND DOSE CRITERIA IN THE REPUBLIC OF KOREA

<table>
<tr><td></td><td colspan="5">Plant Design Envelope</td></tr>
<tr><td rowspan="3">Plant States</td><td colspan="2">Operational States</td><td colspan="3">Accident Conditions</td></tr>
<tr><td rowspan="2">Normal Operation</td><td rowspan="2">Anticipated Operational Occurrences</td><td rowspan="2">Design Basis Accidents</td><td colspan="2">Design Extension Conditions</td></tr>
<tr><td>DEC-A - including combined failures leading to limited fuel failure with core geometry preserved</td><td>DEC-B - including combined failures leading to fuel melt and core geometry loss (severe accidents)</td></tr>
<tr><td>Frequency</td><td>N/A</td><td>(events/reactor year) of > 10^{-2}</td><td>$10^{-5} \leq f < 10^{-2}$</td><td colspan="2">f< 10^{-5}</td></tr>
<tr><td>Acceptance Criteria</td><td>As low as reasonably achievable</td><td>whole-body dose limit of 0.5 mSv</td><td>whole-body dose limit of 20 mSv</td><td colspan="2">Safety goals:
- Dose-frequency criteria from NSN-24 [57];
- CDF< 10^{-4} events/reactor year and LERF< 10^{-5} events/reactor year, for existing NPPs; for new NPPs, these risk metrics are reduced by one order of magnitude.
- New quantitative nuclear safety objectives established in the regulation NSN-21 and regulatory guide GSN-03 [58].</td></tr>
</table>

In accordance with the regulations, the deterministic analyses for Anticipated Operational Occurrences and DBA are presented in Chapter 15 of the Final Safety Analysis Report (FSAR) and the results of the deterministic analyses for DEC, DEC-A and DEC-B are presented in the FSAR, Chapter 19.

When employing updated criteria to categorize design basis events for currently operational plants, which were licensed based on earlier standards, certain events previously classified under the DBA category for CANDU reactors (such as Large LOCA with loss of Emergency Core Cooling) are now included in the category of DECs, as defined by the NSN-24.

The regulatory guide GSN-03 [58], issued to facilitate the comprehension and implementation of the requirements that are provided in the regulation NSN-21, imply the usage of these quantitative nuclear safety objectives:

a) "Frequency of releasing into the environment a quantity of radioactive material that would require the temporary evacuation of the population from the vicinity of the nuclear site, quantified as the sum of the frequencies of all accident sequences with the source term higher than 1000 TBq of Iodine-131, to be less than 10^{-5} / year."
b) "Frequency of releasing into the environment a quantity of radioactive material that would require relocation of the population near the site, quantified as the sum of the frequencies of all accident sequences with the source term higher than 100 TBq of Cesium-137, to be less than 10^{-6} / year."
c) "The cumulative frequency of all accident sequences that can lead to effective doses higher than 100 mSv in the first 7 days, for which the population in the vicinity of the

nuclear facility is required to be evacuated”

d) “The cumulative frequency of all accident sequences that can lead to effective doses higher than 100 mSv in the first year, for which temporary relocation of the population located near the site is required according to the generic criteria of the Regulation of Emergency Situations Management for nuclear or radiologic risk, to be less than 10^{-6} / year.” [58].

Deterministic and probabilistic safety assessments have been carried out and revised by the licensee to further assess the fulfilment of the safety objectives. The assessments cover all NPP operational modes and account for all relevant internal and external initiating events. Both DBAs and DECs, including severe accidents, were considered. Accident scenarios affecting several nuclear installations located on a common site will also be considered.

3.4.3. Requirements for severe accidents

Requirements on severe accident simulation and modelling and on the associated applications, including on design provisions, procedures, facilities and training for severe accident management, are established in the following mandatory regulations:

- NSN-21, Fundamental nuclear safety regulations for nuclear installations [59];
- NSN-22, Regulation on the licensing of the nuclear installations [60];
- NSN-24, Regulation on the deterministic nuclear safety analyses for nuclear installations [57];
- NSN-07 (rev.1), Nuclear safety requirements on the response to transients, accident management and on-site emergency preparedness and response for NPPs [61];
- NSN-23 (rev. 1), Nuclear safety regulation on the training, qualification and authorization of the personnel of organizations operating nuclear installations [62].

Compliance with the provisions of the above-mentioned regulations is mandatory in the licensing process and for the entire duration of the validity of the NPP license. CNCAN regulations are based on the IAEA Safety Standards, including SSG-54 [9].

In addition, CNCAN has included provisions relevant for severe accident analysis and management in the following regulatory guides:

- GSN-03, Guide on fulfilling the overall nuclear safety objective set in the fundamental nuclear safety requirements for nuclear installations [58];
- GSN-04, Guide on the format and content of the Final Safety Analysis Report (FSAR) for nuclear power plants [63].

3.4.3.1. Requirements on severe accident analysis

The regulation NSN-24 contains requirements on the deterministic safety analyses, including severe accident analysis. These requirements include provisions on the numerical and physical models used in the analyses and on the validity, applicability and limitations of the computer codes, taking account of the experimental basis, operating experience and recommendations from the designer of the nuclear installation.

CNCAN has quality management regulations for computer codes for safety analyses since 2003. These are set in the regulation NMC-12 (specific regulatory requirements on the management systems applied to the production and utilization of computer software for research, design, analyses and calculations dedicated to nuclear installations) [64]. These regulations are applicable to the development and utilization of computer codes for all types of deterministic and probabilistic safety analyses. After the severe accident analyses have become part of the licensing basis, CNCAN is verifying compliance with NMC-12 also for the computer codes used for severe accident analysis.

Prior to 2015 and the publication of the GSN-04 regulatory guide, severe accident analyses were submitted as support for the PSA. They are currently part of the licensing basis per the provisions given in the NSN-22 regulation.

3.4.3.2.Requirements on provisions for severe accident management

CNCAN has issued specific requirements on design provisions for severe accident mitigation and on technical and organizational provisions for severe accident management.

Requirements on design provisions for severe accident mitigation are included in the regulations NSN-02 (Nuclear safety requirements on the design and construction of Nuclear Power Plants [65]) and NSN-21 (Fundamental Nuclear Safety Regulations for Nuclear Installations). Requirements on the consideration of severe accident analyses in the site evaluation and in the establishment of the exclusion zone and low population zone are included in the regulation NSN-01 (Nuclear safety requirements on the siting of Nuclear Power Plants [66]).

CNCAN has issued specific requirements for the development, verification and validation of the SAMGs, as part of the NSN-07 regulation. In addition, in accordance with the provisions of the Regulation on the deterministic nuclear safety analyses for nuclear installations (NSN-24), severe accident analyses are required to be used in the development of emergency preparedness plans and procedures and in the associate training.

Another application of severe accident simulation and modelling is related to the capabilities of the full-scope simulators for NPPs, used in the training for control room operators and shift supervisors, addressed by CNCAN as part of the NSN-23 regulation. In accordance with the NSN-23 regulation, the full-scope simulator needs to have the capability, to the extent practicable, to reproduce severe accident scenarios. In order to comply with this requirement, the full-scope simulator for Cernavoda NPP has been recently modernized and upgraded (see Section 2.4.2.1). While severe accident scenarios are not yet part of the practical part of the licensing examinations, CNCAN is considering this option for the future. Currently, the full-scope simulator's extended capabilities are used for supporting operators' training in severe accident management and for making the emergency response exercises more realistic.

3.4.4. Regulatory oversight of provisions for severe accident management

CNCAN has developed internal procedures for review and assessment of the severe accident analyses, as well as procedures for the inspection of their applications.

Review and assessment of severe accident analysis is performed as part of the process for license renewal for the operating reactors and when new or revised analyses are submitted to CNCAN. For new reactors, review and assessment of severe accident analysis will be part of the licensing process from the beginning. Up to date, CNCAN has not developed its own capability to perform safety analyses, simulations and calculations with computer codes for independent regulatory verification of the analyses performed by the licensee, but has taken some steps in this regard, acquiring computer codes and training personnel. Up to present, there are no requirements or arrangements in place independent computer codes for regulatory use. Development of regulatory capability for independent verification of safety analyses using computer codes remains a challenge but is nevertheless a goal to be pursued by CNCAN on the long-term, as part of the strategy for further enhancing regulatory capabilities in the area of nuclear safety.

Periodic regulatory inspections are performed to verify the following:

- The maintenance, updating and revalidation of severe accident management guidelines to account for design modifications, new analyses, new relevant results of research activities, recommendations from the nuclear industry and regulatory bodies from other countries;
- The use of the computer codes for severe accident analysis and the continuous development of licensee's capabilities in this area;
- The performance of the full-scope simulator, including simulating severe accident scenarios and the use of these capabilities in operators' training and in emergency preparedness and response training and exercises.

4. EXPERIMENTAL PROGRAMMES UPDATES

The phenomena associated with a severe core damage progression are highly complex. There is therefore no single experiment that can capture the entire spectrum of possible interaction in the event progression. The key objective of performing such experiments is understanding the interactions of core melt with structures and water, preceded by the release, transport and deposition of fission products carrying vapours and aerosols.

The complex phenomena controlling the progression of the accident in a reactor necessitate the use of computer codes to understand how the reactor and the containment respond to various severe accident conditions. An analyst desires a reliable estimate of the release, controlled by the transport and deposition of fission products carrying vapours and aerosols. Qualified integral codes assess plant-specific severe core damage sequences to quantify the changes to the important plant parameters over time and the timing of significant events.

These simulations are supported by sensitivity studies, expert judgment, separate-effect experiments, and, in some cases, mechanistic code calculations to evaluate the overall uncertainties [67]. The integral codes incorporate models that are compared against data or developed to explain observations from events or tests. The models embedded in integral codes reflect the status of the knowledge. The agreement between the model predictions and the experimental data indicates the degree to which the actual phenomena are understood.

Assessing existing severe accident knowledge and its adequacy is therefore essential to perform consequence analysis of severe accident progression in PHWRs with adequate confidence that the calculated parameters reflect what is likely during the event. The Phenomena Identification and Ranking Table (PIRT) provides a valuable tool for a structured qualitative review of severe accident phenomena, the existing knowledge base, and the level of uncertainty. The main objective of undertaking the PIRT activities is identifying research and development priorities related to severe accidents. Applying a PIRT process can identify phenomena of high safety importance and high uncertainty in the knowledge and therefore deserve further comprehensive analytical and/or experimental studies. As the following sections would attest, quite a few phenomena identified in the PIRTs as having priority research needs have already been investigated, being planned in ongoing research projects or will be studied in the future. The ranking of phenomena reflects the state of knowledge and the perception of their importance at the time of the PIRT activity. With emerging new research results, the ranking of certain phenomena may change.

This chapter presents in Section 4.1 an assessment that relies on systematic PIRT studies completed in Canada, Europe, and India, considering a large pool of experts of diverse backgrounds in severe accident phenomenology using databases of experimental programmes performed in several laboratories. This comprehensive comparison of such a pool of information is valuable for future Research and Development (R&D) activities including experimental programmes among the PHWR MS. The recent updates in those experimental programmes are presented in Section 4.2 for Canada and in Section 4.3 for India.

4.1. SEVERE ACCIDENT PHENOMENA IDENTIFICATION AND RANKING

The PIRT is a widely accepted process for assessing knowledge and its uncertainties. In 1989, as part of the Code Scaling, Applicability and Uncertainty effort [68], the United States Nuclear Regulatory Commission (US NRC) proposed the methodology of PIRT as a rigorous, auditable process. Since then, it has evolved to be a proven formalized decision-making tool that allows the evaluation of a reactor by following the response of a measurable critical parameter, called the "Figure of Merit", chosen by a panel of experts. One of the distinct advantages of the technique is identifying the phenomena's knowledge level, which helps identify additional research and data collection.

The PIRT process generally involves a nine-step process:

- Step 1: define the issue;
- Step 2: define the specific objectives;
- Step 3: obtain database information;
- Step 4: define hardware and scenario;
- Step 5: establish the Figure of Merit;
- Step 6: identify the phenomena;
- Step 7: importance ranking;
- Step 8: knowledge assessment;
- Step 9: documentation.

Further details are available in References [68-69].

4.1.1. Canadian severe accident phenomena identification and ranking

The PIRT assessments for the CANDU 6 and Advanced CANDU Reactor (ACR)-1000 reactors were performed by subdividing the event domain into three convenient sub-scenarios to ensure that the figure-of-merit remains a continuous variable for each of the sub-scenarios [70]. Rather than subdividing the accident scenario into several contiguous time intervals or accident phases, the PIRT panel treated the three sub-scenarios separately with their figure of merit. The Table 6 provides the sub-scenarios.

TABLE 6. SUB-SCENARIO, FIGURE-OF-MERIT AND THE TIME INTERVAL USED IN THE CANDU 6 AND ACR-1000 PIRTS

#	Sub-scenario	Figure of merit	Time interval
1	CV failure	Heat transfer to the calandria vault and end shield water	0 s to CV failure time
2	Containment failure	Containment pressure	0 s to containment failure time
3	Long-term response	Source term (at the containment boundary)	0 s to beyond containment failure time

4.1.1.1. CANDU 6 severe accident PIRT

The CANDU 6 PIRT used the SBO sequence to track and identify essential phenomena and their knowledge level [70]. An SBO was selected because, based on risk studies, the estimated core melt frequencies from station blackouts are a significant risk contributor. Also, the SBO event sequence starts from an intact and coolable core that progressively degrades to a noncoolable state providing insights into most mechanistic challenges expected in most BDBA sequences in a reactor.

The PIRT identified and ranked 283 phenomena relevant to the PHWR SBO sequence. Table 7 lists the number of phenomena ranked as high (H), medium (M), low (L), and insignificant (I) in the right-hand columns, whereas the importance of knowledge level as fully known with small uncertainty (4); known with moderate uncertainty (3); partially known with large uncertainty (2); and very limited knowledge with an uncertainty that cannot be characterized (1) in the left-most column. The expert panel considered that any phenomenon with high or medium importance to severe accidents have a knowledge level equivalent to “known with moderate uncertainty” or higher. The phenomena that did not meet this acceptance level will require additional R&D to refine the existing knowledge to improve the understanding of severe accident behaviour.

After the PIRT exercise, there were 134 high-importance and 52 medium-importance phenomena. Out of the 134 high-importance phenomena, the following three phenomena, were assessed as having very limited knowledge with the uncertainty that cannot be characterized:

— Cracking of crust (chemical interaction) in the terminal debris bed;
— Gap formation between the crust and the CV;
— Coolability of corium in the basement (integral effect).

TABLE 7. COMPARISON OF SEQUENCE OF SIGNIFICANT EVENTS IN A STATION BLACKOUT EVENT FOR THE CANDU 6

CANDU 6 PIRT				
Knowledge adequacy	**Phenomena Rank**			
	H	**M**	**L**	**I**
(4) Fully known, small uncertainty	41	16	19	*
(3) Known, moderate uncertainty	55	27	31	*
(2) Partially known, large uncertainty	35	9	2	*
(1) Very limited knowledge, uncertainty cannot be characterized	3	0	1	*
Total	134	52	53	44

* Knowledge level not assessed

4.1.1.2. ACR-1000 severe accident PIRT

The ACR-1000 PIRT process identified 347 phenomena; among them, six had high importance, and one had medium importance with very limited knowledge where the uncertainty could not be characterized [70] (see Table 8). Also, compared with the CANDU 6 PIRT, the ACR-1000 PIRT elicited 17 new high importance phenomena with partially known or very limited knowledge. Some of these new phenomena also apply to the CANDU 6 and are not due to different engineered features of the ACR-1000. The new phenomena were mainly due to different levels of expertise brought to the table during expert elicitation.

TABLE 8. COMPARISON OF SEQUENCE OF SIGNIFICANT EVENTS IN A STATION BLACKOUT EVENT FOR THE ACR-1000

ACR-1000 PIRT				
Knowledge adequacy	**Phenomena Rank**			
	H	**M**	**L**	**I**
(4) Fully known, small uncertainty	44	16	20	*
(3) Known, moderate uncertainty	50	26	33	*
(2) Partially known, large uncertainty	44	10	3	*
(1) Very limited knowledge, uncertainty cannot be characterized	6	1	1	*
Total	145	53	57	92

* Knowledge level not assessed

The new phenomena identified in the ACR-1000 PIRT were:

- CV critical heat flux;
- The lower zone control thimble corium and debris interaction with nozzle material;
- Critical heat flux on the thimbles;
- Load, temperature, and stress distribution on the thimble;
- The melting, solidification of the terminal debris & corium;
- Natural circulation in the melt;
- Calandria Tube (CT) creep rupture;
- Pressure Tube (PT) pipe whip;
- The core concrete interaction in Reactor Building (RB) basement;
- Interaction with protective layer of the basement;
- Coolability of the Corium on the RB floor;
- Fuel Coolant Interaction (FCI) on the RB liner and in the calandria vault;
- Fission product settling in RB liner;
- Aerosol generation and corium interaction with the refractive protective layer;
- Condensation and heat transfer in the presence of non-condensable gases.

Some high-ranking phenomena identified in the CANDU 6 PIRT were not identified in the ACR-1000 PIRT:

- End fitting ejection (due to PT guillotine failure);
- Perimeter wall fission product settling;
- Upper dome and ring beam fission product settling;
- Fuel-coolant interaction in the calandria vault water;
- Debris and corium interaction with nozzle materials and load distribution / thermal stress distribution in inlet and outlet nozzles.

4.1.2. EURSAFE severe accident phenomena identification and ranking

The European Commission, through a programme called EURSAFE (European expert network for the reduction of uncertainties in severe accident safety issues) [71], conducted a PIRT. The objective was to address uncertainties through appropriate R&D programmes, making the best use of European resources.

This PIRT considered a severe accident situation in LWRs, integrating severe accident issues from core uncovery and onset of debris formation to long-term corium stabilisation, long-term containment integrity, and fission product retention and fission product release to the environment. The final ranking considered both safety and knowledge aspects.

Following the ranking, the PIRT determined 916 severe accident phenomena, with 229 as safety significant. The knowledge ranking indicated 106 safety-significant phenomena with low knowledge. The safety significant phenomena with low knowledge were further regrouped into a limited number of research items according to similarities in their physical processes to set up coherent R&D programmes. The list summarising the outcome of regrouping is provided by [71], and a comparison with the CANDU 6 ACR-1000 and Indian PIRTs is provided in Section 4.1.4.

4.1.3. Indian severe accident phenomena identification and ranking

A Task Force of AERB developed a PIRT in 2013. The PIRT was generated to cover the phenomena that can be encountered during progression of a postulated severe accident in the Indian 220 and 540 MWe PHWRs. The PIRT lists both LCDA and SCDA phenomena. The information available in open literature was reviewed to understand the physics of the phenomena and establish a physical model along with their validation. The knowledge level was named “Information Availability” and importance of phenomena was named “Impact on Severe Accident Progression”. The importance was ranked in a scale of 1 to 5. The rankings were obtained using an expert elicitation procedure. The knowledge level and importance are reconciled to group them to appropriate categories to enable a comparison among the PIRTs conducted for CANDU 6, ACR-100 and EURSAFE. The grouping is provided in Table 9. The three phenomena with ranking of H1:

- Integrity criteria of CV under debris impact;
- Molten Fuel Moderator Interaction (MFMI) under SCDA conditions;
- Material properties of corium-concrete (haematite based) mixture encountered during MCCI.

TABLE 9. COMPARISON OF KNOWLEDGE AND IMPORTANCE CATEGORISED IN THE INDIAN PIRT

Indian PIRT				
Knowledge adequacy	**Phenomena Rank**			
	H	**M**	**L**	**I**
(4) Fully known, small uncertainty	14	8	1	*
(3) Known, moderate uncertainty	5	7	2	*
(2) Partially known, large uncertainty	11	0	1	*
(1) Very limited knowledge, uncertainty cannot be characterized	3	1	0	*
Total	33	16	4	*

* Knowledge level not assessed

4.1.4. A combined overview of CANDU 6, ACR-1000, EURSAFE, and Indian phenomena identification and rankings

A combined overview of the four PIRTs may provide additional details of critical phenomena relevant to PHWRs. They all followed different approaches in identifying, prioritizing, and listing the phenomena and their knowledge using a diverse group of experts. They, therefore, can have contextual differences when combined judgments from different groups of experts. Despite these difficulties, comparing the PIRTs can be a valuable addition to supporting decision making on research and development activities.

Table 10 provides a tabular comparison of the four PIRTs. The table identifies the importance and knowledge ranking together. For example, the symbol "H2" indicates that the importance is "high (H)", and the knowledge level is "Partially known; large uncertainty (2)". The table excludes some of the phenomena identified in the EURSAFE PIRT because they are irrelevant to PHWR scenarios. For example, the core catchers and direct containment heating are irrelevant to PHWR reactors. In PHWRs, the severe accident entry condition depends on the dry out of fuel channels and channel disassembly, and therefore the chemistry and the integrity of the reactor coolant system become immaterial if they are captured under various core degradation states.

In developing the comparison, the decimal ranking method used by the European and Indian PIRTs was converted to the integer ranking used in CANDU 6 and ACR-1000 to maintain consistency in the discussion. Hence, the "Ranking" column shows the importance and knowledge ranking.

TABLE 10. COMBINED RANKING OF CANDU 6, ACR-1000, EUROSAFE AND INDIAN PIRTS

System	Component	Phenomenon	Ranking			
			CANDU 6	ACR-1000	EURSAFE	Indian
CV & end shields	CV	Failure of the vessel	H2	H2		
		(i) Critical heat flux;	None	H2	H1	H2
		(ii) Integrity criteria-debris impact;				H1
		(iii) Integrity criteria: local creep;				H2
		Load distribution / thermal stress distribution	H2	H2		
		Localized melting & failure	H2	H2		
		Molten pool-vessel thermal contact conductance				H2
	End shields	Critical heat flux	H2	H2		
		Failure of lattice tubes	H2	H2		
		Inner tube sheet failure	H2	H2		
		Load distribution / thermal stress distribution	H2	H2		
		Outer tube sheet failure	H2	H2		
	Inlet & outlet nozzles	Debris and corium interaction with nozzle materials	H2	None		
		Load distribution / thermal stress distribution	H2	None		
	Lower zone control thimbles	Corium and debris interaction with nozzle materials	None	H2		
		Critical heat flux	None	H2		
		Load distribution / temperature distribution / thermal stress distribution	None	H2		
Core	Suspended debris	Channel breakup	H2	H2		
		Coolability	H2	H2		
		Inter-channel contact heat transfer	H2	H2		
		Melt relocation	H2	H2		
		Hydrogen generation during reflood or melt relocation into water			H1	
		LCDA event (molten fuel moderator/coolant interaction)				H2
	Terminal debris & corium	Coolability	H2	H2	H2, M1	
		SCDA event (molten fuel moderator/coolant interaction[4])				H1
		Core collapse	H2	H2		
		Corium CV wall interaction	H2	H2		
		Cracking of crust (chemical interaction)	H1	H1		
		Crust formation	H2	H2		
		Debris porosity and pore size distribution	H2	H2		
		Oxidizing environment impact on source term			H1	
		Fuel coolant interaction (steam explosion – shock loading)	H2	H2	M2	
		Fission product release	H2	H2		H4
		Focusing effect (chemical interaction)	H2	H2		
		Gap formation (crust/CV)	H1	H1		

[4] The fundamental phenomena are predominantly corium coolability, for both energetic and non-energetic molten-fuel moderator interaction.

TABLE 10. COMBINED RANKING OF CANDU 6, ACR-1000, EUROSAFE AND INDIAN PIRTS (CONT.)

System	Component	Phenomenon	Ranking			
			CANDU 6	ACR-1000	EURSAFE	Indian
		Heat transfer (convection by natural circulation, nucleate & film boiling, conduction, thermal radiation) / model for molten pool heat transfer	H2	H2		H4
		Melt relocation	H2	H2	H1	
		Melt stratification / model for molten material stratification and fission product partitioning	H2	H2	H1	H2
		Melting and solidification		H2	M1	
		Natural circulation in the melt		H2		
Fuel channel assembly	End fittings, shield plugs and feeder connections	End fitting ejection (PT guillotine failure)	H2			
	CT	Creep rupture (may have impact on CV failure)		H2		
		Properties of CT of Zircaloy2 at high temperature				H2
	PT	Pipe whip		H2		
	Bundle	Fuel Bundle Integrity (LCDA)				H2
	Channel integrity	Exposed channels heat up behaviour (SCDA)				H2
Reactor building	Corium (in basement)	Coolability (integral effect)	H1	H1	M1	
		Fuel coolant interaction	H2	H2	M2	
		Core concrete interaction		H2		H2
		Material properties of corium-concrete (hematite)				H1
		Interaction with protective layer		H2		
	Corium (on the RB floor outside of the vault)	Coolability (integral effect)		H1		
		Fuel coolant interaction		H2	M2	
	Inner dome (dousing tank)	Fission product settling (3 processes)	H2	H2	H1	
	Perimeter wall	Fission product settling (3 processes)	H2		H1	
		Containment atmosphere mixing and hydrogen combustion/detonation			H2	H4
		Hydrogen source term				H2
		Dynamic and static behaviour of containment, crack formation and leakage at penetrations			M2	
		Containment chemistry impact on source term			H1	
	Calandria vault (structure)	Core concrete interaction (local wall)	H2	H2		
	Reactor building liner	Fission product settling (3 processes)		H2	H1	
	Upper dome & ring beam	Fission product settling (3 processes)	H2		H1	
		FP transport through containment concrete cracks				H2

TABLE 10. COMBINED RANKING OF CANDU 6, ACR-1000, EUROSAFE AND INDIAN PIRTS (CONT.)

System	Component	Phenomenon	Ranking			
			CANDU 6	ACR-1000	EURSAFE	Indian
Reactor building (other)	Internal walls	Corium concrete interaction	H2	H2		
Calandria vault	Walls	Corium concrete interaction	H2	H2	H1	
	Water	Fuel coolant interaction	H2		M2	
	Cooling system	Fuel coolant interaction		H1	M2	
		Ex-vessel corium coolability			H2	
	Floor protective layer	Spalling and cracking (thermal stress)		H1		
		Aerosol generation		H2		
		Corium interaction		H2	H2	
Shield Cooling System	Shield cooling circuit	Load distribution (failure)	H2	H2		
Heating, cooling, ventilation common processes & services	Local air cooler	Condensation and heat transfer in the presence of non-condensable gases	None	H2		

The CANDU, Indian and European PIRTs generally agree on the high importance rankings given to critical heat flux outside the vessel, terminal debris, and corium coolability, melt relocation, melt stratification, fission product settling processes, and corium interaction with protective layers. The knowledge level assessment, however, differed for a limited number of phenomena ranked as high importance. The CV Critical Heat Flux (CHF) was given H2 in the ACR-1000 PIRT. In contrast, EU PIRT gave the vessel CHF an H1 rank because perhaps flooding the external walls of the Pressurized Water Reactor (PWR) vessel is only undertaken in a limited number of reactors (Loviisa NPP in Finland [72]) and hence was declared low in knowledge with H1. In the component described as corium (in the basement), the Canadian PIRTs considered the phenomenon of coolability (integral effect) as H1, whereas the EU PIRT ranked it as M1. Likewise, FCI was ranked H2 in the Canadian PIRTs. However, EU PIRT ranked it as M2. Core concrete interaction in the ACR-1000 PIRT and the Indian PIRT was ranked H2. Since the concrete in India is mainly hematite-based, they included the material properties of corium-concrete mixture (hematite) as a phenomenon with H1 rank. The corium interaction with a protective layer (Urania) was considered in the ACR-100 PIRT as a separate phenomenon and ranked as H2. For three phenomena, namely FCI (steam explosion – shock loading), corium melting and solidification, and corium coolability (integral effect), the CANDU PIRTs ranked the phenomena as high in importance; however, the European PIRT ranked the phenomena as medium in importance.

Within the terminal debris and corium component group, the EU PIRT, assessed the oxidizing environment impact on source term and FCI (steam explosion – shock loading) phenomena as H1 and M2, respectively. The latter phenomenon was assessed by Canadian PIRTs as H2 (for both C6 and ACR-1000). In the EU PIRT, hydrogen generation during reflood or melt relocation into water received H1 rank, along with containment atmosphere mixing and hydrogen combustion/detonation receiving H2 rank. In the Indian PIRT, the latter phenomena received the H4 rank, and the hydrogen source term received the H2 rank. In the Canadian

PIRTs, the hydrogen source terms were not considered as a separate phenomenon since it was considered under various other phenomena like the localized melting & failure, debris and corium interaction with nozzle materials, corium and debris interaction with nozzle materials, channel breakup, coolability, and melt relocation.

In the components of terminal debris and corium, the oxidizing environment impact on source term was considered a rank of H1 in EU PIRT, whereas FCI (steam explosion – shock loading) was ranked H2 for both C6 and ACR-1000 reactors; however, the EU ranking was M2. During the Canadian PIRTs, the molten-fuel coolant interaction experiments were incomplete and therefore received a lower knowledge ranking. The knowledge level has increased since completing these experiments successfully, and the importance has also reduced compared to other phenomena [73]. The results indicate that the molten-fuel coolant interaction is unlikely to occur if the melt ejection pressure is near the system pressure. The phenomenon of fission product release was ranked H2 for the two Canadian PIRTs and H4 in the Indian PIRT. Only the Canadian PIRT considered the focusing effect (chemical interaction) and gap formation (crust/CV) and ranked them as H2 and H1, respectively. The heat transfer (convection by natural circulation, nucleate & film boiling, conduction, thermal radiation) model for molten pool heat transfer received an H2 rank in the Canadian PIRTs and a rank of H4 in the Indian PIRT. The melt relocation phenomenon received H2 in the Canadian PIRTs, while the EU PIRT assigned a rank of H1. The phenomena of melt stratification model for molten material stratification and fission product partitioning were assessed by all four PIRTs and ranked H2 (Canadian), H1 (EU) and H2 (Indian). Melting and solidification were ranked H2 in the ACR-1000 PIRT. However, the EU ranked it as M1. The Canadian PIRT also considered natural circulation in the melt as a phenomenon at the H2 rank.

In the component of the perimeter wall, Canadian and EU PIRTs identified the phenomenon of fission product settling (3 processes) with H2 and H1 ranks, respectively. The phenomenon of containment atmosphere mixing and hydrogen combustion/detonation was identified by the EU and Indian PIRTs with H2 and H4 ranking, respectively. The Indian PIRT identified the hydrogen source term as a phenomenon with H2 rank. The EU PIRT identified dynamic and static behaviour of containment, crack formation and leakage at penetrations with M2 ranking and containment chemistry impact on source term with H1 ranking.

The differences in the ranking perhaps reflect the strength of the thick pressure vessel in LWRs compared with the CV in CANDUs. While there were significant similarities in identifying and ranking the importance and knowledge levels of the phenomena, the phenomena related to high-temperature chemistry and source term estimation using iodine species were not included in the CANDU PIRTs. A possible reason for this difference is that the large quantity of water in the CV is considered to act as a scrubber for fission products and iodine in the early phases of the severe accident. The comparison between ACR-1000 and CANDU 6 PIRTs shows that the differences are mainly due to additional features incorporated in the ACR-1000 reactor. One such difference was the refractory protective layer on the basemat in ACR-1000, which increased the number of phenomena related to molten core-concrete interaction and melt coolability. Another difference was the critical heat flux at penetrations at the bottom of the CV.

The Indian and Canadian PIRTs agree, in essence, on the high importance and their knowledge level rankings given to the vessel's failure, including the vessel's critical heat flux; however, the Indian PIRT presents specifically the integrity criteria of debris impact as H1. The knowledge level assessment for the fission product release differs between the two PIRTs since the Indian PIRT considers this phenomenon fully known with small uncertainty while the Canadian PIRTs consider them to have partially known knowledge with large uncertainty. In addition, the Indian PIRT evaluates as H1 the material properties of corium-concrete, not presented in the CANDU PIRTs. The Indian PIRT ranked the integrity criteria of debris impact and local creep as H1 and H2 ratings, respectively. In the Canadian PIRTs, the phenomena like generic failure of the vessel, load distribution/thermal stress distribution, and localized melting and failure encapsulate the same concept with H2 classification. The molten fuel moderator/coolant interaction phenomena (LCDA event) that received an H2 ranking in the Indian PIRT was not considered in the Canadian PIRT as a phenomenon since, in the Canadian context, it is part of the DBAs. In the fuel channel assembly system, the Indian PIRT considered the properties of CT with Zircaloy-2 at high temperatures; the fuel bundle integrity in LCDAs; and exposed channel heat-up behaviour during SCDAs as H2. These phenomena were not assessed in the Canadian PIRTs.

The Indian PIRT accounts for both LCDA and SCDA phenomena, while the Canadian PIRT considers only severe accidents or events with significant core damage. Consequently, the Indian PIRT identifies high-importance level phenomena with partial knowledge (H2) associated with LCDAs which are not in the Canadian PIRT. The LCDA phenomenon of molten fuel moderator/coolant interaction in the Indian PIRT is somewhat similar to "melt relocation" in the Canadian PIRT with the same ranking. The phenomenon of exposed fuel channel heat up behaviour under SCDA in the Indian PIRT with an H2 ranking is somewhat like the suspended debris coolability in the Canadian PIRT that received an H2 ranking. With these close similarities between the LCDA phenomena and the severe accident phenomena, we can conclude that the LCDA ranking and experiments from the Canadian and Indian researchers have contributed to the international severe accident knowledge pool.

4.1.5. Summary

This section has presented the PIRTs performed by four expert panels for four different reactor designs. Generally, the PIRTs agree on the high importance rankings of critical heat flux outside the vessel, terminal debris, corium coolability, melt relocation, melt stratification, fission product settling processes, and corium interaction with protective layers. However, the knowledge level assessment differed for a limited number of phenomena ranked as high importance. These differences arise due to variations in designs and features, the methodology used in the evaluation, and the definition of the phenomena.

The PIRTs identified phenomena of high safety importance and high knowledge uncertainty that needs additional analytical and/or experimental studies, and therefore serve as a basis for establishing research priorities and directions in experimental and analytical programmes. Most of the high-priority research phenomena identified are either planned to be investigated or have already been studied. The discussions in Section 4.2 for Canada and Section 4.3 for India describe how research activities attempt to fill the knowledge gaps. With these emerging new research results, the ranking of certain phenomena may change.

4.2. EXPERIMENTAL PROGRAMME UPDATES IN CANADA

The section provides updates in experimental programmes related to severe accidents in PHWRs carried-out in Canada.

4.2.1. Experimental facilities

Appendix II of IAEA-TECDOC-1594 [1] contains brief descriptions of the main experimental facilities in Canada, at Canadian National Laboratories (CNL) and Whiteshell Laboratories (WL). Brief status updates for these are included in Table 11, and additional experimental facilities are presented in Table 12. Finally, Table 13 links these facilities to the phenomena identified to the PIRT performed for the CANDU 6 in 2008 [70].

TABLE 11. STATUS UPDATES ON MAIN CANADIN EXPERIMENTAL FACILITIES DESCRINED IN APPEDIX I OF IAEA-TECDOC -1594

Location	Facility	Description	Status
WL	RD-14 M loop	11 MW, full height, scaled representation of a PHWR primary heat transport system. Used for experiments investigating phenomena during upset conditions, including loss of coolant accidents.	Decommissioned, before or in association with the broader decommissioning of WL facilities.
	Large scale vented combustion facility	120 m^3, heated volume designed to withstand pressure pulses and elevated static pressures. Designed to quantify the effects of thermodynamic and geometric parameters affecting pressure development during vented combustion in post-accident containment atmospheres.	
	Large scale gas-mixing facility	1000 m^3 room equipped for injection of steam, air, and helium (as a hydrogen simulant). The facility was used to study mixing, condensation, and geometric effects. It has been replaced by the large-scale containment facility at CNL.	
	Radioiodine test facility	Intermediate scale facility used to simulate the chemical system expected in containment following an accident.	
CNL	High temperature heat transfer laboratory	Flexible facility capable of supporting a range of tests involving high temperatures (up to 1700 °C) and high pressures (up to 10 MPa). Has been used to house test rigs investigating various thermal-chemical-mechanical phenomena.	Available.
	Molten fuel moderator interaction (MFMI) facility	Designed to investigate the potential for steam explosions following high pressure (10 MPa) ejection of up to 25 kg of corium melt (at ~2400 °C) into the moderator. Recent updates to the facility have been made to the instrumentation and control systems of the facility, increasing operational flexibility.	Available. Following recent system updates, the MFMI facility has been renamed the molten material laboratory.
	Core disassembly facility	Designed to investigate the sagging and disassembly of fuel channels as they are uncovered by a progressively depleting moderator. Up to three fuel channels were represented at 1/5 linear scale, with individual tungsten heaters representing fuel bundles.	Decommissioned.
	Blowdown test facility	Used for ‘in-pile’ tests investigating the behaviour of fuel and fission products during loss of coolant events. Test assemblies were positioned in the reactor core, in a thick-walled stainless steel PT. Coolant was voided through an instrumented blowdown line (measuring coolant thermalhydraulic parameters and fission product gamma emission).	Decommissioned, following the broader decommissioning of the National Research Universal (NRU) reactor.

TABLE 12. EXPERIMENTAL FACILITIES AT CNL NOT DESCRIBED IN IAEA-TECDOC-1594

Facility	Description	Status
Large scale containment facility	Facility designed for the study of buoyancy-driven gas mixing with steam condensation, and the behaviour of wet aerosols (agglomeration, settling, impingement and leakage out of containment). Experiments are used for multiple purposes, including the validation of containment thermalhydraulics codes (e.g., GOTHIC), decontamination by pool scrubbing, and airborne transmission of viruses by aerosols. Facility characteristics include: — Total volume ~1575 m^3, (height of 10 m); — Instrumentation to gauge temperatures (gas, wall surfaces, temperature profile through walls), heat fluxes to walls, helium concentration, gas velocity, relative humidity, aerosol velocity, condensation rates and aerosol (droplet) size distribution; — The main section of the facility (~1300 m^3) can withstand 100% relative humidity and a maximum temperature of 65 °C. A smaller section of the facility (~275m^3) can withstand 100% relative humidity and a maximum temperature of 95 °C.	Available
Corium convection apparatus	Representation of a partially molten corium pool in the bottom of a CV during in-vessel melt retention (Figure 14). Key characteristics: — 1:5 linear scale, stepped CV (lower portion only); — Boundary condition provided by a water bath; — Corium simulated using a blend of KNO3 and NaNO3. The resulting Prandtl number varies is ~17 (relative reactor prototypical values in the range of 0.18 to 0.8); — Heating using a grid of heaters submerged in the simulant pool, achieving a maximum internal Rayleigh number of ~2.7 x 10^{11} (relative to reactor-prototypic values in the range of 10^{12} to 10^{13}); — Temperature and heat flux measurements using extensive arrays of thermocouples in the melt, in the near-wall solid crusts, and in the CV walls; The apparatus can be run in one of two configurations, impacting instrumentation and the top boundary condition of the corium simulant pool: — An argon cover gas in contact with the top of the corium simulant pool. In this case, an array of probes can be lowered into the pool to perform online mechanical crust thickness measurements. — An actively cooled plate in direct contact with the top of the corium simulant pool. In this case, mechanical probe measurements are not possible, but temperature and heat flux measurements are performed using arrays of thermocouples positioned in the crust region, and the top cooling plate wall.	Available
End shield CHF rigs	A series of experimental rigs simulating a calandria tubesheet abutting a packed bed of shielding balls (Figure 16). The calandria tubesheet is variously represented using: — A thin, resistively heated stainless-steel plates with heights varying from 0.1 m to 0.6 m (the full height of anticipated contact between the corium pool and the calandria tubesheet). Heat flux is inferred by electrical measurements; — A thick block of copper or stainless steel with embedded cartridge heaters, with a height of 0.6 m. Temperature distribution and heat flux are measured through arrays of embedded thermocouples.	Available
CV CHF rigs	A series of experimental rigs simulating the lower portions of CV (Figure 15), using: — Thick copper blocks heated by embedded cartridge heaters. Temperature distribution and heat flux are measured through arrays of embedded thermocouples; — A thick stainless-steel block heated by embedded cartridge heaters. Temperature distribution and heat flux are measured through arrays of embedded thermocouples; — Thin, resistively heated stainless-steel plates with spatially varying thickness. Heat flux is inferred from electrical measurements, and temperature is measured by probes positioned on the dry surface of the plate.	Available, with the exception of the resistively heated apparatus scheduled to be commissioned in 2024.

TABLE 12. EXPERIMENTAL FACILITIES AT CNL NOT DESCRIBED IN IAEA-TECDOC-1594 (CONT.)

Facility	Description	Status
	These are designed to represent various geometric locations, including: — The bottom of the CV main shell; — The bottom of the CV sub shell; — The annular plate joining the main shell and sub shell; — Surface irregularities representing the CV drainline, manhole, and moderator outlets.	
CT rolled joint bending moment apparatus	Apparatus capable of exerting combinations of transverse loads / bending moments and axial tension on full diameter (reduced length) CTs rolled into plates representing calandria tubesheets (referred to as 'spools'). Tests can be performed with spools at temperatures ranging up to 185 °C. Stress and strain are measured, with parallel digital image correlation of the spools (Figure 13).	Available
CV heat stress response apparatus	Apparatus representing a 1:8 linear scale CV containing radiant heaters positioned near the bottom of the vessel (Figure 17). The heaters are capable of delivering heat scaled to conserve the heat flux exiting the vessel, relative to the reactor case during in-vessel melt retention. CV wall temperatures are monitored using exposed-junction thermocouples spot welded on the inner and outer surfaces. The CV is initially surrounded by water, which is subsequently lowered at a controlled rate, simulating an accident scenario in which make-up water is not provided to the calandria vault / shield tank. Vessel deformation and integrity is monitored visually during the test and assessed via post-test examination.	Available
Hydrogen safety test facility	A spherical pressure vessel with an internal volume of ~250 L, having the following characteristics: — Maximum pressure 1.72 MPa at 100 °C; — Systems for controlled addition and measurement of H_2, D_2, CO, air, steam and inert gases; — Vacuum and vent systems connected for purging; — Mixing fan; — Spark igniter for combustion experiments; — Silicone heaters and insulation for operation up to 120 °C; — Fast data acquisition (10,000 Hz) for combustion experiments: transient pressure transducers and thermocouples; — Slow data acquisition (1 Hz) for hydrogen passive autocatalytic recombiner (PAR) experiments: static pressure transducers, thermocouples, relative humidity probe; — Optical measurement capabilities: short and long wave infrared cameras for visualization of combustion and PAR operation, linear-type Schlieren imaging for optical analysis of combustion process.	Available
Cold crucible induction facility (CCIF)	The CCIF (Figure 19) is used for the study of corium properties (phase behaviour / miscibility in particular). Re-solidified material produced by the apparatus is also used for testing of mechanical properties of corium crusts. The facility is designed for charges with a mass of ~2 kg, containing depleted or natural uranium, along with an array of additional corium components. Optimization of the facility is being performed to increase the maximum achievable temperature; however, charges can be heated to ~2,300 °C (near the anticipated maximum temperature of corium in a CANDU reactor).	Available
CV blowdown apparatus	Apparatus designed to simulate the ejection of water (liquid and vapour) through the CV relief ducts following rupture disc burst (Figure 18). The apparatus is ~1:6 linear scale, with a CV geometry (stepped cylinder), and inserts representing fuel channels in the upper half of the vessel. Water can be heated to saturation temperature at pressures up to ~150 kPa(g), before being relieved through fast-acting valves representing rupture discs. Water and steam are expelled into a condenser designed to present very low flow resistance.	Forthcoming – apparatus assembly and component testing underway, with formal commissioning scheduled for 2024

TABLE 13. MAPPING OF CANADIAN EXPERIMENTAL FACILITIES TO PHENOMENA AND PIRT RANKINGS

Facility	**Phenomena Studied**	**2008 CANDU-6 PIRT Rankings**[5]
RD-14 M loop	This facility could be used to investigate a wide array of primary heat transfer system phenomena, not specified here in the interest of brevity.	Not listed here.
Large scale vented combustion facility	No phenomena studied in this facility relevant to CV failure were identified in the PIRT.	N/A
Large scale gas-mixing facility	No phenomena studied in this facility relevant to CV failure were identified in the PIRT.	N/A
Radioiodine test facility	No phenomena studied in this facility relevant to CV failure were identified in the PIRT.	N/A
High temperature heat transfer laboratory	This facility is capable of hosting investigations of a very wide array of phenomena. In the interest of brevity, these are not listed explicitly.	Not listed here.
Molten fuel moderator interaction (MFMI) facility	Fuel coolant interaction (steam explosion / shock loading)	H2
Core disassembly facility	(i) CT perforation from localized strain; (ii) Core collapse.	i. M3 ii. H2
Blowdown test facility	This facility was capable of investigating numerous phenomena falling under the components ‘fuel bundle’, ‘fission products’, and ‘PHTS main circuit’ in the PIRT. In the interest of brevity, these are not listed explicitly.	Not listed here.
Large scale containment facility	No phenomena studied in this facility relevant to CV failure were identified in the PIRT.	N/A
Corium convection apparatus	(i) Heat transfer (convection, conduction, nucleate and film boiling, thermal radiation); (ii) Melting and solidification; (iii) Natural circulation in the melt.	i. H2 ii. H3 iii. H4
End shield CHF rigs	Critical heat flux (end shields)	H2
CV CHF rigs	Critical heat flux (CV)	H4
CT rolled joint bending moment apparatus	Core collapse	H2
CV heat stress response apparatus	(i) Failure of vessel; (ii) Heat transfer (convection, conduction, nucleate and film boiling, thermal radiation); (iii) Load distribution/thermal stress distribution.	i. H2 ii. H4 iii. H2
Hydrogen safety test facility	No phenomena studied in this facility relevant to CV failure were identified in the PIRT.	N/A
Cold crucible induction facility (CCIF)	(i) Melt stratification; (ii) Corium CV wall interaction (chemical interaction).	i. H2 ii. H2
CV blowdown apparatus	(i) Boil off of moderator; (ii) Moderator expulsion rates (periodic choked flow, two phase flow).	i. H4 ii. H3

[5] Note that the rankings are those for the PIRT sub-scenario of CV failure. This limitation is applied to limit the table’s extent.

4.2.2. Key findings of experimental programmes

This subsection summarizes key findings of the experimental programmes performed since the 2008 publication of IAEA-TECDOC-1594 [1].

4.2.2.1. Core collapse

A series of destructive tests were performed with prototypic (partial-length) CTs rolled into a representative calandria tubesheet (cumulatively referred to as 'spools'), using CNL's CT rolled joint bending moment apparatus (see Figure 13) [74]. Combinations of bending moment (transverse loading) and axial tension were applied, with the spools held at temperatures up to 185°C. A key observation was that the spools consistently failed by buckling near the belled region of the CT. Complementary finite element simulations were performed and were found to agree well with the experimental results; although a consistent bias was observed, hypothesized to be a result of misrepresentation of the anisotropic material properties of the CT [75]. The results of this effort are discussed further in Section 6.

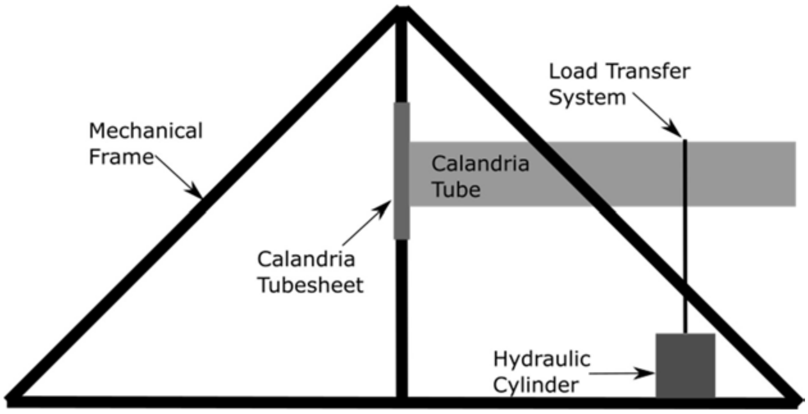


FIG. 13. Calandria tube rolled joint bending moment apparatus schematic (courtesy of J. Spencer, CNL).

These efforts addressed the phenomena of core collapse (ranked H2 in the CANDU 6 and ACR-1000 PIRTs), and of channel breakup (ranked H2 in the CANDU 6 and ACR-1000 PIRTs).

4.2.2.2. Heat flux exiting the CV during in-vessel melt retention

Two experimental campaigns have been performed using CNL's corium convection apparatus (see Figure 14), measuring temperature fields, exiting heat flux distribution, and crust thickness under a range of conditions. Quantitatively, correlation of the internal Rayleigh number with Nusselt number were relatively consistent with existing correlations. Qualitatively, the system's behaviour was consistent with results of existing models, with exiting heat flux lowest and crust thickest near the bottom of the CV. Efforts to extend the results to the reactor case have begun through benchmarking of CFD models against the experiment [76]. The results of this effort are discussed further in Section 6.

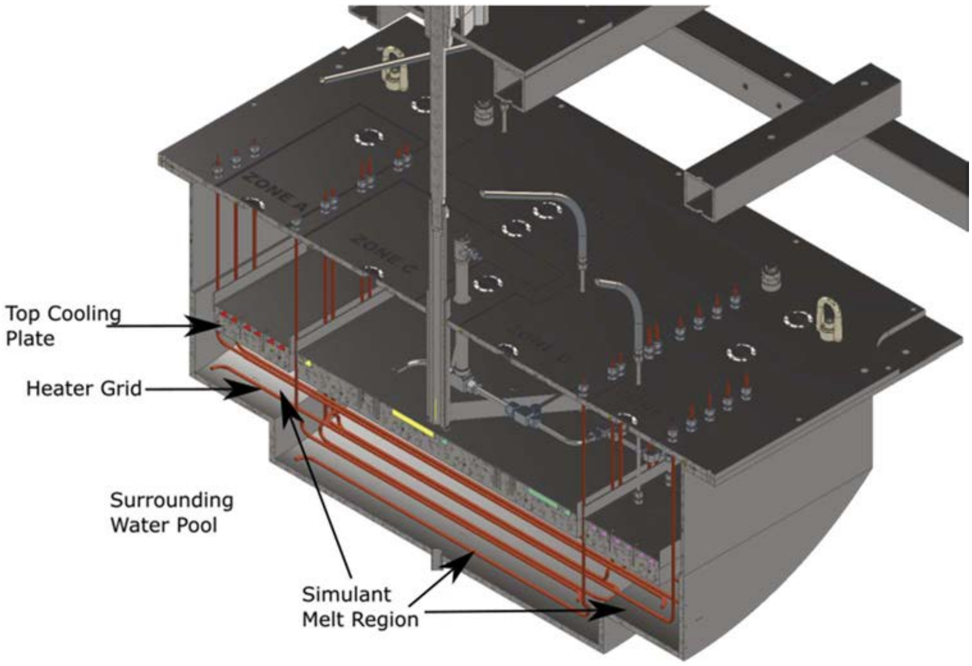


FIG. 14. Corium convection apparatus rendering (courtesy of J. Spencer, CNL).

These efforts addressed the phenomenon of heat transfer (convection by natural circulation, nucleate and film boiling, conduction, thermal radiation) / model for molten pool heat transfer, ranked H2 in the CANDU 6 and ACR-1000 PIRTs, and H4 in the Indian PIRT. They are also applicable to natural circulation in the melt, ranked H2 in the ACR-1000 PIRT.

Critical heat flux on the CV outer surfaces during in-vessel melt retention

One of the conditions for successful in vessel melt retention (IVMR) is that nucleate boiling is maintained on the outer surfaces of the CV in regions of high exiting heat flux. The nucleate boiling regime would cease, and the CV wall temperature would increase significantly, if the critical heat flux (CHF) is exceeded. The CHF is a function of various factors, including the geometry. The outer surfaces of the CV exhibit a great deal of geometric variations, including:

- Cylindrical CV main shell and sub shells;
- Vertical annular plates connecting the main shell and sub shells;
- Moderator outlets, CV drainline, and manhole constituting irregularities on the surface of the main shell;
- Vertical calandria tubesheets abutting end shields filled with spherical steel shielding balls;
- End fittings penetrating the calandria tubesheets.

Experimental programmes have been carried out at CNL, studying CHF in each of these locations. Although most of these results are proprietary and unpublished, it can be stated that none of these unpublished results contradict the feasibility of IVMR.

Several experimental studies of CHF in the unique CANDU end shield geometry have been published (see Figure 15). These included both heated experiments measuring CHF [77-78], and unheated experiments simulating boiling phenomena using the air barbotage technique [79-80]. Empirical correlations were developed, culminating in a best-estimate correlation of

CHF (q''_{CHF}) as a function of ball diameter (d) and elevation (L, defined as elevation on the calandria tubesheet, measured from its bottom), for cases of uniform exiting heat flux:

$$q''_{CHF}\left[\frac{W}{m^2}\right] = C_3 \cdot d^{3/2}[m] \cdot L^{-0.314}[m], 0.00635m \le d \le 0.0127m, 0.144m \le L \le 0.516m, C_3 = 2.63 \cdot 10^8, R^2 = 0.977 \quad (1)$$

Where R^2 is the coefficient of determination.

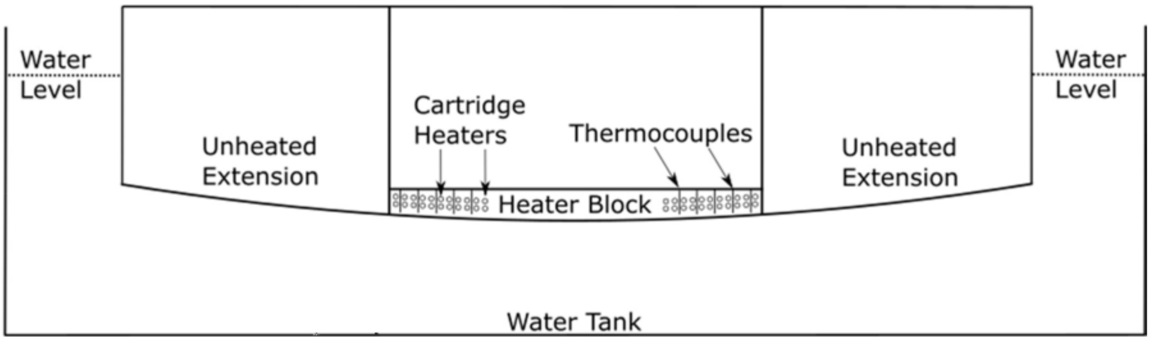


FIG. 15. Calandria vessel CHF apparatus schematic (courtesy of J. Spencer, CNL).

Heated experiments were performed using a copper heating block with a height of 0.6 m (the full anticipated height of the region of contact between the corium crust and calandria tubesheet). The apparatus permits the use of various exiting heat flux profiles, along with a variety of shielding ball diameters and compositions (see Figure 16). Experiments have also been performed with a similar apparatus making use of a stainless-steel heating plate. Publication of the results is anticipated in the near future.

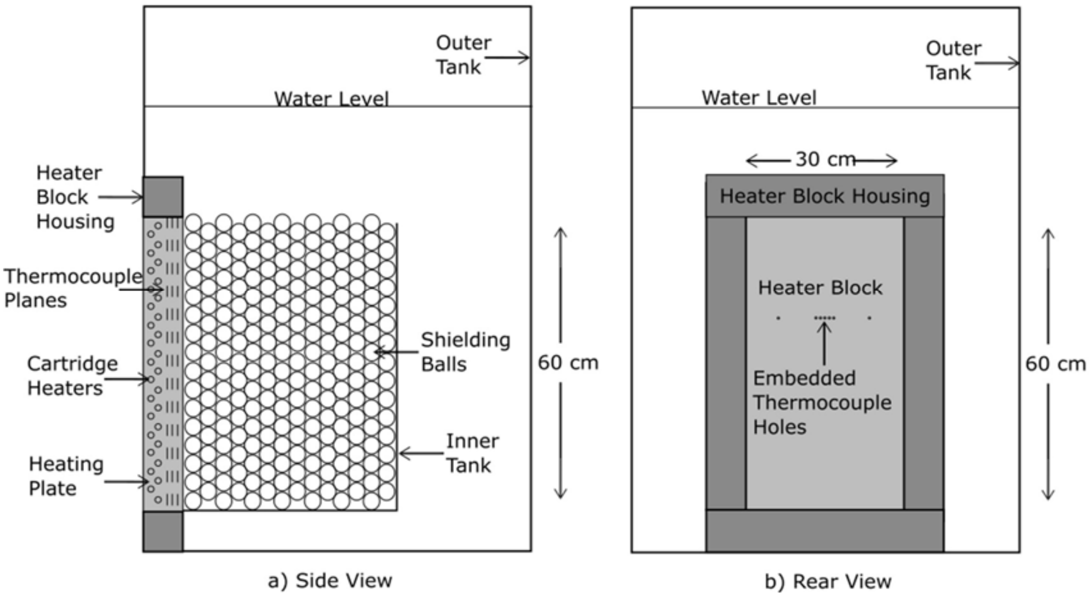


FIG. 16. End shield CHF apparatus schematic (courtesy of J. Spencer, CNL).

Complementary experiments have also been performed using the air barbotage technique, investigating the migration of both water and air (representing water vapour) in the end shield.

Parameters that were varied include air flow rate (hence equivalent heat flux), end shield cavity dimensions, and shielding ball diameter [79-80].

A key observation of this work is that, during boiling, oxide particles are formed on the surface of shielding balls and deposited on the calandria tubesheet surface. In the experiments, this mechanism results in an increase of CHF by up to 18%, relative to the value measured for a clean calandria tubesheet. It is anticipated that the same mechanism would be present in the reactor case, particularly if non-deaerated makeup water is utilized [81].

These efforts addressed the phenomenon of critical heat flux are classified in the PIRT exercises under different components:

- CV, ranked H2 in the ACR-1000 and Indian PIRTs, H1 in the EURSAFE PIRT, and not included in the CANDU 6 PIRT;
- End shields, ranked H2 in the CANDU 6 and ACR-1000 PIRTs.

4.2.2.3. Calandria vessel integrity during in-vessel melt retention with partially depleted calandria vault inventory

In the event that makeup water is not added to the calandria vault / shield tank, depletion will eventually lead the top of the CV to be dry. Radiative heat transfer from the top of the corium pool would lead to nontrivial heat flux upward, with a significant increase in the temperature of the upper portions of the CV in the absence of nucleate boiling. CNL has performed tests of this scenario using the CV heat stress apparatus (see Figure 17). The results of these experiments are unpublished; however, it can be stated that they support the viability of IVMR in the event that the upper portions of the CV are exposed above the water in the calandria vault / shield tank.

These efforts addressed the phenomena of failure of the vessel (ranked H2 in the CANDU 6 and ACR-1000 PIRTs), and load distribution / thermal stress distribution (ranked H2 in the CANDU 6 and ACR-1000 PIRTs).

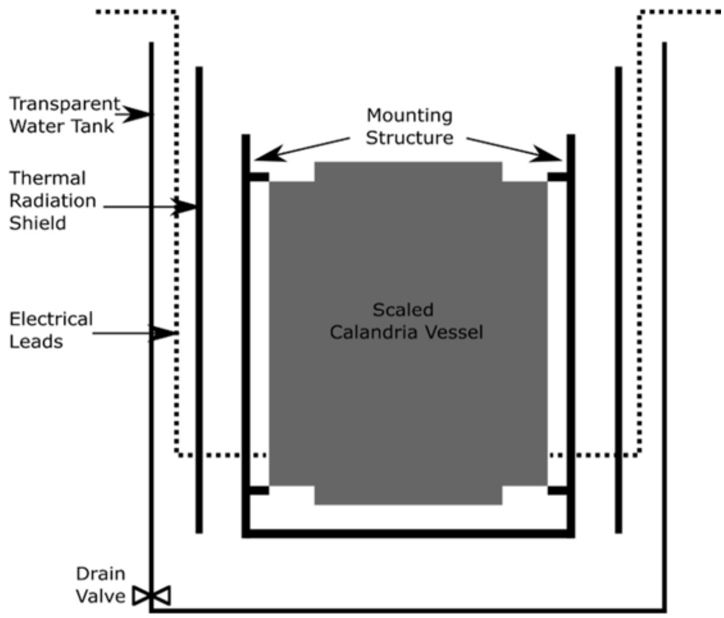


FIG. 17. Calandria vessel heat stress response apparatus schematic (courtesy of J. Spencer, CNL).

4.2.2.4. Corium plug formation in calandria vessel penetrations

The lower portion of the CV contains three penetrations that present potential paths for ingression of corium, which could threaten IVMR:

— Moderator outlets;
— CV drainline;
— End fittings.

The tendency of corium to form frozen plugs has been studied with small-scale experiments using 250 g of CANDU-prototypic corium (produced using a thermite reaction) at CNL [82]. The most significant result of these small-scale tests is the facilitation of progression to larger-scale plug formation experiments that would better represent the reactor case. Such tests (using up to 25 kg of corium) are planned for the near future.

Simulations of the ingression of corium into a CANDU end fitting have been performed using CFD [83]. Further discussion of this work is to be found in Section 6.

These efforts addressed the phenomenon of melt relocation (ranked H2 in the CANDU 6 and ACR-1000 PIRTs and H1 in the EURSAFE PIRT).

4.2.3. Future experimental plans

Plans for future experimental work are subject to uncertainty; however, this sub-section summarizes a range of work that is anticipated to be performed with a high degree of certainty in the near future.

4.2.3.1. Calandria vessel blowdown

The expulsion of liquid-phase water during depressurization of the CV following rupture disc burst has the potential to significantly impact the timing of subsequent progression through core damage states. Experiments using CNL's CV blowdown apparatus (see Figure 18) are planned, with the intent of providing data for model development and validation. The planned combination of test conditions and apparatus geometry will more closely resemble the phenomenon in question than any tests previously performed.

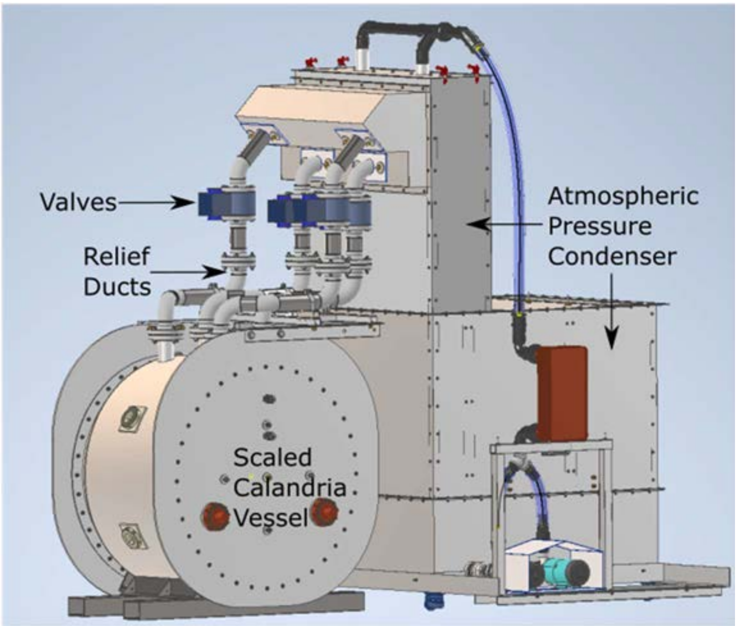


FIG. 18. Calandria vessel blowdown apparatus rendering (courtesy of J. Spencer, CNL).

4.2.3.2. Core collapse

As noted in subsection 4.2.2.1, material properties are hypothesized to constitute a primary source of bias impacting finite element predictions of CT loading limits. In order to address this shortcoming, experiments are planned to measure anisotropic mechanical properties of irradiated CT material taken from the same region as buckling was observed in the experiments and predicted in simulations. Uncertainties in local flux and spectrum will be addressed through the use of actual reactor material, with microstructural characterization providing further insight.

This work is planned to address the phenomena of core collapse (ranked H2 in the CANDU 6 and ACR-1000 PIRTs), and channel breakup (ranked H2 in the CANDU 6 and ACR-1000 PIRTs).

4.2.3.3. Heat flux exiting the calandria vessel during in-vessel melt retention

CNL plans a third experimental campaign using its corium convection apparatus. This will be the first use of the apparatus with a cooled and instrumented plate positioned at the top of the corium simulant pool. This will provide an upper boundary condition more consistent with the reactor case, in that a no-slip condition can be expected (in contrast with the liquid-gas interface present at the top in prior tests). This approach will also reduce uncertainties on the heat flux exiting the top of the corium pool.

This work is planned to address the phenomenon of heat transfer (convection by natural circulation, nucleate & film boiling, conduction, thermal radiation) / model for molten pool heat transfer, ranked H2 in the CANDU 6 and ACR-1000 PIRTs, and H4 in the Indian PIRT. They are also applicable to natural circulation in the melt, ranked H2 in the ACR-1000 PIRT.

4.2.3.4. Critical heat flux on the outer surfaces of the calandria vessel during in-vessel melt retention

CNL plans to perform tests of CHF on both the CV main shell and the calandria tubesheet surface facing the end shield cavity, using stainless steel heating blocks. Multiple campaigns are planned, involving varied surface roughness. This approach is anticipated to address a key uncertainty identified in prior work.

Additional experiments are planned investigating CHF near the junction of the annular plate and the subshell, where exiting heat fluxes are locally elevated.

This work is planned to address the phenomenon of critical heat flux, classified in the PIRT exercises under different components:

- CV, ranked H2 in the ACR-1000 and Indian PIRTs, H1 in the EURSAFE PIRT, and not included in the CANDU 6 PIRT;
- End shields, ranked H2 in the CANDU 6 and ACR-1000 PIRTs.

4.2.3.5. Corium plug formation

Progression from small-scale (250 g of corium) to larger-scale (up to 25 kg of corium) plug formation experiments are planned for the near future. These larger-scale tests will be more representative of the reactor case and are anticipated to be subject to smaller uncertainties in the key parameter of melt temperature.

This work is planned to address the phenomenon of melt relocation (ranked H2 in the CANDU 6 and ACR-1000 PIRTs and H1 in the EURSAFE PIRT).

4.2.3.6. Corium crust mechanical properties

The mechanical properties of corium crusts are a key factor in predicting the likelihood of corium ingression into vessel penetrations. Existing data are limited to non-CANDU corium, and have involved property measurements up to a limit of approximately 400 °C. Preparations are underway to perform material property tests on CANDU-prototypic corium samples produced by CNL’s cold crucible induction furnace (see Figure 19), at temperatures approaching 1000 °C.

This work is planned to contribute to knowledge of several phenomena:

- Cracking of crust (chemical interaction), ranked H1 in the CANDU 6 and ACR-1000 PIRTs;
- Crust formation ranked H2 in the CANDU 6 and ACR-1000 PIRT.
- Gap formation (crust/CV), ranked H1 in the CANDU 6 and ACR-1000 PIRTs;
- Core concrete interaction, ranked H2 in the CANDU 6 and ACR-1000 PIRTs and H1 in the EURSAFE PIRT;
- Melt relocation ranked H2 in the CANDU 6 and ACR-1000 PIRTs and H1 in the EURSAFE PIRT.

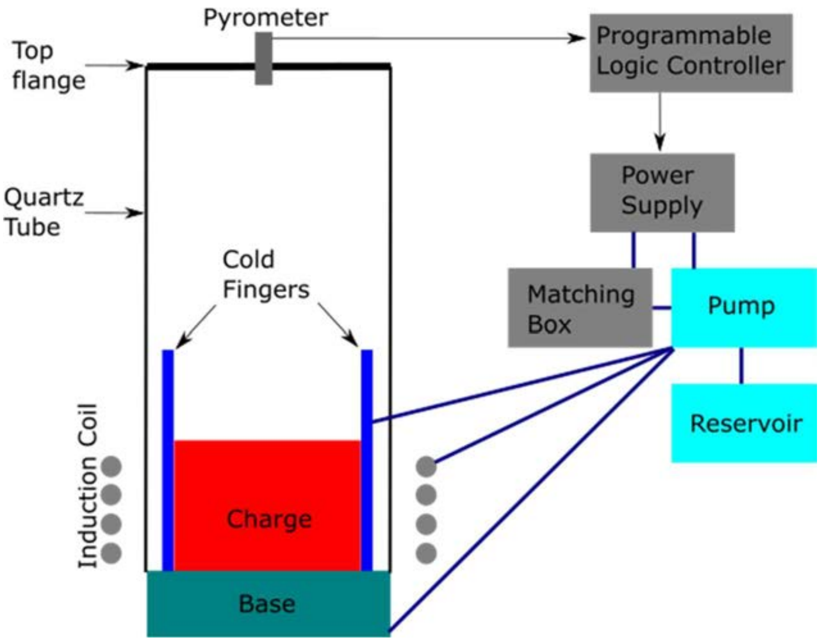


FIG. 19. Cold crucible induction furnace (CCIF) schematic (courtesy of J. Spencer, CNL).

4.2.4. Implications of experimental findings for model development and code validation

Several experimental results discussed in subsection 4.2.1 are particularly noteworthy for their potential use in model development and code validation:

- Existing and planned future results of the loading limits of CTs will constitute an opportunity to improve the current models used to predict core collapse. The timing of core collapse has a significant impact on fission product transport and hydrogen gas production (potentially threatening containment);
- Benchmarking of CFD simulations against CNL's corium convection experiments [76], followed by extension of these results to the reactor case has led to the prediction that the large majority of the corium in the CV during IVMR would be below the liquidus temperature. This result deviates significantly from the prior understanding of the IVMR scenario. Significant reduction in convection within the corium pool (as a result of high viscosity) would call into question the applicability of the models currently used to predict the spatial distribution of the heat flux exiting the CV. In addition, ingression of corium into vessel penetrations could be expected to be less probable with a very thick layer of highly viscous melt located inside the corium crust. Furthermore, if ingression did occur, the tendency of the corium to form a frozen plug would be increased, due both to a reduction in flow velocity and a reduction in the enthalpy of the material [84];
- Predictions of CHF on the outer surfaces of the CV during IVMR in current safety analyses (including those performed using all existing versions of MAAP-CANDU) necessarily make use of correlations that were developed for geometries not entirely representative of some locations on the CV outer surfaces. Future safety analyses may make use of geometrically applicable correlations that have been developed, or that can be expected to be developed, for all locations around the CV outer shell.

4.3. EXPERIMENTAL PROGRAMME UPDATES IN INDIA

There are several experimental facilities that were built and operated over the years since IAEA-TECDOC-1594 was published [1]. The experimental programme for PHWRs in India is aimed at studying entire range of postulated accident conditions, from the LCDA phase to the SCDA phase. In addition, there are specific experiments aimed at establishing/demonstrating the adequacy of prescribed SAMGs. This section therefore describes specific experimental facilities designed to address postulated accident conditions expected to occur chronologically from the postulated initiating event, followed by a sub-section on SAMG specific experiments. The experimental programme is aimed at resolving phenomena of high importance and low level of knowledge on priority. Apart from H1 and H2 rank phenomena (see Table 10 in Section 4.1.4), experiments were also carried out for the following phenomenon (mostly encountered in LCDA) with low or medium impact (L or M) or high impact (H) with limited or adequate knowledge (3 or 4) level. These are briefly covered in relevant sections mentioned in the parenthesis in Table 14 given below.

TABLE 14. RANK OF PHENOMENA AND RELEVANT EXPERIMENTAL PROGRAMME

Phenomena	Rank of Phenomena
Fuel clad behaviour under heat up and burst	L3 (Appendix II.1 and I.2)
Properties Zr, 2.5 wt % Nb (material of PT) at high temperature	H3 (Appendix II. 3 (oxidation studies) & I.6)
Fuel bundle heat up behaviour (LCDA)	M4 (Appendix II.4)
PT-CT thermal contact conductance (LCDA)	L4 (Appendix II.4)
PT-CT gap heat transfer (LCDA)	M3 (Appendix II.4)
Contact angle of PT-CT (LCDA)	L2 (Appendix II.4)
Properties of SS (non-Zircaloy core components) at high temperature	H3 (Appendix II.6)
PT asymmetric ballooning from stratified flow (SCDA)	M4 (4.3.2.1)
Thermo-mechanical deformation of exposed channel (SCDA)	M3 (4.3.2.2)
Suspended and terminal debris heat up	M3 (4.3.2.4)

Most of the LCDA specific phenomena pertaining to these categories have been addressed through experimental investigation. These investigations have not only allowed more accurate modelling of the early phase of event sequences that can lead to a severe accident, but also have produced valuable results, capturing the effects of difference in material properties stemming from different manufacturing routes adopted. The results also indicate that the channel integrity can be maintained in most of the LCDA specific accident sequences if moderator cooling is available. However, since a review of experimental programmes updates related LCDAs would go beyond the scope of this document that focusses on severe accidents, those are placed in Appendix II.

4.3.1. Studies specific to severe core damage phase

This section presents the experimental updates for the studies specific to SCDA phase of accident phenomena.

4.3.1.1. *Fuel channel integrity: asymmetric ballooning of pressure tube*

In the event of flow stratification within the fuel channel, the exposed fuel pins and the exposed portion of PT is likely to get heated up. Prevailing higher pressure within the channel can lead to asymmetric ballooning of the PT. In order to assess channel integrity under such condition, experiments were performed with a 19-pin fuel bundle simulator housed inside PT-CT [85]. The specially made 19-pin bundle simulator facilitated simulation of radial peaking factors and also had a provision to power-on selective fuel pin simulator. Figure 20 shows this 19-pin bundle simulator.

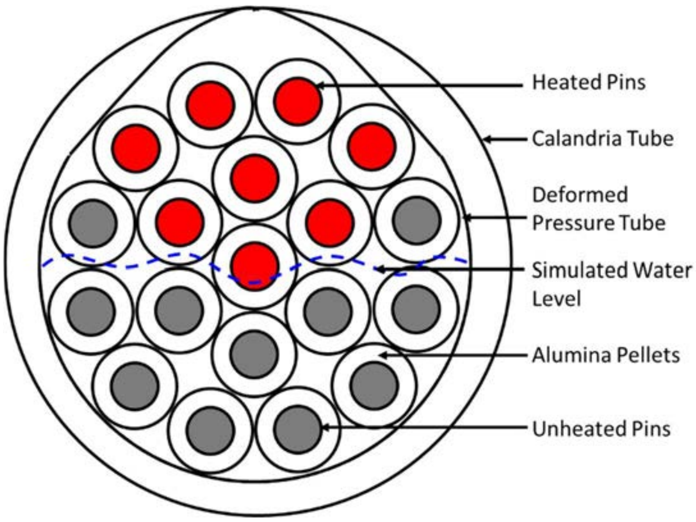


FIG. 20. 19-pin bundle simulator with selective heatup of pins (courtesy of O. S. Gokhale, BARC).

The selection of powered fuel pin simulators was made to simulate heatup of the exposed fuel pin locations. The PT was internally pressurised with pre-defined pressure in the range of 1-6 MPa. The PT-CT assembly was kept in a water tank to simulate moderator cooling.

It was observed during these experiments that the radial expansion of the PT was much lower than in the case of symmetric heating of PT. Ballooning was found to initiate at much lower temperature. The integrity of PT was maintained for internal pressure up to 2 MPa, however, for internal pressure of 4 MPa and above, the PT integrity was compromised [85].

4.3.1.2. *Fuel channel heatup study*

The transition of postulated accident condition in PHWRs from the LCDA phase to SCDA phase is marked with failure of fuel channels. The failure mechanisms are governed by channel sagging characteristics in a postulated initiating event leading to low pressure in the PHTS. Sagging behaviour also determines the nature of suspended debris formed. It also results in sagging pull-out force on the PT rolled joints. Thus, an experimental setup was conceived to study the sagging characteristics of a standard 220 MWe Indian PHWR under postulated initiating event of LOCA with loss of ECCS [86].

The experimental facility made use of a 1:3 scaled down reactor channel. The decay heat was simulated by electrically heating the PT. The fuel weight was simulated using tungsten bundle weight simulator inserts. The facility had a provision to simulate up to 3 such scaled down channel simultaneously. There was a provision for selective heating of a particular or all the channels. The environment in the experimental setup was controlled to be either inert (Argon) or a mixture of argon and steam to study the sagging behaviour under oxidation environment. The experimental measurements included temperature transients at multiple locations, concentration of hydrogen and deformation of the channel measured through optical imaging technique.

For the single channel studies in inert environment, the channel sagging initiated at about 400ºC. Figure 21 shows a typical picture of a single channel in sagged position. Localised thinning of PT was observed at locations between the fuel bundle weight simulators. The sagging was observed to exert a pull-out force of about 1.66 kN. The sagging behaviour continued without failure of channel up to 855ºC and was found to take place through mechanism of thermally assisted creep [86].

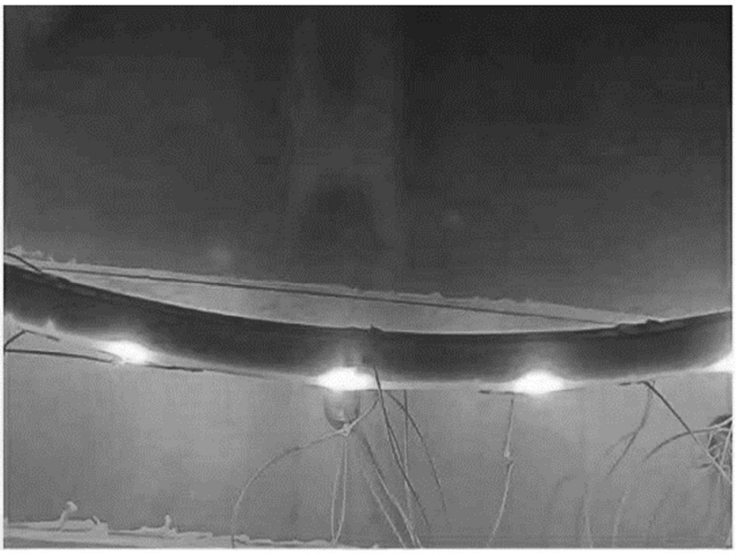


FIG. 21. Sagging study for a single channel in Argon environment (reproduced from Ref. [86] with permission courtesy of Elsevier).

For studies performed under oxidizing environment, the CT was found to undergo significant oxidation, reducing its structural strength [87]. Failure of CT allowed pathways for the oxidising environment to react with PT. For channel average temperatures higher than 760ºC, significant sagging was observed, and channel failed into multiple segments in the cooling period due to transformation of β-Zr into α-Zr.

Experiments were also performed with three channels arranged one below other with all the three channels being heated [88]. The sagging of these channels was found to occur simultaneously with no contact between the channels. Figure 22 displays an example of this behaviour. Insignificant circumferential temperature gradient was observed on the central channel.

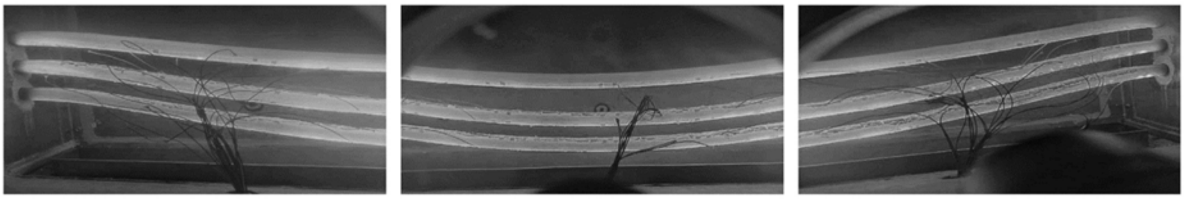


FIG. 22. Sagging study for three channels (reproduced from Ref. [88] with permission courtesy of Elsevier).

Another variant of the three-channel experiment was performed with top two channels heated and bottom channel without heating [89]. The top two channels were found to sag and transfer their weight on the bottom, unheated channel. Severe compressive deformation of the middle CT was observed. However, the bottom CT was successful in supporting the weight of the top two channels.

4.3.1.3. Study on hydrogen generation from fuel and fuel channels

Disassembly of fuel channels from the original position, either due to pull-out from the rolled joints or due to channel rupture, provides a direct access path for CV environment to interact with fuel pin bundles. The disassembled channel lengths can vary from small stubs, of the order of a fuel bundle length, caused due to breakage of channels in-between the bundle positions to large lengths, typically generated due to failure from pull-out of the channels. Presence of moderator under boiling conditions due to submerged channels, provides significant amount of steam to allow oxidation of un-oxidised fuel pin cladding. The extent of oxidation of cladding and associated hydrogen generation depends on the amount of steam ingression possible into the disassembled portions of channels and cladding temperature.

An experimental setup was thus designed to assess amount of steam ingression in a disassembled channel portion of length of 1 m, held in steam flow equivalent to the steam generated from lower, submerged channels [90]. The test setup had the capability to position the channel in different orientations making different angles with respect to horizontal. The channel was made up of 19-pin fuel bundle simulator with radial power profile, housed in a PT-CT assembly. The bundle simulator was heated using electrical heating of Kanthal rods positioned at the centre of each fuel pin simulator. The temperature of the fuel cladding at various axial and radial locations and the hydrogen generation rate were measured to evaluate extent of oxidation and steam ingression.

It was found that the steam ingression is limited to the region near the ends of the channel for the horizontal configuration of the channel [90]. As the channel inclination is increased up to 20º, the amount of steam ingression also increases, leading to availability of more steam for oxidation [91]. However, ingression of steam also produces a cooling effect on the cladding, which becomes dominant as compared to rise in the heat generation due to oxidation reaction, beyond 20º inclination of the channel. Thus, no significant increase in oxidation is observed for channel inclination beyond 20º.

4.3.1.4. Study on PHWR debris bed under boil-off conditions

The dislocation of fuel channels from the original location and relocation into the CV containing moderator can lead to formation of terminal debris bed for PHWRs. The type of debris in this condition are significantly different from the particulate debris expected for the LWRs. For PHWRs, the terminal debris are expected to be in the form of long pipe like structures made up of channel portions, open from both ends, providing a pathway for the moderator to enter in the channels. The PT and fuel bundle are likely to be resting on the CT making this pathway non-uniform in cross section. Figure 23 illustrates this behaviour. The study of coolability of such a geometry is essential to understand further progression of the accident.

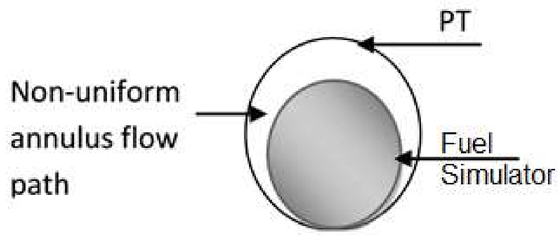


FIG. 23. Schematic of the fuel pin simulator sitting on PT providing non-uniform flow path (courtesy of O. S. Gokhale, BARC).

An experimental setup was thus designed to characterise behaviour of simulated debris bed under different submergence level [92]. The debris bed consisted of a single debris formed from a 1 m length of a channel, containing a fuel pin simulator, PT and CT.

The channel was kept submerged under the moderator with different submergence levels (see Figure 24) and temperature measurements were performed at different locations for various power levels. The debris was found to be coolable up to 12% level of submergence with heat transfer coefficient ranging from 0.73-15.5 kW/m^{2}K [92].

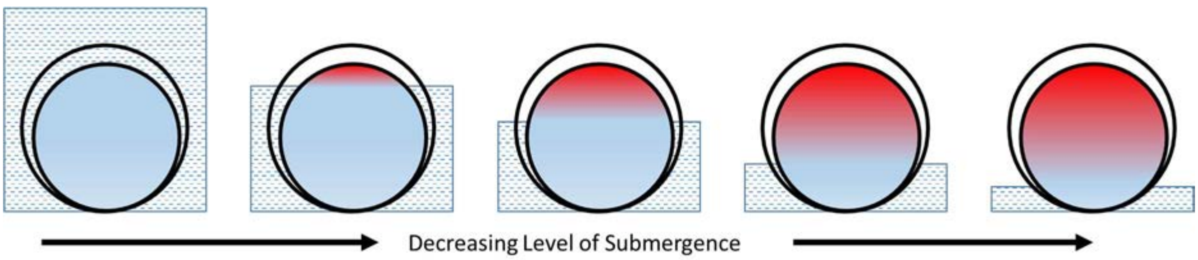


FIG. 24. Debris arranged with decreasing level of submergence for different experiments (courtesy of O. S. Gokhale, BARC).

Moderator boil-off and slow exposure of terminal debris bed was studied [93-94] with the help of scaled down experiments to understand the PHWR specific debris bed heat up. Figure 25 shows the debris bed simulation where eight simulated heated channels along with two un-heated channels were arranged in triangular pitch. The investigation has been carried out at 1% of decay power. The study shows that the majority of debris bed does not undergo any heat up till it is submerged under water. This is in contrary to PWR debris bed where the centre of debris bed gets heated is expected due to presence of high void fraction in localised regions of the debris bed. The exposed debris bed tends to get heated up due to absence of coolant. However, the steam generated from the submerged portion of debris bed limits the exposed channel heat up by convective heat transfer.

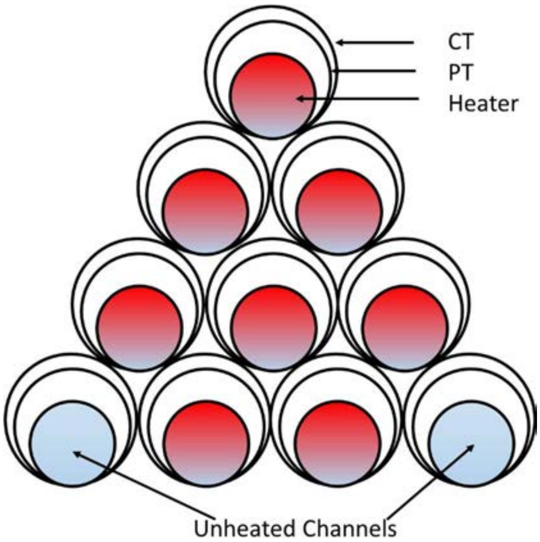


FIG. 25. Terminal debris bed arrangement with simulated channels (courtesy of O. S. Gokhale, BARC).

4.3.1.5.Study on calandria integrity and CHF evaluation

Total core collapse of the channels within the CV will lead to formation of debris bed within the CV. Depending on the magnitude of decay heat and heat removal paths available through the CV and end shield including conduction, convection and radiation modes of heat transfer, the debris bed may undergo melting. IVMR through the external reactor vessel cooling is considered as an effective method for ensuring integrity of the CV during such conditions. Under the IVMR conditions, it is necessary to ensure that the imparted heat flux due to melt is effectively removed by boiling/convective heat transfer over the CV external surface and is less than the critical heat flux (CHF).

Evaluation of the CHF on the calandria vessel

Due to large diameter of the CV, the vertical and upward facing regions of the CV are benefitted in terms of heat transfer by sliding of bubbles along the curved surface, whereas the downward facing surface experiences crowding of bubbles. In addition, there are moderator drainpipes located near the bottom of the CV, which can further affect the local heat transfer coefficients on these downward facing surface.

In order to assess whether the imparted flux over the CV is higher or lower than the CHF value, experiments were performed to obtain CHF for the CV surface. A 25^{o} sector of bottom location of a cylindrical vessel with 3.9 m radius of curvature was used as a representative of the CV. The CV was electrically heated using surface heaters mounted on the inside surface [94]. CHF values were obtained for a range of bulk water temperature and indicated that the CHF is much larger than the heat flux expected from the debris bed, both in solid as well as molten form of the debris bed. Photographs indicating patterns of bubble formation at different heat flux levels. Experiments were also performed by attaching a 50 mm diameter drainpipe on the CV external surface to measure its effect on CHF values at the bottom. The effect was found to be negligible [95].

Calandria vessel integrity under in vessel retention

In order to assess the CV integrity under IVMR, experiments were also performed with simulant melt generated from induction furnace, poured into a stepped, scaled down CV [96]. The CV is kept submerged under water and the decay heat is simulated using electrical heaters provided within the CV, after pouring the melt. The experiments showed that the heat flux is limited to 168 kW/m^2 which is significantly lower than the CHF. The maximum strain in the stepped vessel is limited to 0.18%. Post-test examination indicated that there is no gap formation between the crust and the vessel, indicating a better path for heat conduction available through the crust [96].

Both these studies indicate that the CV integrity can be maintained in case of debris bed formation at the CV bottom takes place.

4.3.1.6. Study on containment behaviour

Assessment of ultimate load capacity of containment of Indian PHWR has been pursued through a 1:4 size Bhabha Atomic Research Centre (BARC) Containment (BAR-COM) Test Model. Figure 26 shows this test model that represents a 540 MWe PHWR pre-stress concrete inner containment with design pressure (Pd) of 0.1413 MPa [97].

FIG. 26. BARC Containment test model at BARC-Tarapur test facility with details of embedded sensors and cable panels (courtesy of BARC, India).

The BARC Containment (BARCOM) test model uses large numbers of instruments for measurement of specific parameters such as measurement of strain using 350 electrical resistance strain gauges mounted on various surfaces, measurement of vibration levels using wire strain gauges, pressure measurement using ordinary dial gauges and potentiometer based dial gauges and pressure cells, measurement of change in inclination of internal structures using tiltmeters, and temperature measurement using resistance temperature detectors.

Figure 26 gives an overview of various embedded sensors in BARCOM along with an exhaustive cable workup to the instrumentation and control building. The experiments in this facility are spread over four phases. The sensors and the data logger systems have undergone functionality tests up to 0.049 MPa (0.35 Pd) as a part of Phase I of the programme. Subsequently, Phase II of the programme was completed by pressurizing the BARCOM facility up to the design pressure (Pd) of 1.44 kg/cm2 (0.1413 MPa) in August 2010 Repeat test were performed up to the design pressure to obtain consistent and repeatable leakage-rate as a part of Phase-II experiments. Under the Phase-III of the programme over-pressure test was performed up to a pressure of 0.2207 MPa (1.56 Pd) in December 2010. The tests revealed "first appearance of crack" near the main air lock and emergency air lock as measured through online measurement of the inelastic strain. This was also confirmed by performing soap bubble test. The strain pattern observed in the repeat experiments also indicated the development of parallel cracks with localized strain field in the fracture process zone near the main air lock, the emergency air lock locations and the first through thickness cracks in BARCOM test-model were identified successfully.

4.3.2. Studies specific to severe accident management guidelines

Injection of firefighting water in the CV and in the calandria vault are two SAMG actions prescribed for PHWRs. The efficiency of these actions in arresting accident progression needs to be verified by analysis and experiments. Towards this, specific experimental setups were designed to demonstrate the effectiveness of the prescribed SAMG actions.

4.3.2.1. Study on in-calandria injection reflooding of exposed channel

The suspended debris bed formed during LCDA phase of accident is cooled either from the PHTS coolant, if PHTS is intact, or by boiling moderator for channels under submergence or by steam generated from the submerged channels present underneath. Injection of water into the CV is intended to flood the CV, establish coolability of the channels that get submerged during the process and ensure that temperature escalation of the fuel pin, PT and CT is arrested.

An experimental setup designed for this consisted of a 2 m long length of PHWR channel with 19-pin fuel bundle simulator contained in PT-CT assembly [98]. The fuel bundle simulator was electrically heated to simulate decay heat with radial power profile. The channel was kept suspended over a water pool containing submerged electrical heaters for simulating steam generation from underneath submerged channels. Water injection was started in the water pool at a pre-specified, scaled-down injection flow rate such that the level in the pool can rise above the suspended channel. Measurements indicated that the prescribed injection was able to arrest temperature escalations in the channel for various conditions of initial channel temperature, decay heat levels and moderator sub-cooling.

4.3.2.2. Study on ex-calandria injection reflooding of exposed calandria vessel

Injection of water into the calandria vault is expected to cause reflooding of already exposed CV. The reflooding of the CV is a combination of complex phenomena that involve level rise above the CV surface, establishment of permanent contact through re-wetting process, formation of water pool above the surface and decrease in CV temperature, all this when the pool is not static but developing. An experiment was performed with the CV top surface approximated as flat plate, heated from below using infra-red heaters and cooled by addition of water from all sides, such that the water pool develops with a pre-defined rate of rise in level [99]. The effect of rate of water flooding on the quenching behaviour of the plate was studied. The experiment demonstrated that 50% of the SAMG prescribed rate is sufficient to rewet the heated surface of the CV.

4.3.3. Summary of R&D gap areas addressed through experimental programme in India and future work

The experiments performed in the last couple of decades have helped in providing significant insights into the phenomenology at various stages of severe accident and also contributed towards development of numerical models to simulate these phenomena. The oxidation studies on fuel cladding and PT material have improved Arrhenius reaction coefficients for Indian PHWR materials, the PT-CT deformation studies have confirmed channel integrity for all types of symmetric deformation with moderator as heat sink. Several experiments on channel deformation with loss of moderator heat sink have indicated channel failure due to oxidation. The suspended and terminal debris bed formed are found to be cooled due to steam ingression for non-horizontal cases and even under very low submergence levels respectively. The external coolability of the CV has been demonstrated, confirming that there is no CHF occurring on the CV, even for areas containing protruding pipes. The adequacy of prescribed SAMG actions has been verified with single channel experiments.

In the future, experiments are planned for the areas already highlighted in the PIRT but not covered by experiments so far, such as large-scale molten fuel coolant interaction and moderator thermalhydraulics.

4.4. HYDROGEN EXPERIMENTS

In nuclear water-cooled reactors, it is well known that one of the principal sources of H_2 gas during a postulated severe accident is the chemical interaction of Zircaloy fuel cladding and the water coolant [100-101]. Similarly, under severe accident conditions in PHWRs, H_2 would be produced by the fuel cladding (zirconium) which would undergo oxidation reaction with steam at high temperature and by the metal content in the CTs and PTs.

However, it has also been alluded that in PHWRs, not only will H_2 be produced by the heavy water is inside and outside the fuel channels in the event of core melting, but rather D_2 could be the dominant species in an accident. H_2 and D_2 have similar character and safety analyses usually assume that the D_2 produced inside the CV is equivalent to H_2 [102].

This section reviews studies and experiments comparing the differences between H_2 and D_2 and PHWR specificities in terms of hydrogen generation under severe accident conditions.

4.4.1. Recent experiments and lessons learned in terms of D_2/H_2 differences

The production of D_2 during a severe accident due to heavy water (D_2O) in the heat transport system and the CV of PHWRs has led to numerous discussions and concerns over the use of H_2 in safety analysis instead of D_2. To date, virtually all testing on PARs has been performed with protium (H_2). It has however been argued that PARs capacity to recombine D_2 may be significantly different compared to H_2. This would be because the molecular weight of D_2 is twice as high as H_2. Since diffusion coefficient is inversely proportional to the square of molecular weight, then the molecular diffusion of D_2 to the catalyst surface would then be much slower than H_2 [103]. It was well known that the catalytic oxidation of H_2 or D_2 reaction in a PAR is limited by one or more of the following: transport through a boundary layer of the gas from the bulk flow by convection and diffusion, the diffusion to the porous surface of the catalyst and the reaction on the surface of the catalyst [104]. The lower diffusion coefficient of D_2 compared to H_2 is expected to reduce the PAR recombination capacity. On the other hand, the higher enthalpy and density of D_2 compared to H_2 are expected to increase convection through the PAR and its recombination capacity. The effect of D_2 compared to H_2 due to these differences needed to be quantified by means of experimental research.

To assess the effectiveness of the PARs to recombine D_2 gas, laboratory-scale experiments were carried out together with numerical simulations using a COMSOL CFD model at AECL[6]. A series of experiments were performed in two small-scale experimental systems designed for catalyst testing in H_2-air mixtures. Tests were performed in a Carberry-type reactor system to evaluate the difference between the reaction rates of H_2 and D_2 on the PAR catalyst surface without mass transfer effect. Similarly, experiments were performed in a catalyst activity bench scale (CABS) system, to assess the difference between H_2 and D_2 reaction rate on the PAR catalyst surface, this time with mass transfer effect.

The results showed reduced recombination rates with D_2 compared to H_2 for both the Carberry type reactor and CABS tests. However, the difference in recombination rates was lesser in the CABS tests (with an average D_2/H_2 conversion ratios of 0.90, or 10% lower rate for D_2) due to mass transfer and free convection effects. Experimental results shows that recombination rate during PAR operation is influenced by mass control effects. Therefore, the CABS tests are more representative of the recombination rates in PAR operation.

The impact of the D_2 lower diffusion coefficients and higher enthalpy of formation on the PAR capacity was also investigated using a COMSOL CFD model of the PAR and the output was compared against those experimental results. The simulations also indicated that PAR recombination capacity is not significantly affected by exchanging D_2 with H_2. However, the recombination capacity of D_2 was 5% higher than H_2 in the simulations (see Figure 27), which was inconsistent with the experimental tests.

[6] The experimental and analytical results conducted by AECL and presented in this section are reproduced with the agreement of AECL. Detailed results can be found in Gardiner, L., Test Report, Experimental Report on small scale testing to Assess the effect of Deuterium on PAR recombination, PAR-68480-TR-010, Revision 0, March 31. 2015.

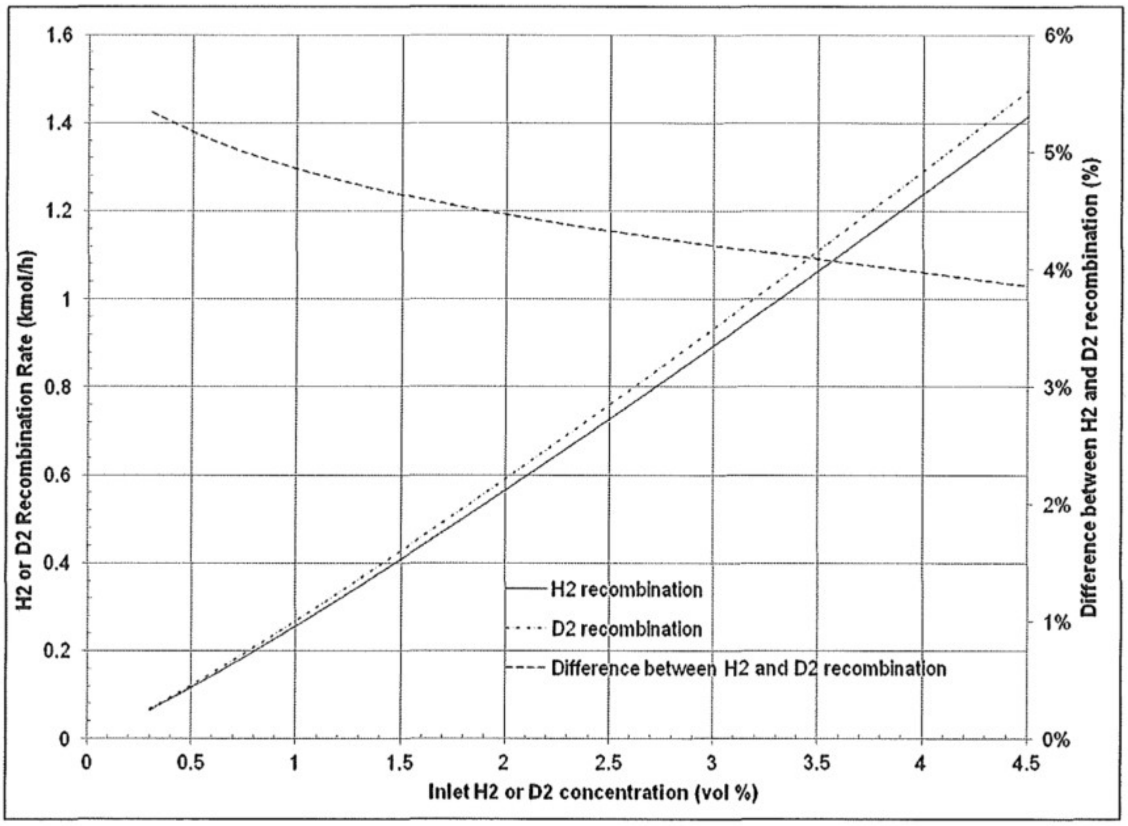


FIG. 27. Model capacity prediction comparison between hydrogen and deuterium (courtesy of CNSC, Canada).

Given this inconsistency, experiments were performed by the CANDU Owner's Group (COG)[7] to compare the difference in PAR performance (self-start threshold and recombination rate) between H_2 and D_2 based on small/bench-scale experiments, which eliminated the effects of mass transfer. It was found that the reaction rate of D_2 was lower than with H_2 due to the lower diffusion rate of D_2 through the boundary layer to the catalyst surface. However, the large-scale experiments, which included all the phenomena incorporated in a full-size PAR, demonstrated that there is no significant difference in the hydrogen recombination rate between D_2 and H_2. The results suggest that, when PAR is fully operating, the PAR recombination rate is primarily dominated by the fluid dynamics. Further, the tests to investigate the PAR self-start behaviour with D_2 versus H_2 on a large scale indicated that D_2 required a higher concentration to self-start compared with H_2. However, the difference remained minor. Overall, it was found that there is no significant difference in PAR performance with D_2 compared with H_2.

Recently, in the frame of the Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) THEMIS project, large scale PAR experiments have

[7] The high-level summary of those experiments presented in this section are published with the agreement of COG. The experiments results are available to COG members under:

— L. Gardner, T. Clouthier, B. Thomas, and R. MacCoy, Experimental Data Report on PAR Performance with Deuterium, COG-16-2010, May 2017;
— L. Gardner, Experimental Data Report on PAR Self-Start and Capacity Performance with Deuterium, COG-18-2003, April 2020.

been conducted covering typical DECs [105]. One of the tests from the PAR-test series was performed (initial conditions: 3 bar, 117 °C, air-steam mixture with 60 % steam) to determine the influence of changed thermophysical properties onto the catalytic oxidation of the combustible gas. A Framatome FR-380 PAR was installed in all tests in the THAI test vessel, which was reduced to half of the number of catalytic foils of the actual PAR. Both THAI vessel and PAR were well instrumented (see Figures 28 and 29) to investigate the PAR performance behaviour (start-up, recombination capacity and recombination efficiency) and the interaction between an operating PAR and the surrounding containment atmosphere.

The test results revealed a slight reduction in PAR efficiency (10 %) for D_2 compared to the reference case conducted using H_2 at same initial conditions. The recombination efficiency is calculated as a ratio of concentration between PAR inlet and outlet to PAR-inlet concentration. The difference in recombination capacity (in g/s) is consistent with the H_2/D_2 molar mass ratio. The performance of the test data in reference to vendor correlations was also analysed while scaling the vendor correlation with the molar mass ratio of hydrogen and deuterium. The vendor correlation for H_2 provided satisfactory results when applying for D_2 using H_2/D_2 molar mass ratio.

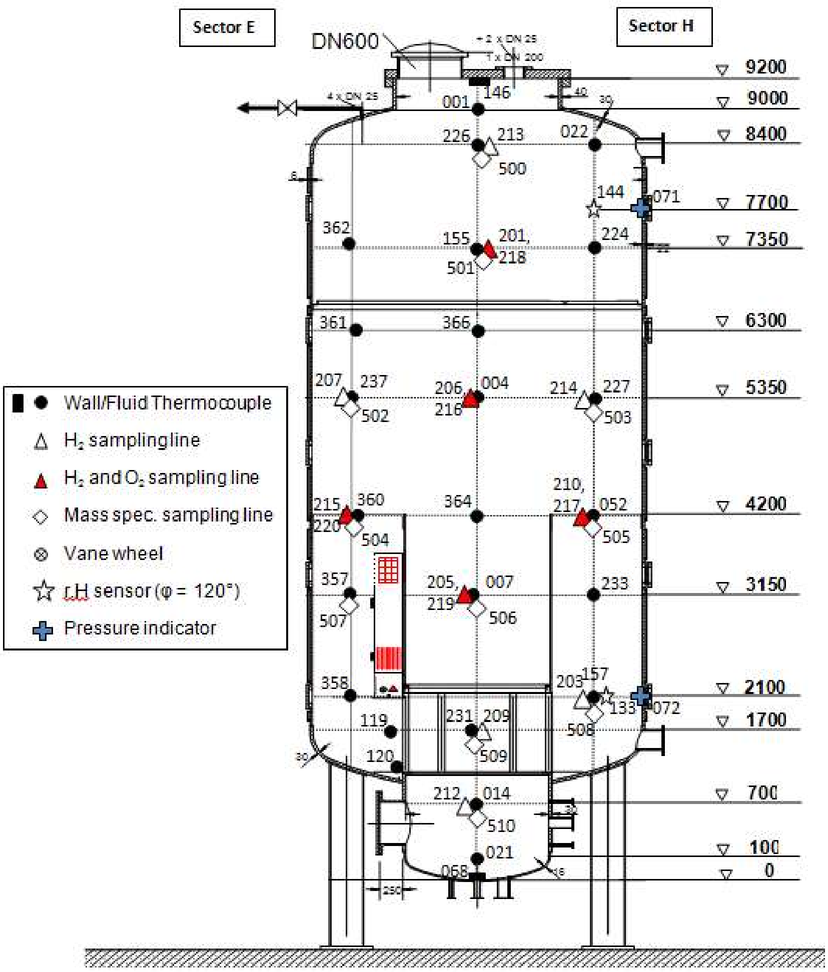


FIG. 28. THAI vessel in the PAR performance behaviour experiment (courtesy of Becker Technologies, Germany).

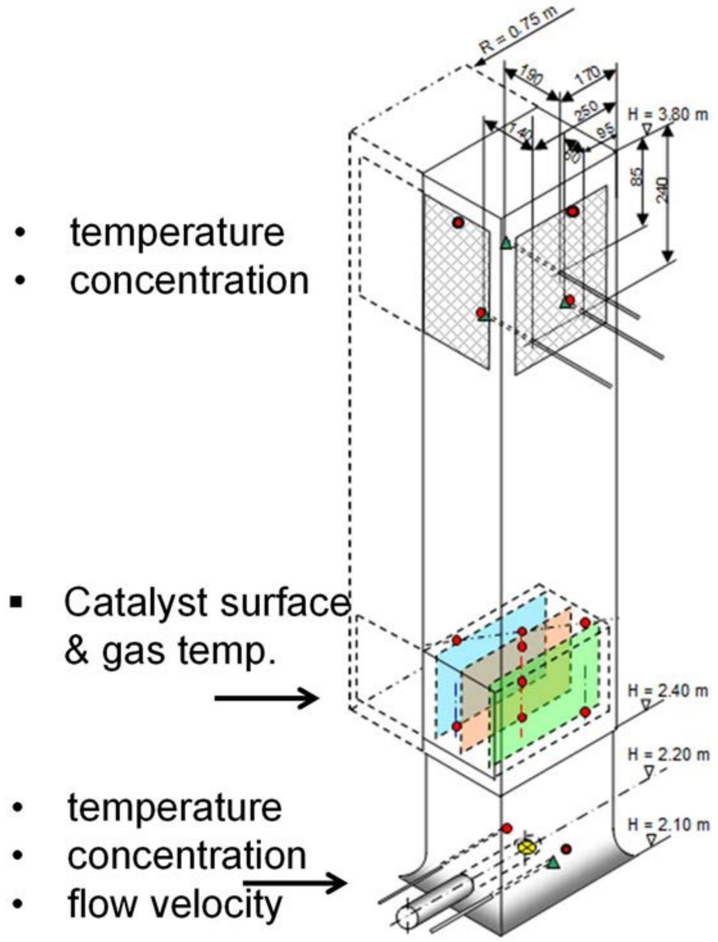


FIG. 29. PAR in the PAR performance behaviour experiment (courtesy of Becker Technologies, Germany).

4.4.2. Recent experiments on late phase of severe accident

The late phase or ex-vessel phase of a severe accident in a PHWR is not very different than in a typical PWR when it comes to hydrogen source term and the implications of molten core concrete interactions. However, since PHWRs have hundreds of feeder pipes connected from fuel channels at the reactor face to the inlet and outlet headers, it has been asserted by some researchers that the feeders and end shield plugs could also produce H_2 under severe accident conditions [106]. The oxidation of carbon steel could therefore contribute to the hydrogen source term, especially in the long term [107]. The assumption however only relies on very limited experimental studies and computer simulations on correlations for the generation rate.

For instance, CANDU 6 severe accident simulations using MAAP-CANDU have indicated that feeder oxidation for periods of over ~20 hrs could be possible, but the calandria shell oxidation may not be significant. The studies confirmed that in some cases, the oxidation of steel during a severe accident cannot be ignored especially during the late phase and/or ex-vessel phase [107]. Accident scenarios leading to those conditions are however considered unlikely.

This nonetheless motivated the CNSC to examine a wider range of situations and sensitivities to different parameters and assumptions during oxidation of the end fittings and feeders before and after channel rupture. These studies are still ongoing and are concentrated on 7 specific steel alloys (from feeder tubes, cast shield and channel closure plugs), covering a temperature range from 900 to 1600 K (627 to 1327°C) and using the Ultra High-Temperature Apparatus, to develop conservative correlation models for the steels alloys considered.

In regards of the PAR response during the late phase of the accident, the ongoing OECD/NEA THEMIS project also has a focus on severe accident "late phase" conditions, e.g., reduced O_2 concentrations, presence of CO, representative fission products [105]. Topics related to combustible gas include hydrogen/carbon monoxide combustion and PAR performance behaviour under typical late-phase conditions (O2-lean, CO, steam and elevated pressure up to 4.5 bar). Possible poisoning of PAR catalyst in O_2-lean atmosphere has been investigated under challenging conditions, namely O_2-lean, ambient temperature. PAR tests carried out with the sequential release of CO and H_2 demonstrated that under ambient conditions, the onset of PAR is delayed by CO (in an atmosphere free of H_2), and the onset of PAR recombination does not commence until H_2 is introduced. However, at elevated temperatures (>117 °C in THAI tests), the behaviour of PAR onset is not compromised by CO, since CO undergoes oxidation by PAR, even in the absence of H_2.

Regarding the effect of CO on recombination capacity, comparison with another PAR test conducted in the frame of OECD-NEA THAI project with identical initial conditions but lacking CO showed a slight decrease in H_2 depletion efficiency under conditions of CO recombination [108]. The PAR demonstrates approximately half the efficiency in oxidizing CO compared to its efficiency in oxidizing H_2. In contrast to H_2, the PAR's ability to recombine CO remains unaffected by oxygen deficiency as long as the O_2 concentration exceeds its stoichiometric demand.

4.4.3. Conclusions

Hydrogen risk assessment focussing on both generation and distribution behaviour inside containment has been subject of investigation in recent national and international projects and significant progress has been reported towards management of combustible gases under accident conditions. However, there remain still some uncertainties, particularly in assessing combustible gases risk under representative boundary conditions for severe accident scenarios. The input on representative boundary conditions is also needed to facilitate reliable assessment of installed mitigation systems and for validation of advanced safety analyses tools. In this context, hydrogen generation sources (amount and rate), such as due to feeder pipes oxidation would need further efforts. Regarding performance behaviour of hydrogen mitigation systems, such as PARs, reasonable efforts have been put on investigations related to assessing the effect of different properties of H_2 and D_2 and effect of late phase accident conditions, such as effect of CO, O_2-lean.

4.5. UPDATES FOR IMPORTANT PHENOMENA

Using expert judgment, the PIRT identified important severe accident phenomena and the knowledge level. Despite the benefits and efficiency of sketching new, emerging, and complex issues using this process, it represents an informed opinion or belief about the state of knowledge of a technical issue based on experts' training and background. This process uses simple rules of thumb to successfully guide the judgement of experts who have a large base of experience and knowledge to draw on. However, the potential for missing essential issues due to expert fatigue or bias exists. An approach to minimizing bias and expert fatigue is increasing and calibrating the amount of information considered.

One such calibration exists with the severe accident computer code benchmarking activity presented in IAEA-TECDOC-1727 [2] that determined the deficiencies in the knowledge from an analysis code perspective to predict the various phenomena confidently. The benchmarking made nine suggestions. The experimental studies on the CV behaviour beneath the molten pool and the end shield behaviour have closed the knowledge gap. One suggestion was for consistency in input parameters. The remaining suggestions from the benchmarking that deserve further consideration in the future were:

(a) A rigorous definition of the porosity and the reasons for its use in the analysis;
(b) The mechanism of steam generation after CV failure;
(c) The core collapse and disassembly modelling;
(d) The oxidation of debris under different configurations (suspended and terminal debris bed) to estimate hydrogen generation;
(e) The fuel channel rupture area that ruptures first by initial fuel heat-up; and
(f) The rate and species of fission product released from the terminal debris bed.

The planned activities in member countries are encouraging, notably the following studies fill the knowledge gaps identified in the PIRTs:

- Fuel channel integrity under asymmetric heatup condition under oxidising environment;
- CV blowdown;
- Core collapse;
- Hydrogen generation due to ingression of steam in to disassembled channels;
- Coolability of terminal debris within the CV;
- Heat flux exiting the CV during IVMR;
- Critical heat flux on the outer surfaces of the CV during IVMR;
- Corium plug formation;
- Corium crust mechanical properties;
- Gap formation (crust/ CV);
- Adequacy of prescribed SAMG actions.

In summary, all the completed and planned activities offer a healthy environment for simulating severe accident event sequences and assessing the safety of PHWRs.

4.6. CONCLUSIONS

The advancements in the severe accident experimental database are remarkable. The CV component experiments have improved the understanding of the vessel's failure concerning critical heat flux, and load distribution / thermal stress distribution. In the end shields, several studies have improved the predictive capabilities of the critical heat flux in a porous medium abutted by a tubesheet. The destructive tests on partial length CTs, rolled into a representative calandria tubesheet have improved the understanding of the component suspended debris, specifically on channel breakup. In the terminal debris and corium component, the phenomena like core collapse, heat transfer (convection by natural circulation, nucleate & film boiling, conduction, thermal radiation), including the model for molten pool heat transfer, melt relocation, and natural circulation in the melt have received significant investments in reducing the uncertain in the knowledge. Within the LCDA and SCDA domain, studies on the properties of CT of Zircaloy2 at high temperatures, fuel bundle integrity (LCDA), and exposed channel heat-up behaviour (SCDA) have made significant contributions to the knowledge database.

In addition, recent national and international research projects had led to significant progress towards management of combustible gases under accident conditions, including assessing the effect of different properties of H_2 and D_2 on the performance behaviour of hydrogen mitigation systems. However, some uncertainties remain, particularly in assessing combustible gases risk under representative boundary conditions for risk-relevant accident scenarios and hydrogen generation sources in the late phase of the accident.

The experimental programmes generate the fundamental data, providing the mechanisms for computer codes to become the repository of knowledge utilized by practicing safety analysts, ensuring the safety of nuclear installations. In an ideal scenario, the incorporation of lessons learned from experimental databases need to go hand in hand with model development in analysis computer codes. The following section presents an overview of the latest development in computer codes for analysis of severe accidents in PHWRs.

5. ADVANCES IN INTEGRATED SIMULATION CODES

Historically, integral computer codes have been developed for severe accident analysis. Those have been developed either specifically for PHWRs or adapted from a more generic LWR code to form an independent code applicable to PHWR. Those codes are used to assess the capability of the plant to prevent and mitigate accidents that are more severe than design basis accidents, and to support the development of severe accident management and corresponding actions of the emergency personnel. This section presents updates in PHWR specific integral severe accident computer codes in the PHWR MS of Canada, India and the Republic of Korea, including the incorporation of the findings from experimental programmes presented in Section 4. Those includes:

- Modular Accident Analysis Program (MAAP5-CANDU);
- PRogram for Anwik BHatti Apakarsh and VIkiran NIrdharan (PRABHAVINI);
- CANDU Advanced Integrated SEveRe code (CAISER).

In parallel, work has been conducted in PHWR MS to adapt or modify LWR severe accident computer codes to PHWR, namely:

- SCDAP/RELAP5;
- Accident Source Term Evaluation Code (ASTEC).

The description of these codes, their latest developments, models, limitations etc. are discussed in the following sections.

5.1. HEAVY WATER REACTOR CODES IN CANADA – UPDATES IN MAAP-CANDU

The section describes the changes to the MAAP4-CANDU code introduced during the upgrade of the code to MAAP5-CANDU, the current version in use in Canada [109-110]. MAAP is also used in other PHWR MS. The main features of MAAP4-CANDU have been described in previous IAEA TECDOCs [1] and [2], therefore, the following focuses on the new features and refinements since then.

5.1.1. Reactor core and fuel channel nodalization upgrades

During the upgrade of MAAP4-CANDU to MAAP5-CANDU, the following changes were implemented:

- Increased the axial core nodalization from 5 to 13 nodes;
- Increased the vertical core nodalization from 6 to 26 nodes;
- Increased the channel nodalization from 18 to (up to) 480 per loop;
- Increased the lateral core nodalization from 1 to 24 nodes per loop, i.e., added a third special dimension, (see Figure 30);
- Increased the number of channel axial power profiles from 1 to 480;

The two-dimensional MAAP4-CANDU suspended core debris model was refined with an additional lateral dimension in MAAP5-CANDU, allowing for a 3-dimensional representation of the CANDU core and significantly enhanced core disassembly and collapse modelling (see Figure 30).

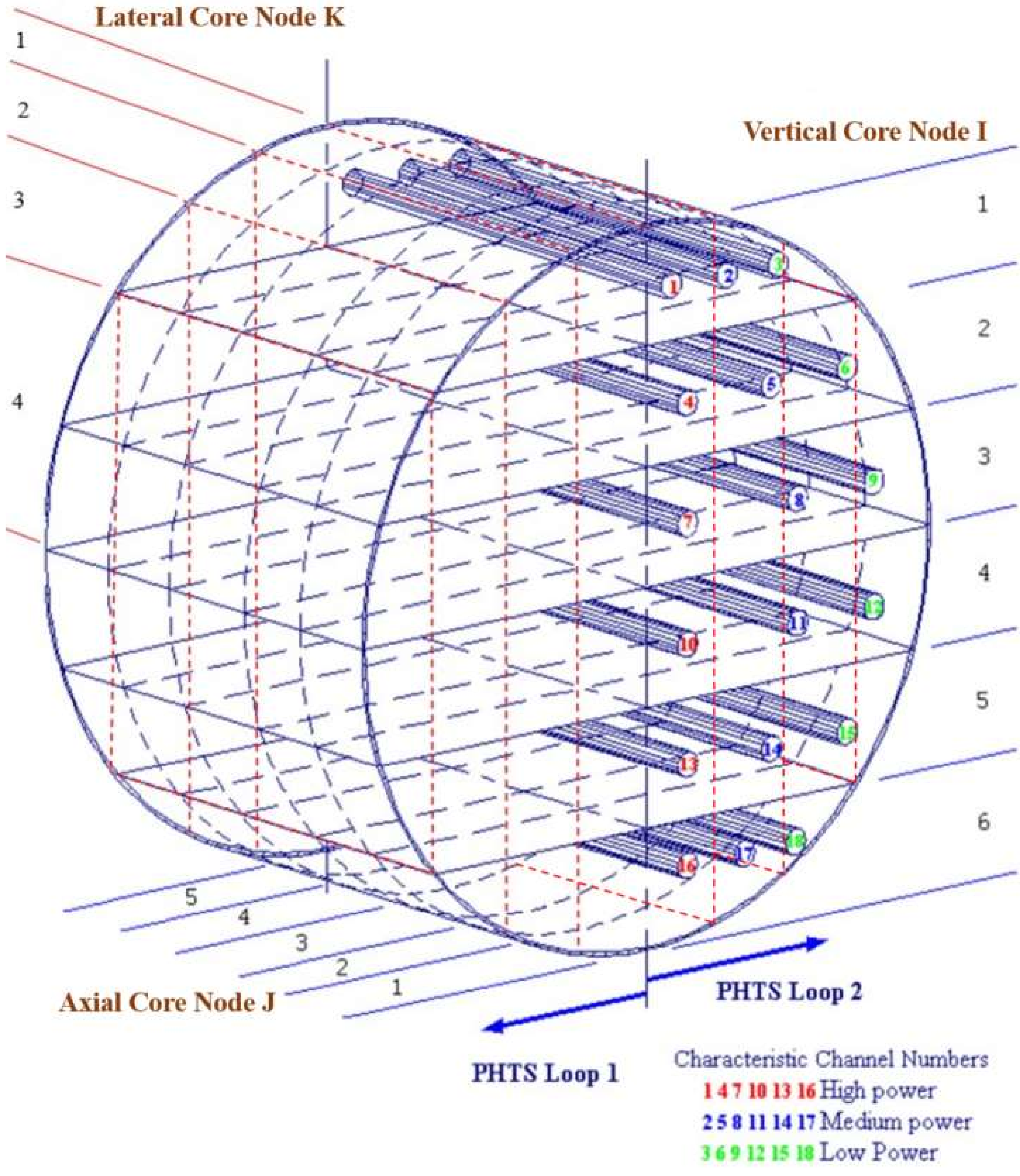


FIG. 30. CANDU core updated nodalization including new (lateral) dimension (courtesy of S. Petoukhov, CNL).

5.1.2. Core collapse model – new options

During an unmitigated severe accident, the moderator level in the CV would decrease, and the top rows of fuel channels would be uncovered and no longer cooled directly by the moderator. Due to the decay heat, fuel would heat up. When hot enough, the fuel channel (the PT and CT) and fuel would likely disassemble into large pieces of several bundles in length and would come to rest upon the underlying intact channels. Some smaller fragments plus some initially molten material would drop to the bottom of the CV where they would be quenched by any water present, but it is anticipated that a significant amount of the disassembled fuel channel sections (still containing fuel) would pile up as suspended debris.

If a severe core damage accident progresses further, the moderator level will continue to decrease, and more channel sections would disassemble to the suspended debris, while some suspended debris might melt and pour to the bottom of the CV. Figure 31 illustrates this partial disassembly configuration. Figure 32 shows a graphic of the postulated axial load distribution of debris and channel stubs during core disassembly is presented.

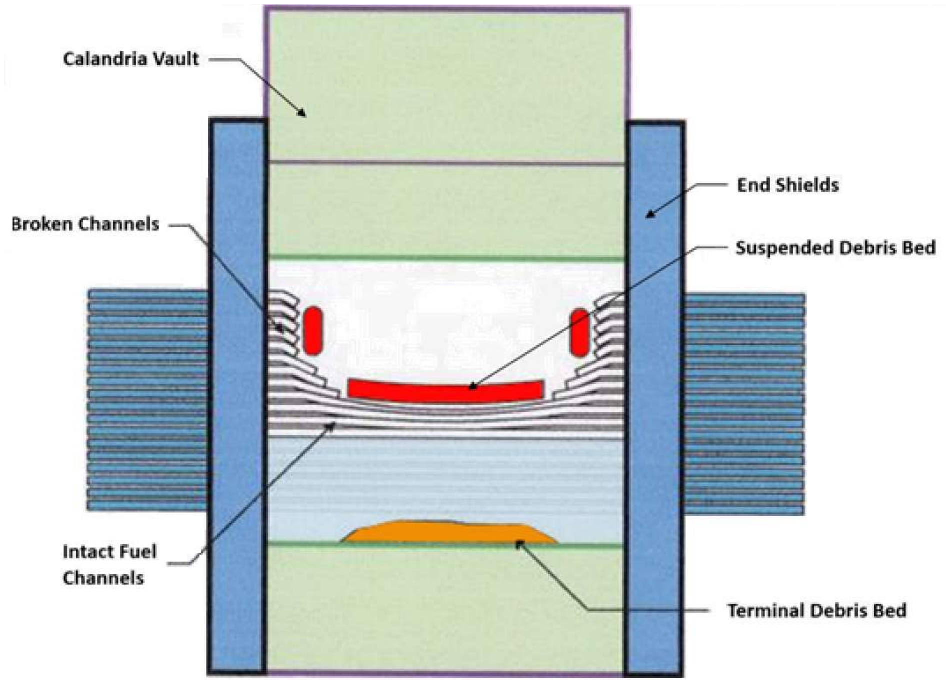


*FIG. 31. Side view of a partially disassembled reactor core (adapted from Ref. [111]***,** *courtesy of S. Petoukhov, CNL)***.**

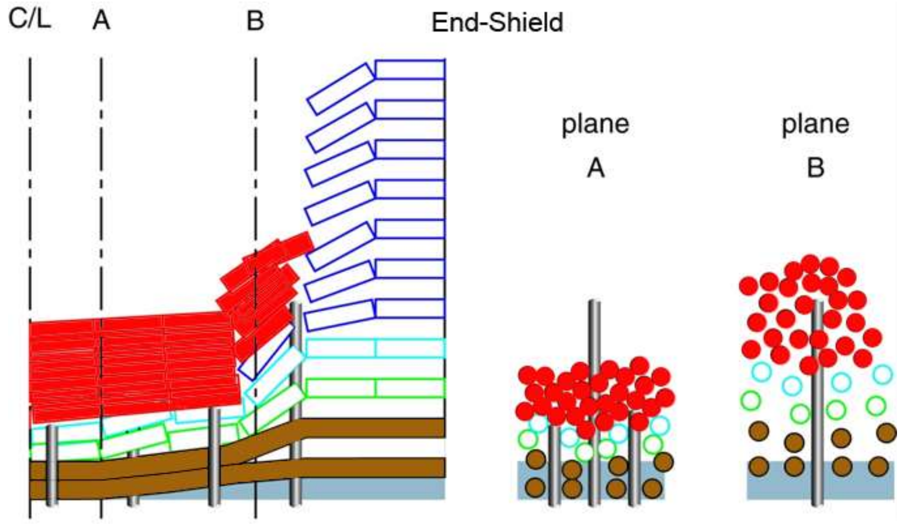


FIG. 32. Postulated axial load distribution of debris and channel stubs during core disassembly (courtesy of S. Petoukhov, CNL).

As the core disassembly progresses, eventually the top row of CTs still immersed in the moderator would fail due to an excessive load from:

- The suspended channel debris;
- The sagged / partially disassembled channels above;
- The fuel and PT inside the top channel.

The CT failures might occur due to CT rolled joint pull out / twist out, CT shear failure, CT axial strain (necking), CT buckling due to bending moment [74]), or a combination of causes, all of which would be induced by the load.

When a row of CTs fails (or perhaps only a few well-cooled CTs, depending upon the load distribution) the collection of debris would move downwards, along with the newly failed channels. The next row of supporting CTs would have to support a larger mass of debris, so it is conceivable that the core would collapse in a cascading fashion to the bottom of the CV. Figure 33 illustrates this core collapse in its right-hand portion. This process is usually called "core collapse". Since the CANDU core consists of two symmetrical halves, it is also possible that the channels of one core half (i.e., one loop) could disassemble and collapse, while the channels of the second half remain intact, provided the channels in the second half were still internally cooled by primary coolant.

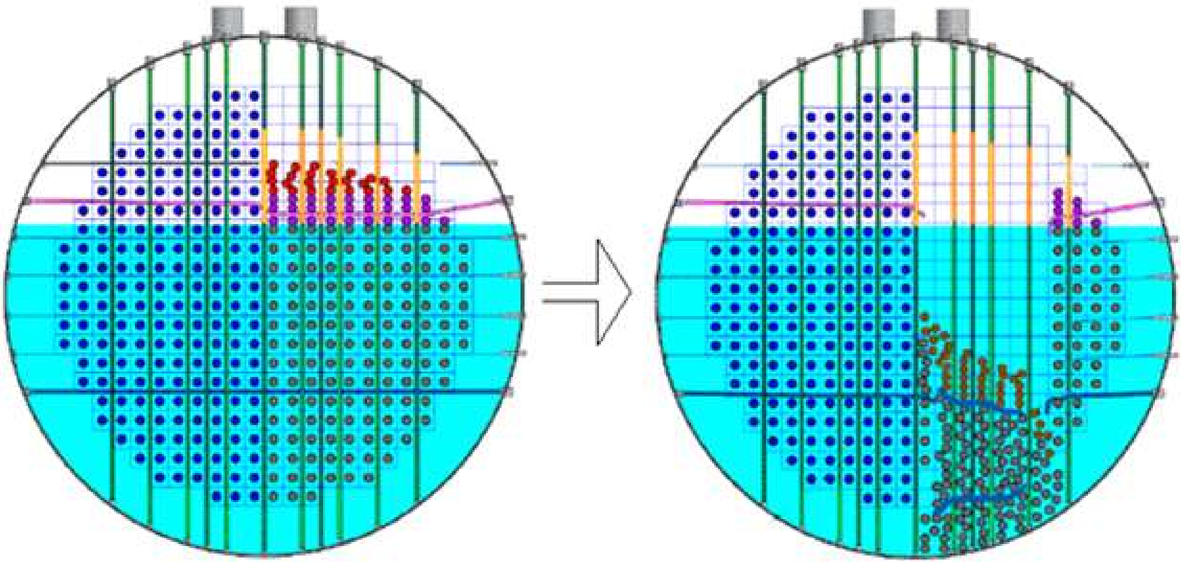


FIG. 33. Core collapse of half CANDU core (adapted from Ref. [111], courtesy of S. Petoukhov, CNL).

MAAP-CANDU models the collapse of the core in a parametric manner. Historically, there are two core collapse models, and they are retained in the current MAAP5-CANDU code:

- The original ("classic") parametric model initially introduced, and retained in the code as a sole option up to the latest revision of the MAAP4-CANDU code;
- A semi-mechanistic model added in MAAP5-CANDU.

5.1.2.1. Original ("classic") core collapse model

In MAAP4-CANDU, core collapse is triggered, when all three of the core-collapse criteria are met (in the current PHTS loop):

(1) The number of vertical core nodes not covered by moderator is greater than or equal to user input parameter NDCLPS;
(2) The average CT temperature for uncovered axial core bundles in the medium and high-power channels is greater than user input parameter TCTCLPS;
(3) The total mass of suspended debris in a PHTS loop exceeds the value defined by the user-input parameter (MLOAD). Note that for MAAP5-CANDU, the MLOAD parameter is now an array based on the number of lateral nodes with a specific value MLOAD (k) for each of the 'k' lateral nodes.

The first two criteria are normally not used, therefore, MLOAD becomes the primary core collapse initiation criterion.

5.1.2.2. New (semi-mechanistic) core collapse model

The classic core collapse model is parametric – collapse occurs simply because the mass of suspended debris in a loop or lateral node exceeds the user-defined mass MLOAD. MLOAD is determined by the user depending upon the reactor design and the scenario being modelled.

Beginning in 2009, AECL (now CNL) developed a semi-mechanistic model of core collapse, initially known as the "Dynamic MLOAD" model. This model, further developed and briefly described in reference [109], assumes the underlying cool and strong CTs were simple uniformly-loaded cantilevered beams, supported at either end by the tube sheet rolled joints (see Figures 31 and 32). The vertical transverse load on the CTs imparts a bending moment to each supporting CT, and the maximum moment occurs at the tube sheet. The model calculates the maximum tensile stress, induced by the maximum bending moment at the top of the CT. When the stress exceeds the ultimate tensile strength, the CT is assumed to tear and lose its ability to support the debris.

This model takes the moderator level into account, and thus the width (number) of channels available to support the debris and estimated the loads from sagging channels and from the fuel within the supporting channel.

The semi-mechanistic core collapse model was incorporated into the MAAP5-CANDU code [109], and the input parameters to this model are now included in the parameter file. This implementation takes advantage of the increased core nodalization in MAAP5-CANDU, which allows better determination of the specific potential loads on the chosen lateral nodes based on those channels no longer submerged in the moderator. In MAAP5-CANDU, the core collapse calculation is computed for each of the user defined lateral nodes in each loop of the core.

The relationships used in the semi-mechanistic core collapse model were derived from elastic beam theory and are described in reference [109]. Key features of this model are presented below.

Identification of the supporting fuel channel row

In MAAP5-CANDU, new coding has been added to the source code to dynamically determine the channel row that supports the hot sagging channels and suspended debris above. As the moderator water level drops below the current supporting row, the CT in this row will begin to heat up. Once the average CT temperature in the current supporting row exceeds a user-defined value, the supporting channel row is moved down to the next channel row.

Calculation of the ultimate tensile strength of the calandria tube

The ultimate tensile strength of a CT at a given temperature is calculated based on the correlation shown in Equation (2). To make the correlation more flexible a user defined coefficient (FUTSCOEFF) was introduced.

$$\sigma_{UTS\ i,k} = \sigma_{UTS0} - FUTSCOEFF \text{ x } (TCT_{Ave\ i,k} - 273.15) \text{ [MPa]} \quad (2)$$

In Equation (2), σ_{UTS0} is the ultimate tensile strength of a CT when at 0ºC, FUTSCOEFF is a user defined coefficient, and $TCT_{Ave\ i,k}$ is the average CT temperature within a vertical core node *i* and lateral column *k*.

Calculation of the maximum supportable load

Given the ultimate tensile stress, the maximum load on a single CT in a given vertical/lateral core node (i.e., *i* and *k*), $W_{chanmax}$, can be evaluated using Equation (3):

$$W_{CHAN\ MAX\ I,K} = \frac{I_0 \sigma_{UTSi,k}}{R_{CT0}} \left[\frac{12L}{L^2+2aL-2a^2}\right] \quad \text{[N]} \quad (3)$$

Where *L* is the length of the CT, *a* is the length of one side of the CT which is assumed to be unloaded (see Figure 34), R_{CTo} is the outside CT radius, *Io* is the moment of inertia of an area. In the case of a CT with inside and outside diameters *DCTi* and *DCTo*, the moment of inertia of an area *Io* is as defined in reference [112] and presented in Equation (4):

$$Io = (\pi/64) (D_{CT0}^4 - D_{CTi}^4) \quad (4)$$

Thus, the code computes a normalized maximum bending moment (i.e., at the tubesheet) of the same total load W with varying load distribution (see Figure 34).

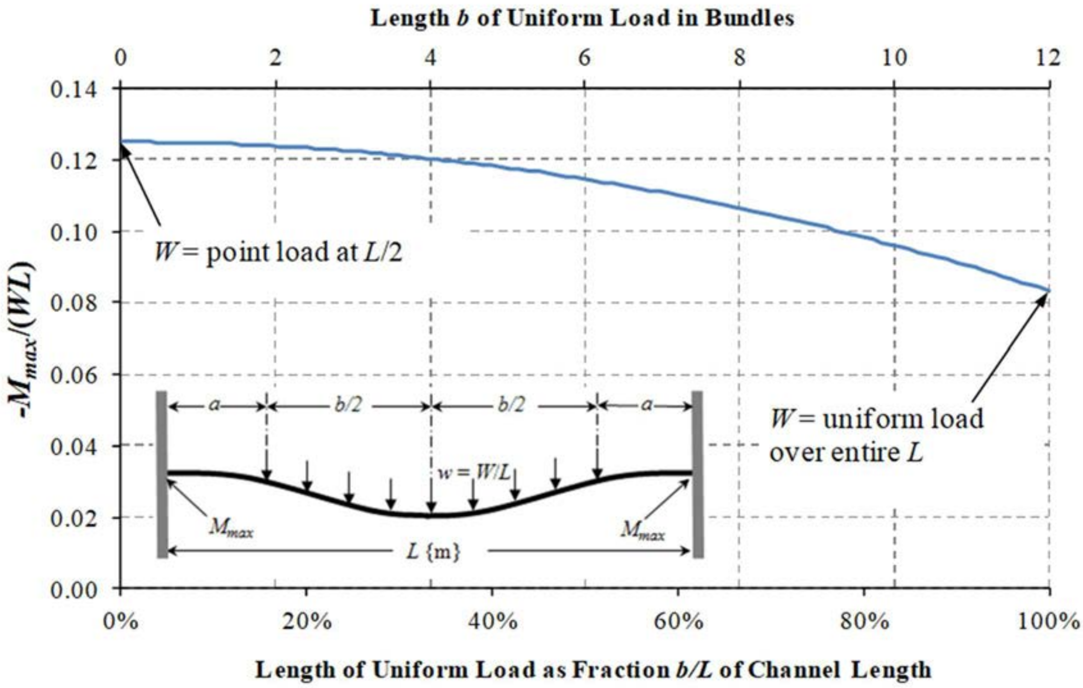


FIG. 34. Normalized maximum bending moment (i.e., at the tube sheet) of the same total load W with varying load distribution (courtesy of S. Petoukhov, CNL).

Calculation of the maximum total supportable mass

The maximum total supportable mass in a specific vertical/lateral core node is calculated by multiplying the maximum total supportable load from a single channel (which is a function of the average CT temperature of the channels in the vertical/lateral node) by the number of supporting channels in the node and dividing by the acceleration of gravity (g), as shown in Equation (5):

$$M_{sup\ i,k} = W_{chan\ max\ i,k} \times N_{sup\ i,k} / g \qquad (5)$$

Where $N_{sup\ i,k}$ is the number of supporting channels in a given vertical core i and lateral core node k.

Mass of intact channels sagged on supporting row

In order to evaluate the load of intact channel segments which have sagged into contact with the supporting row of channels submerged into the moderator, the temperature of each axial channel node in the channels that are located above the supporting row is compared to a user-specified input temperature above which the CT strength is significantly reduced (and thus the CT and fuel channel would sag). If the temperature of the axial channel node exceeds this user-defined sagging temperature (variable TSAGROW) then the mass of this axial segment is added to the total load that needs to be supported. Another user-specified MAAP-CANDU input parameter (NSAGROW) limits the number of channel rows that can contribute to the total load. This value is typically be of the order of 2 or 3 actual lattice pitches as it is not feasible for a channel to sag more than this distance without breaking apart.

Mass of internal supporting row

In addition to the mass of suspended debris and mass of intact sagging channels, the load of the supporting row itself is accounted for. A reduction in the supporting row load due to buoyancy forces is not evaluated as it is considered to be significantly smaller than the sum of the internal and external loads.

Determination of core collapse in MAAP5-CANDU

The final step when checking conditions for core collapse is to compare the total supportable mass to the mass of the suspended debris plus mass of the intact sagging channels plus the internal mass (i.e., fuel and sheath, PT and CT, etc. inside the fuel channel). If the total mass exceeds the supportable mass, then a core collapse is predicted. This calculation is performed for each lateral core node (k) in each PHTS loop as per the core nodalization chosen by the MAAP5-CANDU analyst.

5.1.3. Calandria vessel/debris modelling improvements

In upgrading the code to MAAP5-CANDU, some of the CV and debris models received minor updates compared to their implementation in MAAP4-CANDU. The refinements are as follows:

- Convective heat transfer from the molten oxidic pool to the solid crust:
  A new correlation was added based on the experiments conducted by Bonnet and Seiler with the BALI facility [110];
- Heat transfer through the molten steel layer in the terminal debris bed:
  A new correlation was added that considers the effect of axial and lateral heat transfer;
- Heat transfer from the molten steel layer (in the terminal debris bed) to the vessel wall:
  New correlations (Laminaire, Churchill and Chu) were added to calculate the heat transfer coefficient from metal layer to vessel wall [113];

5.1.4. Containment model changes

There were some updates and refinements to the containment modelling made between MAAP4-CANDU and MAAP5-CANDU. In the containment, the heat transfer coefficients for radiation, natural convection and forced convection to a surface of various heat sinks and walls can be calculated using various user-defined options.

The forced convection model is as it is used in the Wisconsin flat plate benchmark. The modified Reynolds analogy is used for both laminar and turbulent flow to calculate the heat transfer coefficient using the Stanton number.

The default natural convection heat transfer coefficients are determined through the use of a heat and mass transfer analogy. The natural convection heat transfer coefficient is calculated by correlations of the form for the average Nusselt number. Alternately, the Tagami (Reference [114]) and Uchida (Reference [115]) correlations can be used by setting the corresponding user-input flags: ITAGAMI=1, or IUCHIDA=1.

The most important updates between MAAP4-CANDU and MAAP5-CANDU consisted in the addition of the Tagami and Uchida correlations. A model for re-vaporization of liquid film on containment heat sinks was also added.

Other miscellaneous changes in containment modelling were implemented in MAAP5-CANDU. Those include:

- The maximum number of containment nodes was increased from 40 to 120;
- A new, more flexible model for PARs was added;
- Suspended water treatment is now included in MAAP5-CANDU, so it can now model water droplets in containment from:
  - Bulk condensation;
  - Two-phase LOCA blowdowns;
  - Spray system operation;
- A new diffusion flame model in containment water pools and in containment junctions was added;
- Momentum induced containment node flows are now accounted for, including:
  - Flows resulting from discharge of single or two-phase steam-water jets;
  - Water injection from containment water spray system;
  - In addition, the velocity in a node is used to solve rates for heat, mass and momentum transfer from the gas space to the heat sinks.

5.1.5. Implementation/integration of the MAAP-DOSE into MAAP5-CANDU

A dose model has been developed as part of the MAAP5-CANDU code to provide information on the dose consequences of severe accidents in CANDU plants. The current dose model in MAAP-CANDU is an upgraded version of the stand-alone dose model in the MAAP-AST (Alternative Source Term) code, a special version of MAAP-LWR for performing design basis AST estimations. The MAAP-AST code was developed from the original MAAP4-DOSE as a special version of the MAAP4 (LWR) version 4.0.4+ tailored for design basis AST applications.

The dose model is now a fully integrated component of the MAAP5-CANDU code (as well as the MAAP5-LWR code), enabling the code to calculate radiation dose simultaneously with the overall calculation of the progression of an accident.

MAAP-CANDU DOSE offers the unique capability to calculate radiation doses both internal as well as external to the plant, during the accident progression. The internal/on-site dose estimations can be computed for the control room, fuel handling room, stairwell, pump room, and any other area of significance where the plant operators and emergency personnel may be present or need to be present during the severe accident (i.e., in order to execute mitigating actions). Similarly, for the external/off-site dose assessments, estimations can be made for the exclusion area boundary, low population zone, shelter zone, and other areas where populations may reside.

There are two dose estimation techniques in the dose model, which allow the estimation of doses to equipment, site personnel, and the public during the modelled accident progression. These techniques are based on the AST and Point Kernel Method.

- The AST based dose model has been developed as per the US NRC's Regulatory Guide 1.183 [116] and can be applied for both in-plant and ex-plant dose estimations. The dose estimation using the AST method calculates the committed effective dose equivalent from inhalation, deep dose equivalent from external exposure, and the total effective dose equivalent, which is the sum of committed effective dose equivalent and deep dose equivalent;
- The Point Kernel Method based dose model relies on the traditional fluence to dose and mass attenuation coefficients [117].

In order to turn on the dose model in MAAP5-CANDU, users can set flag IM4DOS = 1 and define the MAAP-DOSE parameter file in an input file using a "DOSE PARAMETER FILE" statement in the input deck files.

5.1.6. Fission product model improvements

One of the processes modelled during the severe accident progression is fission product releases from fuel into the reactor vessel, into containment and, eventually, into the environment. Amongst the FP released, iodine (in both inorganic and organic chemical forms) can be highly volatile even at lower temperatures and can significantly contribute to the potential radiation doses to the surrounding population. There were some upgrades to the FP release models from the fuel when moving from MAAP4-CANDU to MAAP5-CANDU, specifically the addition of new models. The Appendix III p.167 presents a supplement to the MAAP-CANDU description that details the fission product model in the code as well as presents the upgrades implemented in MAAP5-CANDU.

5.1.7. Addition of a graphical user interface in MAAP5-CANDU

For MAAP5-CANDU, a Microsoft Windows application, or Graphical User Interface (GUI) was added to manage running multiple simulations instead of the MS-DOS interface used for MAAP4-CANDU, which allowed to handle only one simulation at a time.

5.1.8. EPRI MAAP6-CANDU modernization

As part of continued modernization efforts, the Electric Power Research Institute (EPRI) (who owns the MAAP and MAAP-CANDU codes) has proposed to consolidate all existing versions MAAP (PWR, Boiling Water Reactor (BWR), CANDU, Water-Water Energetic Reactor (VVER)) into a singular codebase. This integration would allow significant reductions in code maintenance costs and unify the code release cycle.

This modernization/unification task is realistic and practical for the MAAP-CANDU code as it generally fits into the MAAP-LWR modularization philosophy. MAAP-CANDU has both CANDU plant modules (for unique components such as horizontal fuel channels, CV, shield tank, etc.) as well as modules common with MAAP-LWR (such as containment, FP releases, hydrogen behaviour, etc., see Figure 35 and Figure 36 below). It is also planned that the programming language will be changed in MAAP6 (and hence in MAAP6-CANDU) from Fortran to C++.

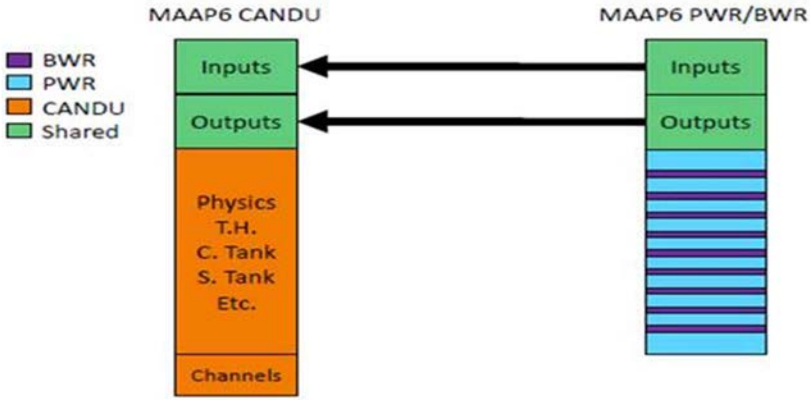


FIG. 35. Schematic representation of MAAP6-LWR and MAAP6-CANDU input/output data and code structure – current status (courtesy of EPRI).

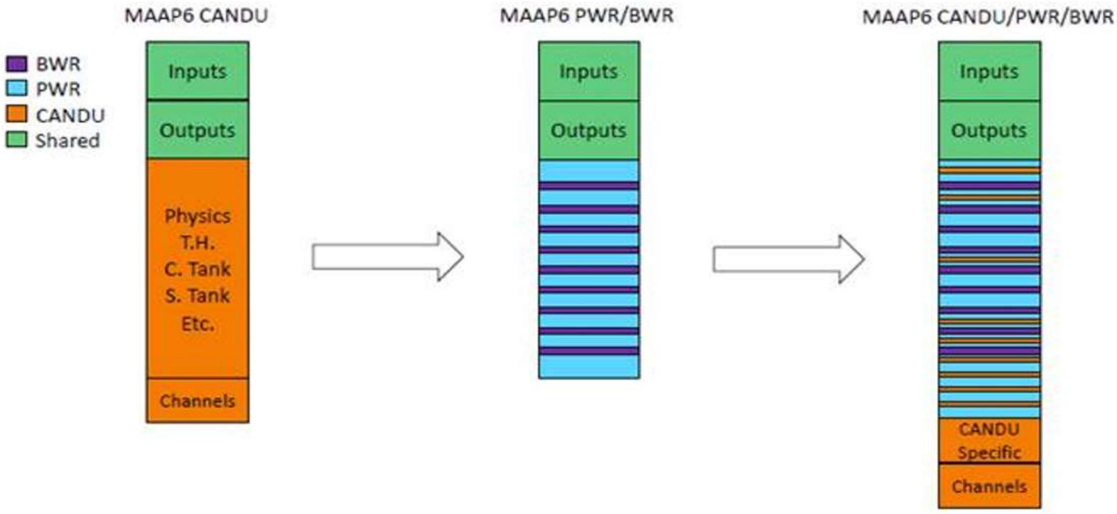


FIG. 36. Schematic representation of MAAP6-LWR and MAAP6-CANDU input/output data and code structure – future steps (courtesy of EPRI).

Figure 37 shows the overall MAAP-CANDU code structure, and MAAP-CANDU Channels System modularization can be found in Fig. 3.2 of the IAEA-TECDOC-1727. The CANDU Owners Group (COG) owns MAAP-CANDU Channels System Code, while the overall MAAP-CANDU code is owned by EPRI. It is planned that MAAP6-CANDU will be distributed/maintained by COG under the overall EPRI oversight. A beta version of MAAP6-CANDU is scheduled for release in 2024, with a final version expected in 2025.

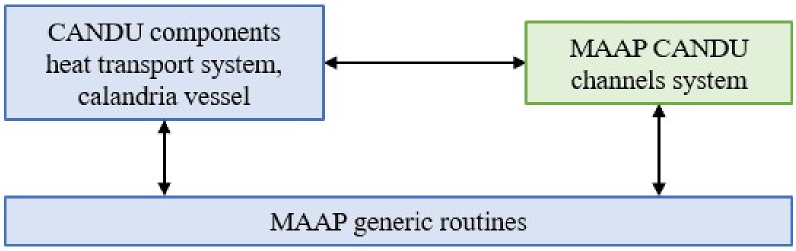


FIG. 37. The MAAP-CANDU code architecture.

5.2. HEAVY WATER REACTOR CODES IN INDIA – DEVELOPMENT OF PRABHAVINI

PRABHAVINI code is an integrated computational tool being developed by BARC as the lead developer along with participation from other units of the Department of Atomic Energy and the educational institutes in India [118]. The code serves the purpose of performing safety analyses of normal and postulated off-normal conditions of power and research reactors along with public risk assessments for design and licensing purposes. The code supplements the multi-step methodology described in IAEA-TECDOC-1594 and IAEA-TECDOC-1727.

The code is generic enough to accommodate all the fission reactor configurations of present reactors (PHWRs, VVER, BWR, and Advanced Heavy Water Reactor) and future reactors (Small Modular Reactors, High Temperature Reactors and Molten Salt Breeder Reactors) configurations, as planned under Three Stage Nuclear Power Program of Department of Atomic Energy. The code is also helpful to provide an aid to design an experimental test facility as well as simulation of experimental test facilities for code benchmarking purposes. Currently the code is matured enough to carry out the following analyses:

- Analysis of present nuclear reactor (LWRs, PHWRs, fast breeder reactors) of different states (normal operation to DECs without fuel melt conditions);
- Uncertainty analysis;
- PSA level-2 and level-3 analyses;
- Accident management analyses.

The other usage of the code in its present form are limited application to fusion reactors/facilities safety assessment and operational transients of solar thermal power and conventional thermal power plants.

5.2.1. Overall structure of PRABHAVINI

PRABHAVINI makes use of a modular structure, wherein, the engineering modules pertaining to specific phenomena can be developed independently of other modules. Figure 38 shows a schematic of the different engineering modules presents in PRABHAVINI v3.0. A brief description of the major engineering modules follows.

PRAVAH: The module solves one-dimensional, two-phase, six equation model for fluid flow in the piping system. Closure relations between the two phases (steam and water) for mass, momentum and energy are accounted for several vertical and horizontal flow regimes. The heat transfer closure is based on pre- and post-departure of nucleate boiling heat transfer regimes. PRAVAH employs a pressure-implicit formulation. Any number of thermalhydraulic systems and various types of fluid (for e.g., water, sodium coolant, non-condensable gases) can be defined and explicitly coupled in a plant model. The model also facilitates modelling of special components such as pressurizers and accumulators. It also incorporates special flow models such as choked flow models, counter current flow limitation models which can be invoked by the user, wherever necessary.

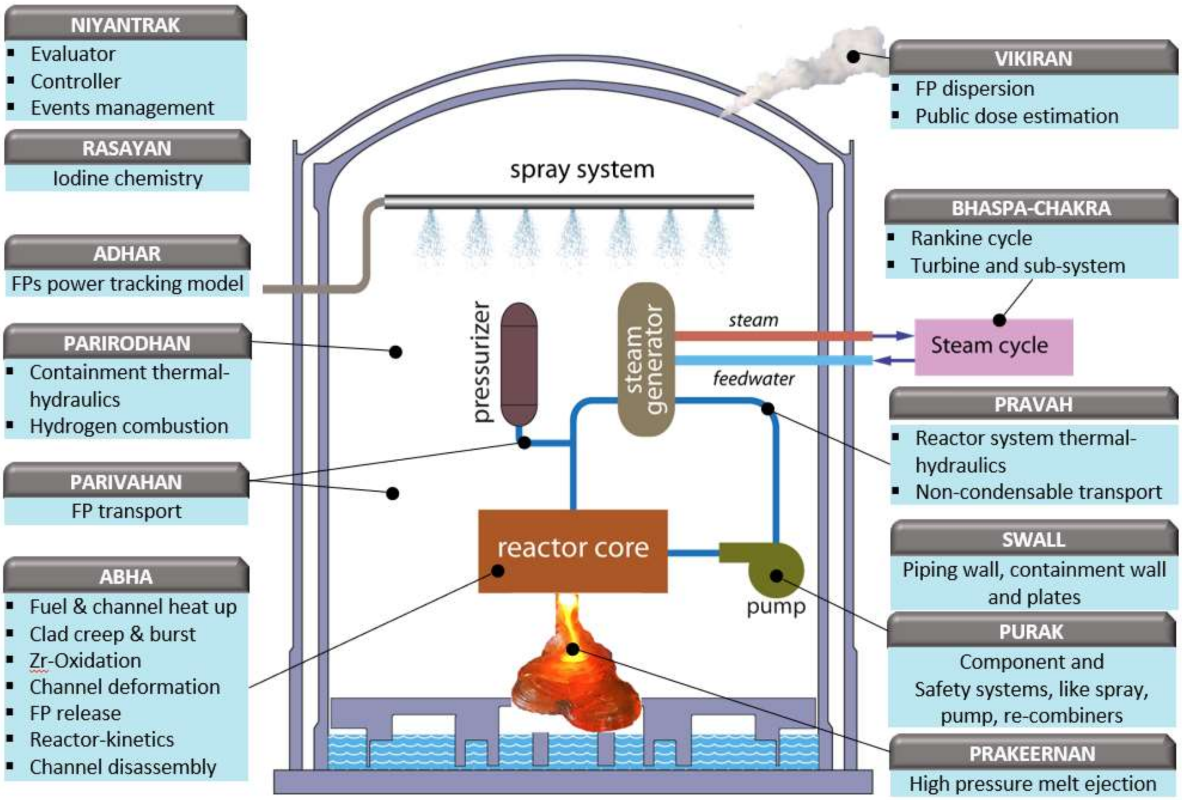


FIG. 38. Schematic of PRABHAVINI modular structure (courtesy of O. S. Gokhale, BARC).

SWALL: Piping wall, plates and containment walls can be modelled using this module. It solves one dimensional heat conduction equation in the radial direction through implicit formulation. The heat transfer from the SWALL component to PRAVAH fluid component is handled by the heat transfer package present in PRAVAH module.

ABHA: The module is responsible for reactor core component modelling. It solves 2D heat equation in the structural components, power generation using point kinetics model, multi-point kinetic models and its distribution in the core. It also does mechanistic modelling of core degradation, such as cladding burst, PT sagging and ballooning and calculates fission product release from the failed fuel pins. The module models radiation heat transfer between the fuel pins, between fuel channels and from the core to the CV.

PARIVAHAN: The module manages the retention and transport of 84 different FPs and actinide in the primary heat transport system and containment system. In the containment, iodine is considered to be in the form of aerosol, molecular and organic forms. The rest FPs (volatile, semi volatile and refractory) are considered to be in aerosol forms. The AERB's draft safety manual on Radiological Impact Assessment Guideline [119] has been followed to build the module. In addition, a detail mechanistic model has been developed and implemented.

PARIRODHAN: This module is responsible for containment system model [120]. It comprises of two sub-modules - 1) VAHATI (flow): the module simulates containment thermal hydraulics in one dimension along with wall condensation. It also solves hydrogen and non-condensable transport in the system, and 2) DAHAN (combustion): the model simulates combustion of hydrogen and generation of energy and steam using the Adiabatic Isochoric Complete Combustion model.

PURAK: This module includes various system components and safety systems of NPPS such as centrifugal pump, valves, passive auto-catalytic re-combiners for hydrogen mitigation and containment engineered safety features such as the primary containment controlled discharge and secondary containment filtration, re-circulation and purge system.

NIYANTRAK: The module deals with the control functions which users would like to perform based on parameters that are calculated during run time. NIYANTRAK performs fundamental tasks of measuring certain run time parameters, performing mathematical and logical calculations, taking decisions based on outcome of these calculations and executing actions based on the decisions taken. NIYANTRAK can access data and perform actions across all modules of PRABHAVINI.

RASAYAN: In the event of severe accidents, the release of iodine is a major concern. Large inventory of iodine in irradiated nuclear fuel, a half-life 8.04 days, complex chemistry, formation of volatile species, hazardous biological effects lead to potential isotope for radiation risk. In the event of a core meltdown, a significant portion of iodine within the core may be discharged into the containment in the form of aerosols, encompassing metal iodides and gaseous iodine. Gaseous molecular iodine is depleted through trapping on the containment surface like steel, concrete and paint. Also, the gaseous molecular iodine reacts with air radiolysis products. These processes are modelled to quantify the various iodine species. The module covers about 40 reactions that are the most prominent ones which take place in the gaseous and liquid phases.

VIKIRAN: The module estimates atmospheric dispersion, ground deposition of radionuclides, cloud immersion and dose to the public through different food chains. Gaussian Plume model is used to evaluate the atmospheric dispersion of radionuclides using joint frequency distribution of wind speed class, stability class and wind sector. The ground deposition due to dry deposition is calculated using source depletion model. The module also offers dose estimation with site counter measures.

PRAKEERNAN: High pressure melt ejection from pressurized reactor pressure vessel for LWRs and its dispersion into the containment is an important aspect of direct containment heating event that needs to be modelled for analysis of severe accident scenario. A computational model named PRAKEERNAN (dispersion) is developed to address the complex processes involved during high pressure melt ejection and dispersion. The model simulates ablation of the reactor pressure vessel breach, high pressure molten corium ejection followed by molten corium fragmentation and dispersion. Entrainment physics has been applied to the molten corium (U-Zr-Fe-O eutectic) for dispersion of melt into the different compartments of the reactor building. Convective and radiative heat transfer from the corium along with exothermic metal-steam reactions have been accounted in the model. The model has been validated against two DISCO-H experiments performed at Karlsruhe Institute of Technology, Germany.

BASHPA-CHAT: This is a generic steam cycle model developed with a focus on adaptability and flexibility to handle the systems that use superheated/saturated steam for simulating steam cycle. The module can be applied to different types of power plants for analysing plant dynamics of secondary side of nuclear and thermal power plant. This model has a modular structure, in which the stand-alone modules for simulating different sub-systems of steam cycle are developed and integrated in such a way that users can plug in and combine the necessary modules suitably as per the requirement via a connector module. Presently the stand-alone module is developed for steam throttling, high pressure turbine, moisture separator, reheator, low pressure turbine, condenser, pump, feed water heaters and de-aerator systems.

5.2.2. Application of PRABHAVINI modules for heavy water reactors

The PRABHAVINI modules described in the previous section can be used to model different components/phenomena pertaining to PHWRs. The thermal hydraulics in the PHTS is modelled using PRAVAH module and the piping walls are modelled using SWALL module. Figure 40 shows a typical nodalization of the PHTS for an Indian PHWR.

The reactor core, fuel channels, fuel behaviour and the core degradation are modelled using ABHA module components. Figure 39 shows a typical nodalization for the ABHA components. ABHA module also facilitates creation of zones that divides the reactor core in Y-Z plane. Figure 41 illustrates this aspect. The channels can be arranged within the zones. This arrangement helps in identifying the channels participating in radiation heat exchange in a particular configuration, modelling failure of channels based on criteria discussed in IAEA-TECDOC-1594 [1] and relocating the failed channel to appropriate zones.

The current version of PRABHAVINI can simulate LCDA phase of accidents. The PT creep deformation such as ballooning creep is modelled with the help of experimental transverse creep correlations assuming that PT volume remains constant, its shape remains circular, and effect of axial stress is negligible. The sagging deformation model accounts for the PT support at both ends (fixed or simple support condition), the supports provided by garter springs and the fuel bundle loads. Elastic bending due to mechanical loads, thermal expansion and deformation due to thermal creeps are considered. Formation of contact between PT and CT during sagging is also considered as simple support during the calculation.

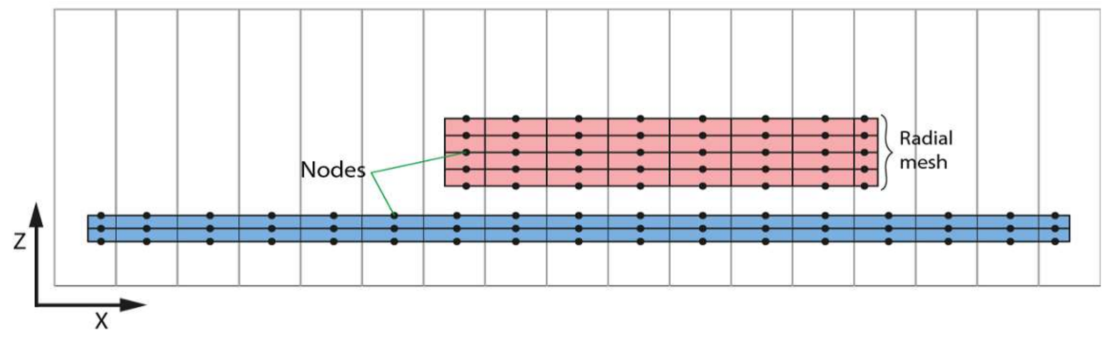


FIG. 39. Nodalization for structural components within the reactor core created using ABHA module (courtesy of O. S. Gokhale, BARC).

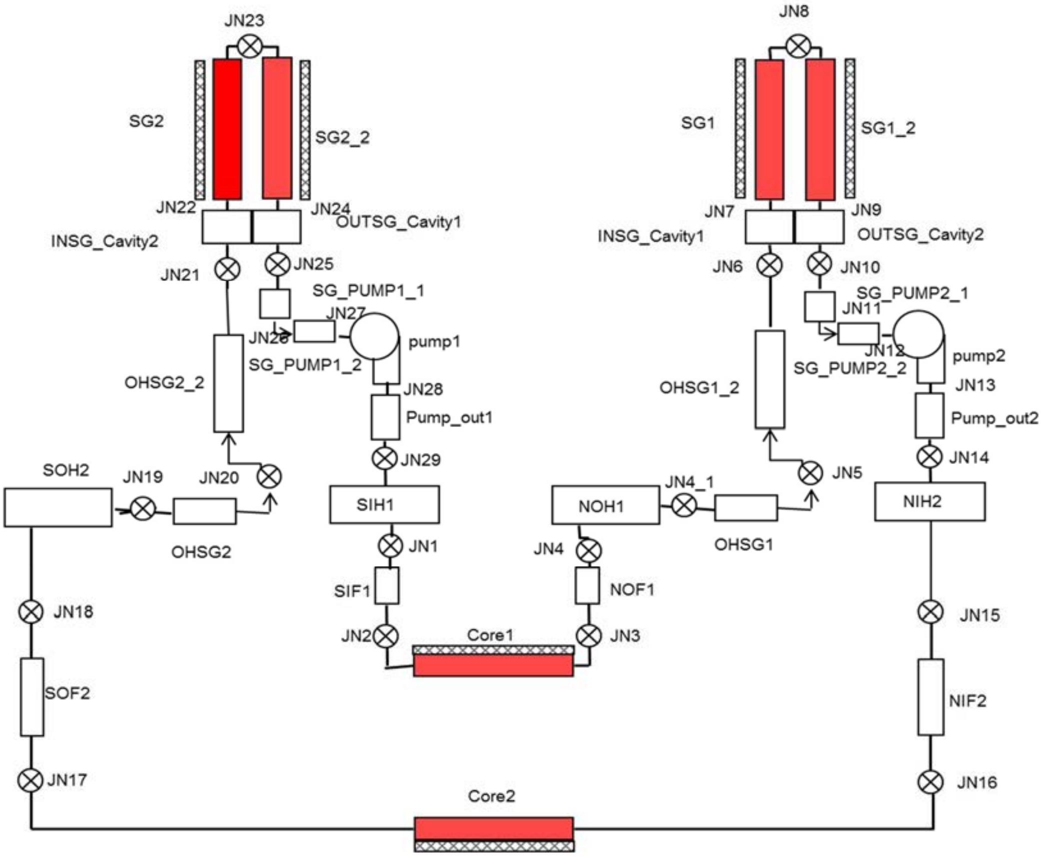


FIG. 40. Typical nodalization of PHTS of Indian PHWRs using PRAVAH module (courtesy of O. S. Gokhale, BARC).

The heat transfer between PT and CT post PT-CT contact makes use of experimental derived contact conductance model such as the model by Shewfelt [121], a model developed in India [122] and a semi-empirical correlations by Yovanovich [123]. The contact pressure is evaluated based on special interpolation scheme as a function of distance between the PT and CT represented as contact zone spanning over contact length and contact angle (see Figure 42). The transition from LCDA phase to SCDA phase of accident is under development.

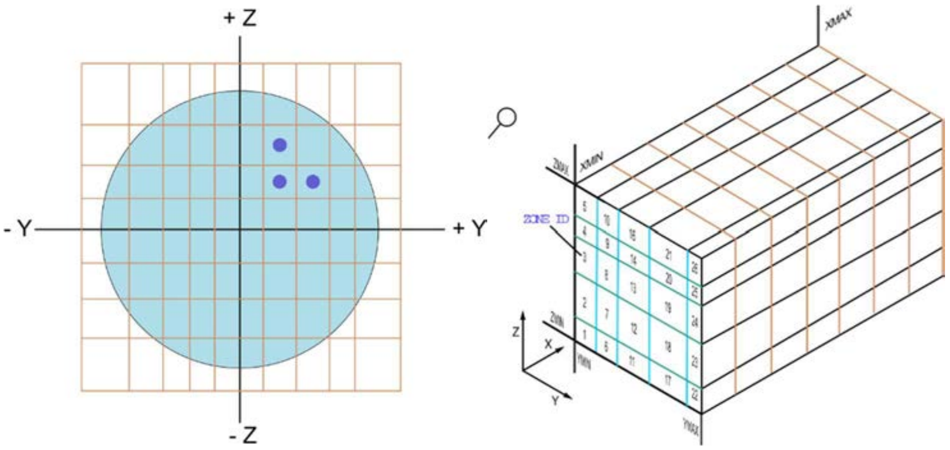


FIG. 41. Creation of zones to divide the reactor core into zones Y-Z plane (courtesy of O. S. Gokhale, BARC).

The containment thermalhydraulics is modelled using PARIRODHAN module. The secondary circuit which comprises of steam generator secondary side, turbine, condensers, re-heaters, boiler feed pumps is modelled using BASHPA-CHAKRA module. The release of fission product inventory from the fuel, its transport in the PHTS and containment is modelled using PARIVAHAN module. The iodine chemistry is handled by the RASAYAN module.

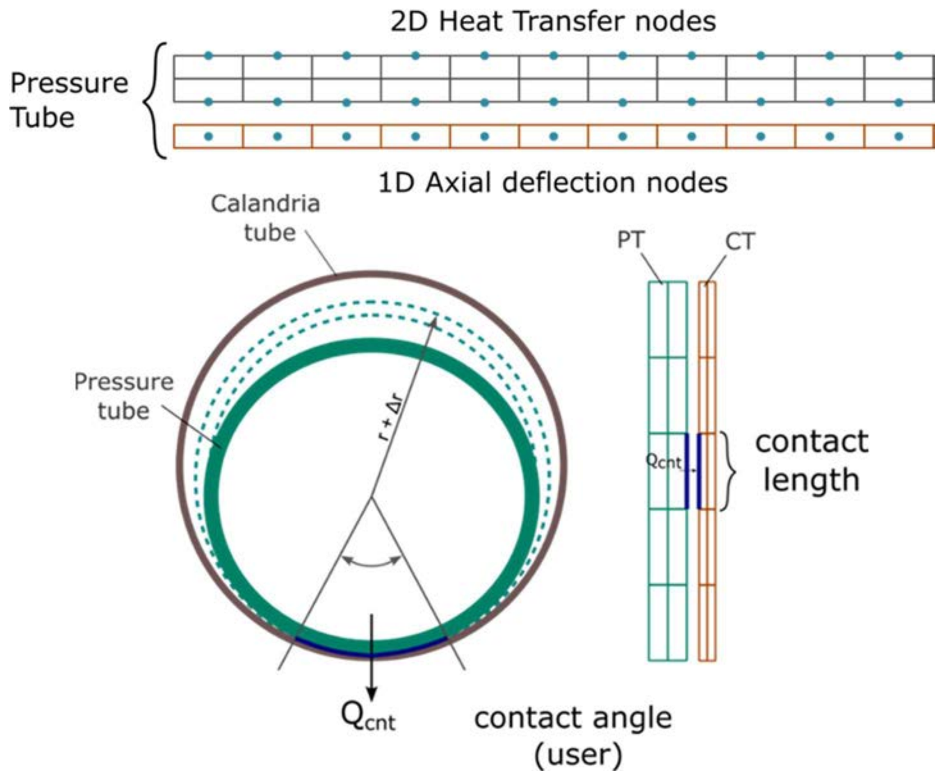


FIG. 42. Nodalization use for heat transfer within PT and from PT to CT (courtesy of O. S. Gokhale, BARC).

5.2.3. Verification and validation

As a part of code quality assurance programme, a V&V programme exists. This programme is in line with the guideline provided by AERB SG/D-19 Guide [124] and international practices for safety related software (ASME NQA-1). Under the V&V programme, the code modules have undergone various benchmark exercises which includes experimental validation for phenomena such as sub-cooled and saturated blowdown, natural circulation flow, flow boiling in pipes of various orientations, vent clearing of suppression pool, radiation heat transfer from the fuel pins to enclosures, oxidation kinetics, as well as numerical (code to code comparison) and analytical validation [125]. Nevertheless, a large number of exercises still need to be carried out to understand the shortcomings of the model to capture various phenomena as well as the computational efficacy of the code with respect to international codes.

5.3. HEAVY WATER REACTOR CODES IN REPUBLIC OF KOREA – DEVELOPMENT OF CAISER

Following the Fukushima Daiichi NPP accidentin 2011, the Republic of Korea implemented various emergency measures, one of which was the introduction of new legislation specifically designed to address severe accidents in nuclear power plants (refer to Section 3.3 for details). Per this severe accident legislation, when conducting PSA, the occurrence frequency of accidents resulting in a cesium-137 release rate of 100 TBq or higher should be less than 10^{-6} per reactor-year.

In the context of PHWR, utility companies operating NPPs are required to submit their AMP to the regulatory body. AMP reports are developed using the MAAP-ISAAC tool. To ensure thorough and impartial evaluation of these AMP reports, regulators need an independent severe accident code characterized by high precision.

In response to the demand for an accurate and realistic severe accident code in PHWR safety assessments, the Korea Atomic Energy Research Institute (KAERI) has recently introduced a comprehensive CANDU severe accident code system known as CAISER (CANDU Advanced Integrated SEveRe code). The primary objective of the CAISER code system is to model the progression of severe accidents in CANDU reactors, covering relevant phenomena from core overheating to containment failure. It accomplishes this through mechanistic modelling, which accurately represents the various phenomena occurring during severe accidents.

5.3.1. Overall structure of the CAISER code

CAISER code system consists of a series of codes, which are separately developed:

- CAISER-C code simulating the core degradation phenomena in the CV;
- MARS-KS code simulating the thermalhydraulics of the PHTS;
- CONTAIN code simulating the severe accident phenomena in the containment;
- SIRIUS code simulating the fission products behaviour.

The CAISER-C code [126] is designed to replicate the core degradation phenomena within the CV. It comprises two primary modules: a fuel rod degradation module and a fuel channel degradation module. These modules simulate severe accident events occurring within a fuel channel and a CV, respectively. They work in close conjunction to replicate phenomena in both the fuel channel and CV concurrently.

Within the CAISER code system, the behaviour of coolant in a fuel channel is replicated using the pre-existing thermal hydraulic system code known as MARS-KS [127].

The module responsible for fuel rod degradation in the CAISER-C code models the solid components in the fuel channel, while MARS-KS is responsible for simulating the coolant in the channel. Consequently, the two codes exchange pertinent variables at each time step. More specifically, CAISER-C conveys the convective heat transfer rate to MARS-KS and receives back the convective heat transfer coefficient and the coolant temperature.

The fuel rod degradation module within the CAISER-C code simulates the solid components in the fuel channel, while MARS-KS handles the coolant within the fuel channel. Consequently, these two codes are designed to exchange relevant variables at each time step.

Specifically, the CAISER-C code conveys the convective heat transfer rate to MARS-KS, while receiving coolant temperature and the convective heat transfer coefficient data from MARS-KS.

The fission product inventory for a CANDU 6 plant has been calculated by using ORIGEN code [128] and implemented in the SIRIUS code [129], which simulates the fission product release from the fuel pellet, transport in the thermal hydraulic system, and deposit on the circuit solid walls. The fission products in the primary circuit leaks to the containment through the degasser condenser relief valve or through the corium discharge if the CV fails.

The severe accident phenomena in the containment is simulated by using CONTAIN code [130], which deals with a thermalhydraulics behaviour including hydrogen combustion, the fission product behaviour in the containment of an NPP, the MCCI, direct containment heating, and engineered safety features such as spray PARs. In the CAISER code, the CONTAIN code has been coupled with MARS-KS, CAISER-C, and SIRIUS codes, in order to provide the information of the liquid and vapour discharge from the safety valve, and the corium discharge from the CV, together with a fission product transport from a primary circuit.

The detailed modelling for CAISER code system has been described in reference [131] with the simulation of the CS28-1 experiment.

5.3.2. Reactor core and fuel channel modelling

Figures 43 and 44 illustrate the nodalization used in the CAISER-C code, which consists of a fuel rod degradation module and a fuel channel degradation module.

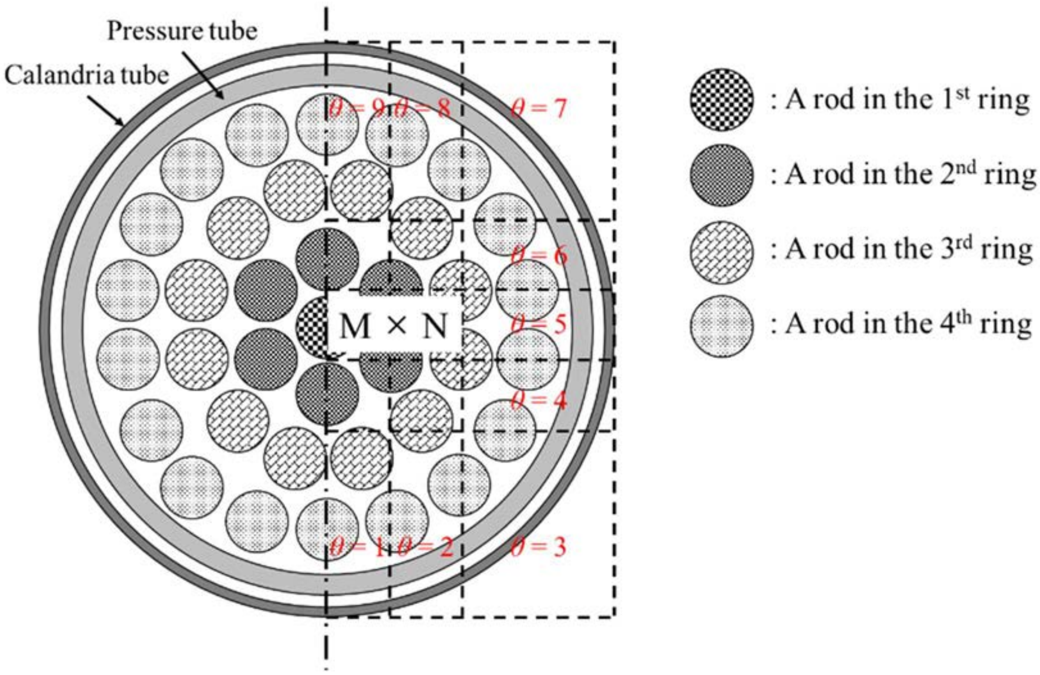


FIG. 43. CAISER nodalization for the fuel rod degradation module (reproduced from Ref. [126] with permission).

The fuel rod degradation module uses a modelling approach that considers only half of the fuel bundle, exploiting its inherent symmetry. To represent the cross-section of a fuel channel, cartesian nodes are employed, identified by their respective indices [M, N]. Additionally, the fuel channel incorporates an extra set of 1-dimensional nodes along the flow direction, denoted as [K]. Consequently, the modelling of fuel rods involves a 3-dimensional node system characterized by the node indices [M, N, K]. Similarly to the node system in a fuel channel, the fuel channel degradation module employs a 3-dimensional node system in Cartesian coordinates for 380 fuel channels in the CV, the numbers [I, J, K]. The number of nodes [M, N, K] and [I, J, K] are given by user inputs.

Therefore, the CAISER-C code is designed to have generalized 3-D Cartesian coordinates for the severe accident analysis in the fuel channel as well as in the CV, which is an appropriate node system to model the detailed severe accident phenomena in the fuel channel and in the caldaria tank.

As depicted in Figure 44, within the fuel channel degradation module, each individual node encompasses multiple fuel channels. The number of fuel channels within each node is determined geometrically based on its specific position, denoted as [I, J], within the CV. Given that each node accommodates multiple fuel channels, the governing equations are established for a representative fuel channel. This representative fuel channel is defined as one with an average power rating among all the fuel channels and is typically situated at the centre of the node. To account for the presence of multiple fuel channels within each node, considerations are made in the mass and energy equations of this representative fuel channel.

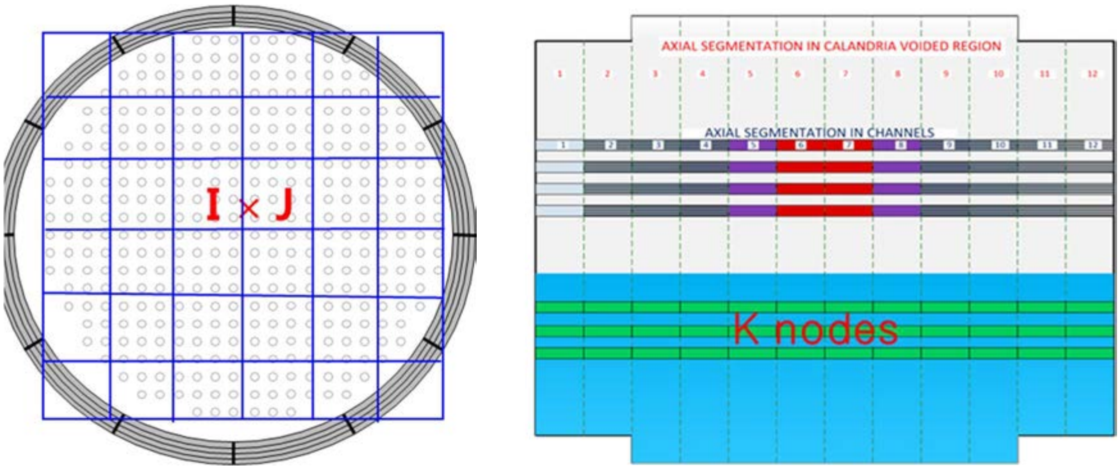


FIG. 44. *CAISER nodalization for the fuel channel degradation module (reproduced from Ref. [126] with permission).*

5.3.3. Core degradation model

The CAISER-C code employs two primary modules to model the core degradation phenomena. These modules are the fuel rod degradation module and the fuel channel degradation module. The fuel rod degradation module is responsible for simulating severe accident events within the fuel channel. This includes processes such as fuel rods uncovery, heating up, hydrogen generation due to steam-Zr oxidation, slumping, melting, relocation, and the subsequent thermal interaction of the molten mass with the PT or CT. Meanwhile, the fuel channel degradation module is tasked with simulating the broader severe accident phenomena within the CV. This encompasses fuel channel sagging, the formation of debris beds resulting from fuel channel failure, the formation of molten pools, and the CV failure due to wall ablation and creep rupture.

The progression of a severe accident within the fuel channel is significantly influenced by external factors such as the moderator water level and radiation heat transfer with the CV structure. Conversely, the overall severe accident progression within the CV is primarily determined by events occurring within the fuel channels. These events include the temperature of the CT and fuel channel failures. Therefore, in CAISER-C modelling, the fuel channel degradation module maintains close communication with the fuel rod degradation module. Specifically, the fuel rod degradation module conveys pertinent information regarding the severe accident progression within the fuel channel to the fuel channel degradation module, including the status of fuel channel failure and the CT temperature. Likewise, the fuel channel degradation module shares information with the fuel rod degradation module regarding the overall severe accident progression within the CV. This includes data on the moderator water level and the external heat transfer rate of a fuel channel. Both modules track changes in mass and energy for key components, such as fuel, cladding, PTs, CTs in the fuel rod degradation module, and debris beds, metallic, and oxidic corium pools in the fuel channel degradation module.

When a fuel channel fails, it becomes part of a debris bed. A portion of this debris bed stacks onto intact fuel channels, forming a suspended debris bed, while the remaining debris bed relocates to the bottom of the CV, creating a terminal debris bed. The suspended debris bed, along with the weight of the sagging fuel channel in the node above, increases the load on the supporting intact fuel channel. It is assumed that the suspended debris bed relocates to the node below when the intact fuel channel supporting it fails. The failure of intact fuel channels in each node is calculated using beam theory, considering the given load and temperature conditions. The gradual progression of suspended debris bed relocation continues as the intact fuel channels in the nodes below fail, ultimately reaching the bottom node of the tank, where it combines with the terminal debris bed.

Over time, the temperature of the debris bed increases due to decay power, potentially reaching the melting temperature of metallic and oxidic components. The thermal energy from the corium pool is transferred to the coolant in the cavity and leads to increase of the CV wall temperature, which can result in the CV failure. CAISER-C code simulates a debris bed layer, a metallic layer and an oxidic layer by computing the mass and energy equation for each component. CAISER-C also considers the crust layer which is formed inside the vessel wall and the inter-region between metallic and oxidic pools. To predict the CV failure time, the ablated wall thickness has to be calculated. Hence, in the CAISER-C simulation, the vessel wall has radial and azimuthal nodes. Figure 44 shows those radial and azimuthal nodes.

5.3.4. CAISER modelling for the heat transport system and the containment

In the CAISER code, the coolant in a fuel channel is simulated by using the existing thermal hydraulic system code MARS-KS [127], which has 1-dimensional nodes in the flow direction for the fuel channel. The PHTS nodalization used in MARS-KS consists of 420 nodes and 455 junctions. The core consists of 16 flow channels having 12 nodes in the flow direction and each flow channel is coupled with the CAISER-C code. The over-pressure protection safety valves are modelled, including a pressurizer safety valve, a liquid relief valve, a Degasser Condenser Tank (DCT), a degasser condenser relief valve. The degasser condenser relief valve is connected to the containment volume of the containment analysis code, CONTAIN. Additionally, when the CV fails, the corium discharges into the containment through a control functional valve which is designed to open under a tank failure condition.

The containment is modelled using CONTAIN code which has been developed for the simulation of severe accident phenomena in a NPP containment building including transport of aerosol or fission product, hydrogen combustion, MCCI and general containment thermalhydraulics [130]. It has been coupled with MARS-KS and CAISER-C codes which simulate the in-vessel phenomena. It obtains information from CAISER-C regarding corium properties and fission products following a CV failure. Simultaneously, it acquires information from MARS-KS concerning fluid properties and thermal-hydraulic boundary conditions for the specified coupled nodes. The pressure operated relief valve flow is connected to the S/G room cell, while a corium discharge is connected to the reactor vault cell. The containment is modelled with 41 cells, 72 links and 222 heat structures.

5.3.5. Fission product model

The SIRIUS (SImulation of Radioactive nuclide Interaction Under Severe accident) code is used for the simulation of the fission product behaviour in the CAISER code system. The SIRIUS code had been developed at KAERI, and is used for the analysis of the fission product transport in the core, the PHTS, the steam generator secondary side [129]. SIRIUS describes the behaviour of fission products and actinides in the form of gas and aerosols. The radionuclides are classified in the following 8 groups:

- Noble gases: Xe, Kr;
- Alkali metal iodides: I;
- Alkali metal hydroxides: Cs;
- Chalcogens: Te, Sb, Se;
- Alkaline earths: Ba, Sr;
- Platinoids: Ru, Mo;
- Rare earths: La, Zr;
- Structural materials: Zr, Fe, Cr, Ni, Mn.

SIRIUS simulates the generation and transfer of gas-phase and aerosol-phase fission products, and it also considers the condensation, evaporation, adsorption, impaction, gravitational precipitation, dispersion, and elimination by deposition on the walls due to the temperature, while the motions of fluid carrying the gas-phase and aerosol-phase fission products are calculated by MARS-KS for the PHTS. The fission product behaviour in the containment is modelled by the fission product analysis module in the CONTAIN code.

5.4. ADAPTED LIGHT WATER REACTOR CODES

The computer codes developed for LWRs are having certain limitations in applying for PHWR severe accident progression. An overview of those limitations is presented in the following subsection, which is followed by two examples of adaption of LWR codes to model severe accidents in PHWRs.

5.4.1. Limitations and adoption of alternatives codes for heavy water reactors

PHWR severe accident progression differs from the LWR progression during the phase beginning from the PT-CT contact to core disassembly phase. The debris composition, size and geometry are also quite different from the LWRs. The late phase scenario of PHWRs could however be similar to the LWR accident scenario. The following provides an overview of the limitations LWR codes have when modelling severe accidents in PHWR.

5.4.1.1. Fuel Channel Deformation Models

The simulation of severe accident for PHWRs requires PT deformation modelling. Under some scenarios, the SBO for instance, the PT can balloon and completely touch the CT as shown in Figure 45. This also impact the cross-sectional flow area within the fuel channel. LWR codes typically do not model this phenomenon, and specific models need to be added.

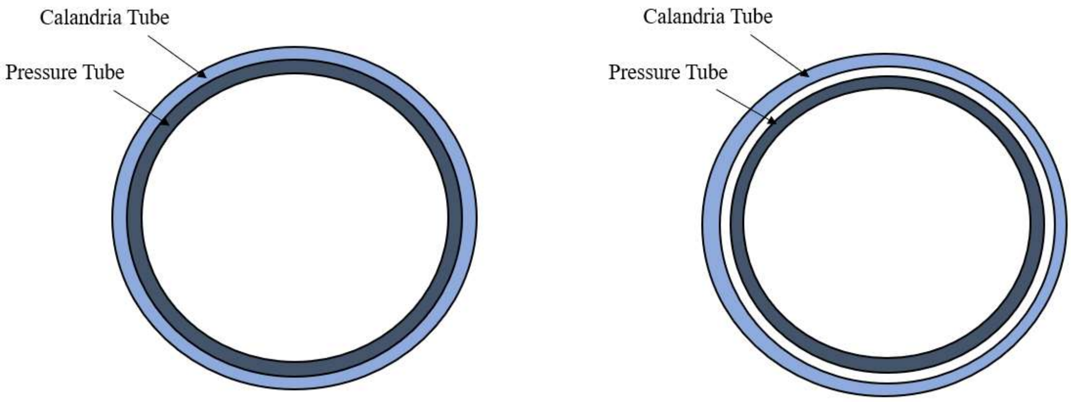


FIG. 45. PT-CT contact during the SBO initiated severe accident scenarios.

Similarly, the PT may sag and touch the CT during low pressure accident scenarios initiated with loss of coolant accidents. Modelling of the contact angle and length is also difficult in the LWR specific codes as such models do not exist in these codes, which could be modelled by assuming equivalent conductivity for the area of contact or through inclusion of fuel channel specific deformation models. Figure 46 presents the equivalent conductivity assumption.

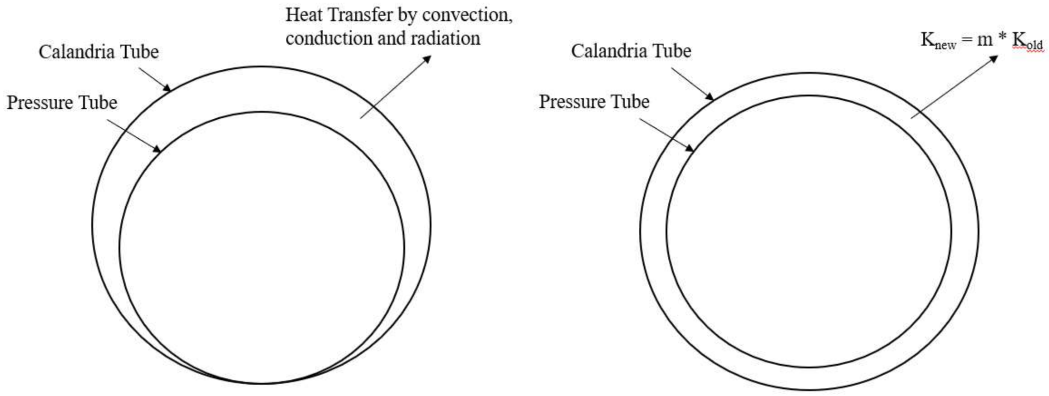


FIG. 46. PT-CT contact during the LOCA initiated severe accident scenarios.

In addition, different contact timings of PT with CT for different channels are necessary to capture the heat transfer phenomena correctly, which can be a limitation in LWR codes that typically assume single contact timing for all the channels. Also, LOCA in affected loop results in earlier PT-CT contact while in the unaffected loop, PT-CT contact may never happen.

5.4.1.2. Fuel channel failure and core disassembly

Due the main difference in core geometry between LWRs and PHWRs, channel failure and core disassembly are one of the key modelling challenges encountered while using LWR computer codes. Appropriate models for fuel channel failure and disassembly accounting for fuel and coolant conditions, moderator level and conditions and supported suspended debris therefore have to be developed and incorporated in those codes.

In addition, in light water reactors, enhanced hydrogen generation might occur during the reflooding phase, when the core is hot (e.g. due to embrittlement of the clad). In PHWRs, when channels disassemble and slump, it is likely that some water is left at the bottom of the CV, may enter into the hot channels and produce more hydrogen. Additional models are then also needed to analyse this phenomenon.

5.4.1.3. Calandria vessel geometry and terminal debris bed

The configuration and heat exchange of the terminal debris bed in the bottom of the CV might need incorporation of specific models to enable representing the geometry of the CV that differs considerably from LWR reactor pressure vessels. Similarly, modelling end shields might be difficult without code modifications when using LWR codes.

5.4.2. Development of heavy water reactors specific models in RELAP/SCDAP-SIM

RELAP/SCDAPSIM code was developed in in the United States for best-estimate modelling accidents in LWRs. It employs SCDAP/RELAP5/MOD3.2 models developed by the US NRC and by code members that are publicly available. In the past, the RELAP/SCDAPSIM MOD 3.4 version of the code was used for CANDU severe accident analysis [132]. Although many improvements were made to MOD 3.4 compared to MOD 3.2, this version of the code has no specific CANDU models.

For PHWR analyses, RELAP5, used for thermalhydraulic modelling, control systems, reactor kinetics and non-condensable gases transport, has been coupled with SCDAP that models the reactor core behaviour during severe accidents and the COUPLE subroutine is used to model the debris bed on the bottom of the vessel.

RELAP is based on a non-equilibrium and non-homogeneous model for the two-phase system. A partially implicit numerical scheme is used to solve the two-fluid equations and uses constitutive relations, including models for defining flow regimes and flow regime related models for interphase drag, wall friction, heat transfer and interphase heat and mass transfer and are formulated using empirical correlations. A generic modelling approach is used that allow the simulation of a variety of thermal hydraulic systems [133].

SCDAP is modular; the user can define components in the input deck, similarly as in RELAP5. In addition to the general input options, the user can introduce generic SCDAP core components in his model.

The RELAP/SCDAPSIM MOD 3.4 was used as independent code for CANDU sever accident analysis. The applications of the code include simulations of SBO and large LOCA scenarios, in-vessel retention studies and inside fuel channel early phase degradation [134-135].

5.4.2.1. Development of fuel channel and pressure tube deformation models

In SCDAP MOD 3.4, although several models are general enough to be applicable to the analysis of CANDU severe accidents, some of those models are however dedicated to PWRs or BWRs and cannot be used for CANDUs.

SCDAP code is used to model core components. This includes modelling fuel rod heat-up, ballooning and rupture, fission product release, rapid oxidation, zircaloy melting, UO_2 dissolution, ZrO_2 breach, flow and freezing of molten fuel and cladding, and debris formation and behaviour [136].

A mechanistic model for PT ballooning and sagging phenomena during the fuel channel heat-up phase and for the sagging of fuel channel assemblies during the core disassembly phase were developed during a PhD thesis at McMaster University, Canada [137]. These mechanistic models for CANDU were integrated into RELAP/SCDAPSIM/MOD3.6 code as new SCDAP subroutines and are merged with severe accident original models.

The description of the PT mechanistic deformation models developed and implemented in MOD3.6 are given below.

Model for pressure tube ballooning

Figure 47 illustrates the model for PT ballooning which uses the average temperature along the PT circumference to predict the average transverse strain at each axial location in a fuel channel. As the thicknesses of the PT and the CT are small compared to their diameters, the thin-wall assumption is employed for the hoop stresses calculation.

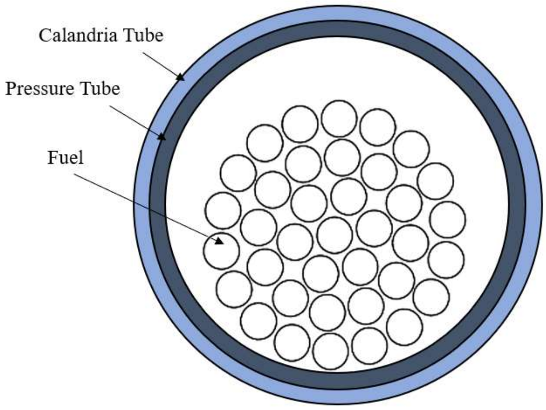


FIG. 47. Contact between pressure tube and calandria tube after ballooning.

For the PT and CT creep strain rate the Shewfelt's correlations [121] were implemented in the code. The creep strain rate of the PT is calculated only for temperatures above 450°C and is the sum of two terms representing the two mechanisms of possible PT creep deformation: the power law creep in α- phase and the grain boundary sliding. For the CT creep strain rate calculation, the sum of a dislocation creep term and a grain boundary sliding term is considered [138].

The contact conductance after contact between the PT and CT is calculated based on the mechanistic model developed by Yovanovich [123], in which the interfacial conductance is assumed to be the sum of the metal-to-metal contact term and the conductance across the gas gap term. For the contact pressure the approach employed in Cziraky's model [139] was used to estimate the PT/CT contact pressure equating the PT creep strain rate with that of the CT and the equation for contact pressure is solved iteratively.

Prior to contact between PT and CT, the PT/CT failure is determined by comparison of the calculated PT strain (averaged over the circumference) with a user input value for creep strain. To determine the channel failure after ballooning contact between PT and CT, an additional user input for the CT transverse failure strain has been implemented. The default value for this criterion is based on experimental evidence from a number of full-scale contact boiling tests [138].

Model for pressure tube sagging

When the PT experiences overheating, sagging of the PT can occur caused by longitudinal bending stress (see Figure 48). The model implemented in MOD 3.6 is based on simple beam theory, where the PT is represented by a beam with two fixed ends and the load on the PT is uniformly distributed. The model also considers the presence of spacers which are assumed to be rigid, and the PT is assumed to be perfectly horizontal and provides support to the PT via the four spacers [138].

The sagging of the PT is the sum of elastic and plastic bending, and the Shewfelt [140] et al longitudinal strain rate equation is employed to calculate the curvature of the beam. If the PT is predicted to contact the CT, a constant contact area and contact conductance will immediately be applied to the location of contact. Also, the user can input the contact angle for pressure-tube-to-calandria-tube sagging contact [138].

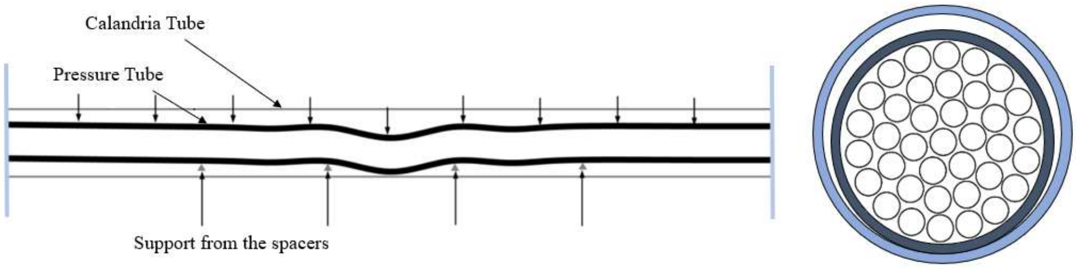


FIG. 48. Contact between pressure tube and calandria tube after pressure tube sagging.

5.4.2.2. Development of core disassembly and terminal debris models

Model for sagging of uncovered channels

The model developed to predict the phenomena during the core disassembly phase considers two cases: prior to contact and after channels contact.

Prior to channel contact, the channel is simplified to a single beam with fixed ends and neglecting the interactions between the PT and the fuel bundles, and between the PT and the CT. Both PT and CT are modelled as a single structure and follow the same strain rate correlation and creep strain law as for the PT.

After channel-to-channel contact, the interaction force between the two channels is considered. After the contact between a channel and its lower channel at a specific axial node, that node is assumed disengaged from the remaining of the channel, and the weight of the affected node is transferred to its contacting lower channel (see Figure 49) [138].

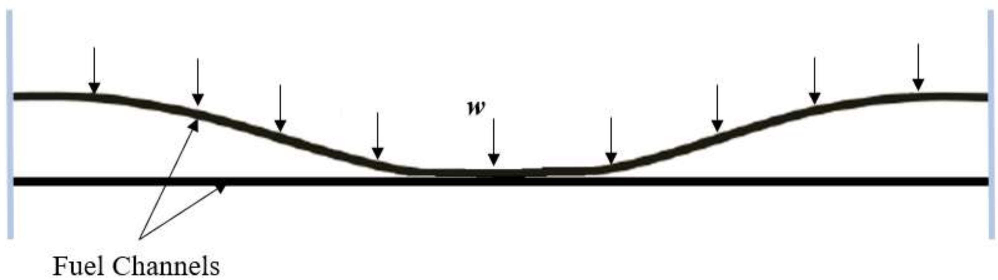


FIG. 49. Channel-to-channel contact.

Following, the position of this node is updated according to the lower channel position, while the rest of the channel is model as two separate beams each with one fixed end, considering thus two segments of the assembly from the fixed end to the position of contact. When the following next assembly node contacts its lower channel, it also disengages, the contact area increases node by node and the mass of the contacting node is transferred to its lower channel. Thus, after contact, all contacting nodes transfer loads to lower fuel channel and the remaining channel segments use cantilever (1-end fixed) deformation model to continue the sagging transient [138].

The following criteria were implemented in MOD 3.6 to determine disassembly of a fuel channel or a segment of the channel assembly [138]:

(1) the node disassembles when the CT temperature exceeds a user-input threshold;
(2) the node disassembles when the longitudinal strain of a node exceeds a user-input value;
(3) the node disassembles when the displacement of a node exceeds a user-input value (by default, two lattice pitches). This criterion is used to limit the extent of channel sagging as was observed from the Core Disassembly Test Facility at AECL (see Section 4.2.1) [141].

Currently, in MOD 3.6, after disassembly, the channel segments are completely supported by the lower channel of the same column. This will form a suspended debris bed that is maintained until core collapse.

Core collapse and end stub of the upper channels behaviour

The core collapse criterion and the relocation of pulled-out channels together with the supported suspended debris to the bottom of the CV is based on static beam loading calculation and the ultimate tensile stress theory.

For the upper channels, the end stubs that are not pulled out are left on the tube sheet after core collapse. The user has two modelling options to consider this situation [138]:

(1) Option 1: after core collapse, the fuel bundles in the end stubs slide and relocate at the bottom of the CV. Only PT and the CT segments are left behind;
(2) Option 2: regardless of the PT slope, the fuel bundle in the end stub do not slide. The end stubs continuously heat up and, if any of the disassembly criteria are met, can disassemble or relocate downward later. The heat conducted from the end stubs to the end shield is considered negligible and therefore is not modelled.

Debris bed modelling

In order to calculate the heat-up in the debris and/or its surrounding structures, a two-dimensional finite element model based upon the COUPLE subroutine has been used.

The model for the debris and the surrounding structures considers the initial internal energy and decay heat within the slumped debris. The model then calculates the heat conduction through the wall structures and the water surrounding the debris in both the radial and azimuthal directions. A key usage of the model consists in calculating the vessel wall heat-up in order to estimate the time at which the vessel would rupture. Noteworthy capabilities include the modelling of the thermal conductivity of porous material, a debris bed whose height grows sporadically with time and natural circulation of melted debris. The limitations of this model are its inability to simulate the flow of molten material in an adjacent porous region, the oxidation process in the debris bed, and the release of fission products within the debris bed.

The COUPLE subroutine has been successfully used for PWRs and BWRs, with vertical cores and with hemispherical lower heads. Although the hemispherical geometry has been used to model CANDU severe accidents [134], improvements in the COUPLE subroutine were implemented for more accurate modelling of the CV horizontal cylinder [142].

5.4.2.3. Simulation flowchart with Development RELAP/SCDAPSIM/MOD3.6

Figure 50 presents the flowchart for severe accident simulation with RELAP/SCDAPSIM/MOD3.6 code. As in previous version, the severe accident progression until core collapse and water depletion in the CV was simulated as a single code run. However, a two-step approach can be used to increase the channel resolution in the model, especially in terms of elevation as the channels at different elevations will heat up at different times and with different rates depending on their uncovery times and channel power. Also, in this approach, the interaction between channels at different rows in terms of heat and mechanical load transfer would be better accounted for.

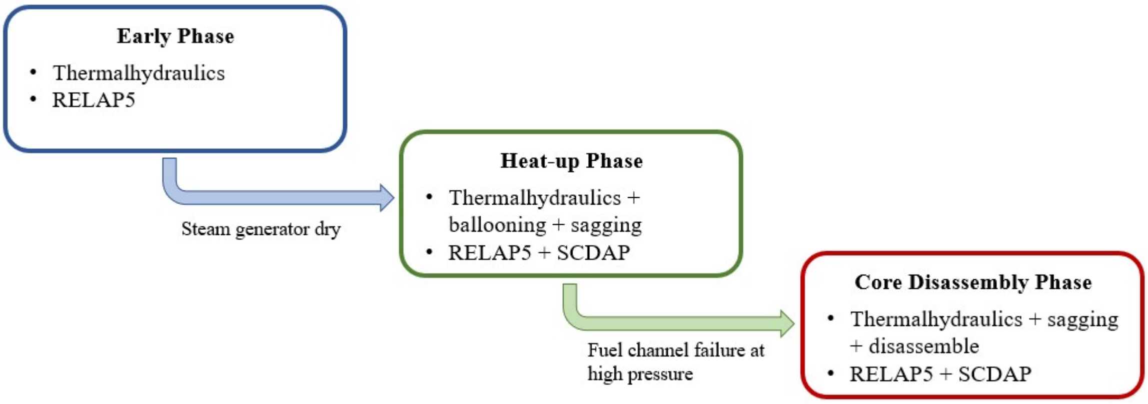


FIG. 50. Flowchart for severe accident simulation with RELAP/SCDAPSIM/MOD3.6 code.

In this two-step approach, the early phase of the severe accident (i.e., from the initiating event until failure of the channel and HTS depressurization) is modelled using a full plant RELAP5 model that includes the PHTS, the feed and bleed system, the secondary side, the moderator system, and the shield-water cooling system. The fuel channels can be grouped in a number of characteristic channels grouped by both elevation and channel power.

After the rupture of the first channel, the thermalhydraulic response above the headers, the feed and bleed system and the secondary side have limited impact on the accident progression. Therefore, for the core disassembly phase, a new nodalization is adopted where the initial conditions are inherited from the full-plant simulations after first channel rupture and prior to significant core degradation. In this representation, a large number of fuel channels can take into account the channel location/elevation to allow more accurate treatment of the moderator boil off phenomena as well as to capture channel-to-channel interactions.

In summary, in the two-step approach, the sever accident is first run using the full-plant model and is terminated soon after the first channel rupture (i.e., prior to channel heat-up and after PHTS depressurization). Then, the initial and boundary conditions are transferred to the core disassembly model and the transient is continued until the terminal debris bed is formed [143].

5.4.2.4. Validation

All models implemented in RELAP/SCDAPSIM/MOD3.6 code were validated against PT contact boiling tests [144-145], for PT ballooning model and against sagging test [146] for PT sagging model. The model predictions agreed well with the experimental data. Regarding the contact boiling tests, the contact temperatures are well predicted compared to the measured temperatures. However, tests where film boiling occurred in patches overpredicted the time to quench. This result is due to the limitation of the one-dimensional codes in predicting two-dimensional phenomenon.

As a general conclusion, the models implemented are conservative in predicting the occurrence in film boiling and the duration of film boiling [138]. When two-dimensional phenomena are not present, for instance when the entire CT surface in sustained film boiling or when no film boiling occurred, the predicted PT and CT temperatures during the post contact phase are reasonably close to the measured temperatures. Also, when small circumferential PT temperature gradients are present, the results for a PT failure strain of 20% (averaged over the circumference) are in agreement with the experimental data for different heatup rates and internal pressures up to 5 MPa [138].

Validation of the sagging models against several low-pressure PT deformation tests showed good agreement between the results obtained and the experimental data. However, for the tests in which the spacers (garter springs) were used, the sagging models overestimated the temperatures at a given deflection (or at PT-CT contact). The model of the spacers, assumed to be rigid, was found to be responsible for this behaviour, whereas in the experiments they were found to deform under high temperatures [138].

5.4.3. Development of heavy water reactors specific modules for code ASTEC

ASTEC is an acronym for Accident Source Term Estimation Code developed jointly by Institute for Radiation Protection and Nuclear Safety (IRSN), France and GRS, Germany [147]. Under co-operation agreement between IRSN France and BARC India, BARC carried out development of PHWR specific modules for ASTEC to facilitate its adaptation for simulation of severe accidents in PHWRs. The development of PHWR specific modules is discussed below.

5.4.3.1. Development of limited core damage accidents specific modules

Modification of the mesh

The mesh used by the ASTEC code and mesh used by the PHWR specific module of ASTEC are different for a reactor channel. Figure 51 shows the mesh generated by the PHWR module. The thermalhydraulic and thermo-chemical parameters calculated by ASTEC are used by the PHWR module to predict the PT deformation, the temperature distribution in the radial and azimuthal direction in the PT and CT and the transfer of heat from the CT surface under boiling regime.

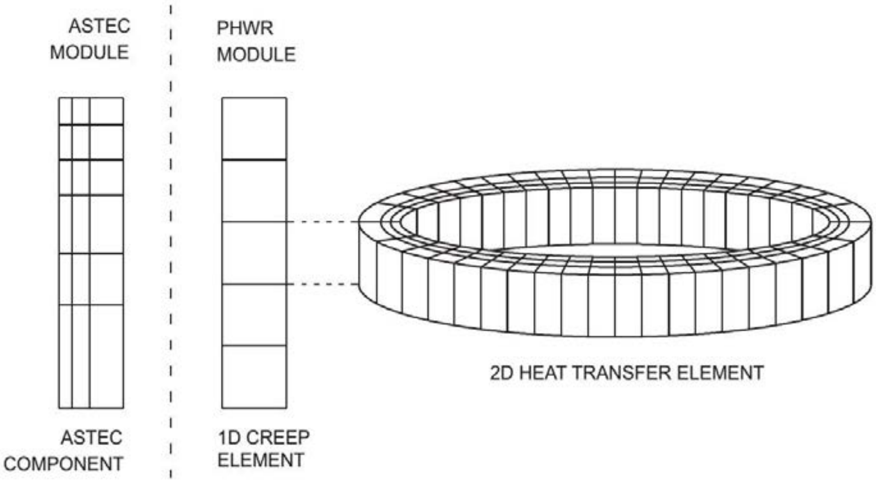


FIG. 51. Elements generated by the PHWR module in ASTEC (reproduced from Ref. [148] with permission courtesy of Elsevier).

To calculate the creep and the contact heat transfer, the PHWR specific module uses a one-dimensional mesh for calculation of axial creep for PT and a two-dimensional mesh for heat transfer in the PT and the CT. For conservation of energy the mesh in the radial direction is consistent for ASTEC and the PHWR module.

Calculation of pressure tube deformations

The internal pressure and PT temperature predicted by ASTEC is used for the prediction of ballooning and sagging deformation of the PT using the PHWR module. The CT is assumed to be rigid and does not undergo deformation. The module predicts contact between the PT and the CT and then creates two-dimensional mesh for calculation of the heat transfer through the contact. The heat fluxes are calculated by ASTEC and the PHWR module together and are used in the two-dimensional mesh to evaluate temperatures in radial and azimuthal directions.

Management of contact locations and associated heat transfer

After establishment of contact between the PT and the CT a two-dimensional mesh is created near the contact location. This facilitates the creation of multiple heat transfer nodes which can operate independently based on the prevalent heat transfer modes.

Figure 52 shows the heat transfer paths available after contact between the PT and the CT. The PHWR module does not allow the simulation of natural convection between the tubes.

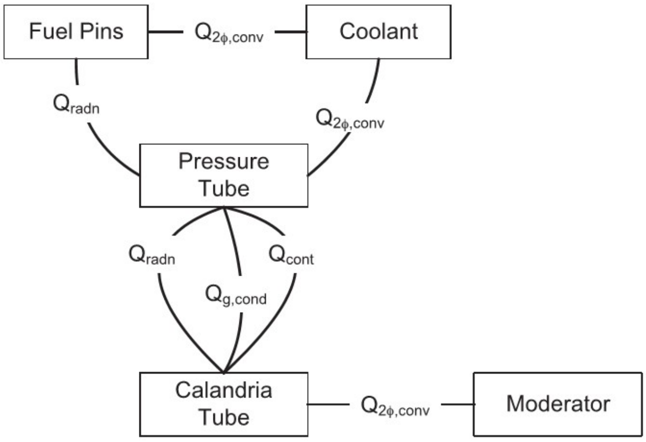


FIG. 52. Heat flow from fuel pins to moderator (reproduced from Ref. [148] with permission courtesy of Elsevier).

5.4.3.2.Transition from LCDA phase to SCDA phase

The modelling of transition from LCDA phase of accident to SCDA phase of an accident is currently performed based on the fuel channel disassembly and reactor core collapse criteria discussed in the IAEA-TECDOC-1594. The inputs required for the modelling of SCDAs are derived from the calculated output for the LCDA phase and the core collapse criteria.

5.4.3.3. Development of severe core damage accidents specific modules

Modification of the geometry of the lower plenum to model the horizontal calandria vessel

The simulation of the lower plenum of PWRs is performed using the LOWERPLE component in ASTEC [149]. The LOWERPLE has a semi-ellipsoidal shaped vessel which can contain the debris. In case of PHWRs, the cylindrically shaped CV acts like a lower plenum. The LOWERPLE component is modified to suit the cylindrically shaped CV geometry in two dimensions (radial and azimuthal).

The symmetric nature of the CV allows modelling half of the CV as a macro component in ASTEC. The surface area and heat capacity of the component is calculated considering the mirror component present across the axis. Figure 53 shows that each mesh of the component has four interacting surfaces that allow interactions with neighbouring meshes and interactions with the CV internal (debris or molten magma or moderator) fluid as well as with the external fluid (vault water), if present.

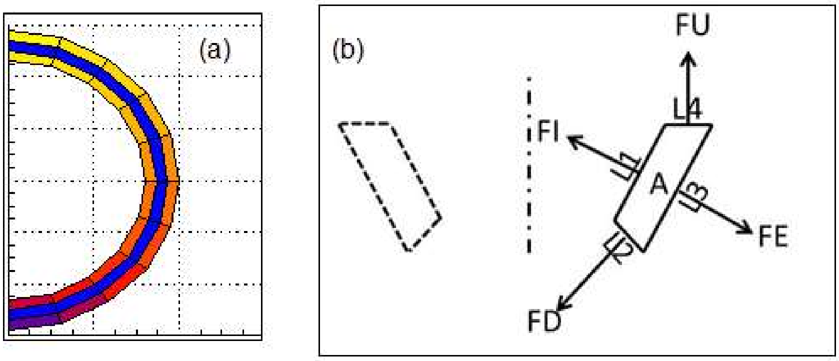


FIG. 53. (a) Calandria vessel with discretization in the radial and azimuthal directions; (b) A discrete volume of the calandria vessel and its mirror volume with interaction faces (courtesy of O. S. Gokhale, BARC).

Figure 54 (a) shows a representation of the magma or debris layer within the CV. The faces FI and FE represent the surfaces available for heat exchanges with internal structures and the CV surface respectively. The upper face FU and downward face FD allow interactions with further layers of magma or debris if they are present. The font and backward facing area (marked as A in Figure 54 (b)) allow interaction with the END SHIELD component of the CV.

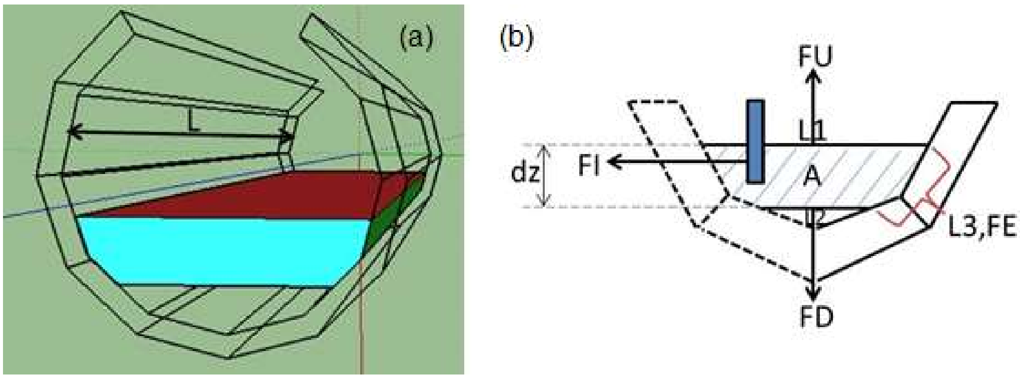


FIG. 54. (a) Representation of the debris/magma layer in the calandria vessel; (b) Interaction faces for the layer (courtesy of O. S. Gokhale, BARC).

Modelling of end shields and associated internal components

The LOWERPLE component of ASTEC was modified to allow closing of the CV on both ends by simulating end shields. The end shields are modelled as layered structures with a maximum of three number of layers which can represent the Calandria Side Tube Sheet (CSTS), the layer of carbon-steelballs and the fuelling machine side tube sheet. Figure 55 shows a schematic of the modelled end-shields. The number of holes in the end shield and its diameter can be selected by the user to model coolant channels passing through the end shields. The layer of the end shield can interact with other layers, or components present on the CV side or a boundary condition can be simulated on the fuelling machine vault side.

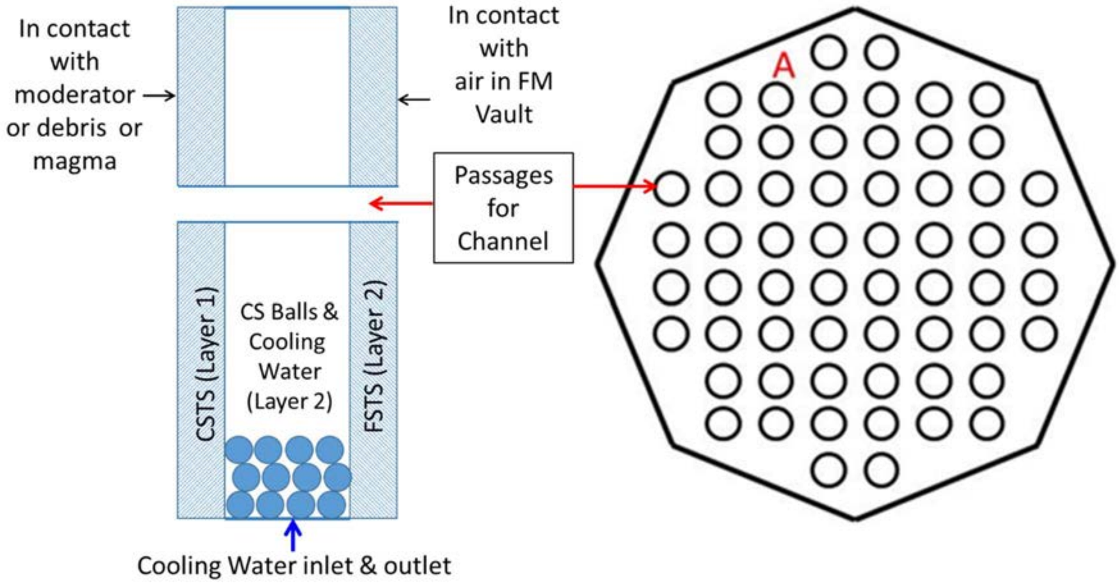


FIG. 55. (a) Schematic of the end shield with modelled layers; (b) Cross section of the end shield with representative holes for channels passage (courtesy of O. S. Gokhale, BARC).

Management of conductive, convective and radiative heat exchanges within the end shields and within the calandria vessel

Due to presence of several more components in the form of end shields and changes in the geometry of the LOWERPLE component in ASTEC, the heat exchange paths were suitably adopted for PHWRs.

Heat conduction:
The layered structure of the end shields allows modelling conduction heat transfer to adjacent layers. The modelling of heat conduction through the layer of carbon steel balls is approximated based on the effective conductivity approach. The effective conductivity is calculated based on the same spherical debris bed equations used for debris bed of typical LWRs.

Convective heat exchanges:
In PHWR geometry, it is assumed that the inner surface of CSTS of the end shield, part of the CV inner surface and the top surface of the magma/debris can contact the water or steam present within CV. The heat transfer from these surfaces through convective mode is calculated based on the area available for the heat exchange. The area calculation takes into consideration the effect of height magma layers / water level and part of the CV surface area which is submerged under the magma layer or the water.

Convective/conductive heat exchanges within debris/magma layers:
The inner surface of the CSTS and the CV can also interact with the magma or the debris layer through convective or conductive modes, respectively. The PHWR specific modification takes into consideration these heat exchanges and the calculation of heat exchange areas is performed as described earlier. The heat transfer coefficients are calculated based on the same set of correlations used for the LWR calculations available in ASTEC.

Development of radiation heat transfer module specific for PHWR SCDA phase

During SCDA phase of severe accidents in PHWRs, the radiation heat transfer can be significant due to the temperature difference between the molten/solid debris and the CV, externally cooled by vault water. The surfaces that can participate in the radiation heat transfer include topmost layer of magma/debris, the CV inner surface present above the magma/debris layer and the inner surface of CSTS. To model the radiation heat transfer between these surfaces, the RADPHWR model was developed based on the radiosity approach. The model calculates the view factors transient manner between participating surfaces based on their geometry and the availability of surface areas based on formation of new debris layer or changes in existing debris layers. The calculation of view factors is based on:

- The finite surface approach: the CV is modelled to have a finite length, closed at both ends by modelling end shields, or;
- The infinite surface approach: the end shields are not modelled, and the CV is assumed to be a long cylinder.

The calculation of the view factors is appropriately performed by the code.

The RADPHWR has been verified by comparing results of pure radiation test case with results obtained from COMSOL multi-physics code.

Development of the ex-vessel heat transfer module specific to PHWRs

In order to facilitate heat transfer from a horizontally placed cylindrical CV to the calandria vault water, in which it is submerged, the heat exchange coefficient of the CESAR module of ASTEC was modified. The correlations used in this model were specifically selected to model the CV ex-vessel heat transfer. The correlations use geometrical configurations specified by the local inclination angle of the surfaces to account for the curvature of CV surface.

5.4.3.4. Validation

Validation against experimental data for LCDA

The sagging deformation model for the PT and the CT has been validated against experimental data available in the literature [148]. The experiment simulates a short channel containing a CT and a bent PT which is in contact with CT at certain locations. The coolant flow in the channel is simulated using high temperature oil flow. By comparing the temperatures in the experimental setup and the temperatures predicted by the model, the interfacial conductance is calculated using inverse techniques. The model calculations are found to be consistent with literature data.

The PT deformation prediction was validated against PT ballooning experimental data [150]. These experimental data set consist of PT deformation measurements as well as its measurement of temperature at various locations. The experimental investigation indicates a decrease in PT temperature after the establishment of a contact with the CT through ballooning. The predictions of the model are consistent with the experimental findings.

Figure 56 shows the calculated profile for the deformed PT and the temperature distribution in both the PT and the CT at different stages of PT–CT contacts. The predictions were found to be in good agreement with experimental results [148].

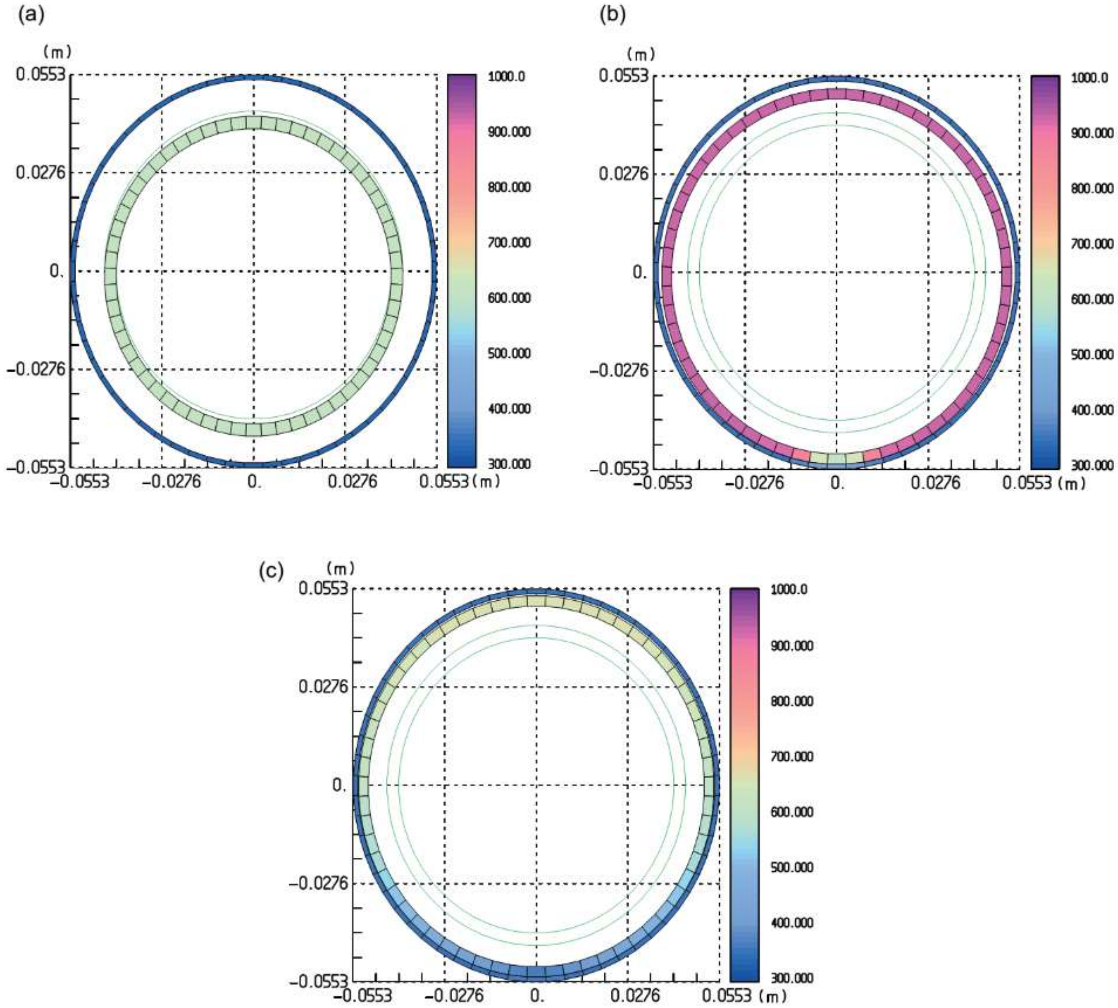


FIG. 56. PT and CT profile and temperature distribution: (a) in steady state; (b) at partial contact between PT and CT; (c) at complete contact between PT and CT (reproduced from Ref. [148] with permission courtesy of Elsevier).

Capability demonstration with sample calculation of the LCDA phase for 220 MWe Indian PHWR

In order to demonstrate that the clad and PT temperatures remain below the melting point under LOCA conditions, a full-length single channel was modelled, including the PT, the CT, twelve fuel bundles, four garter springs and the surrounding moderator. Figure 57 shows the ASTEC nodalization used for this simulation.

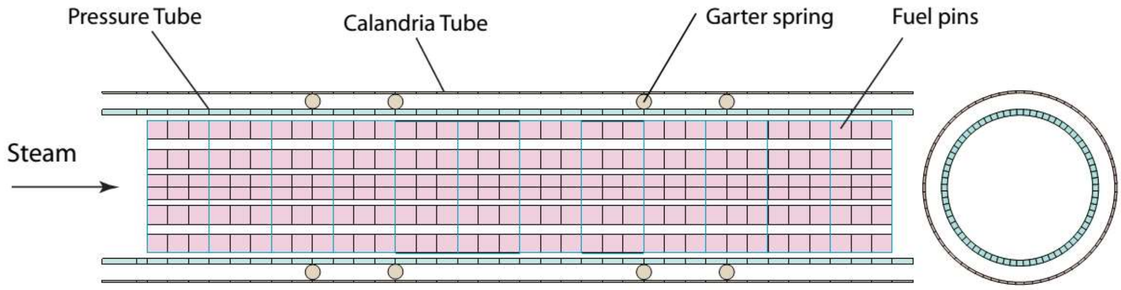
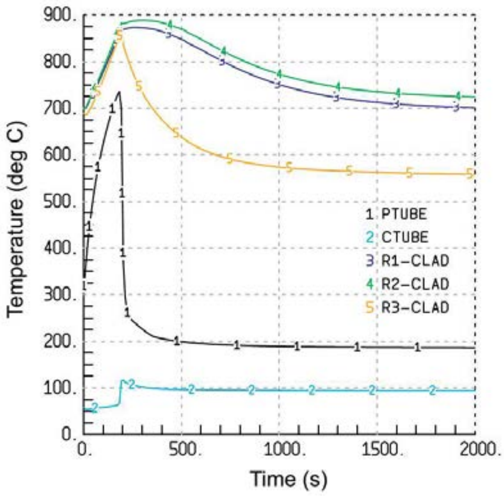
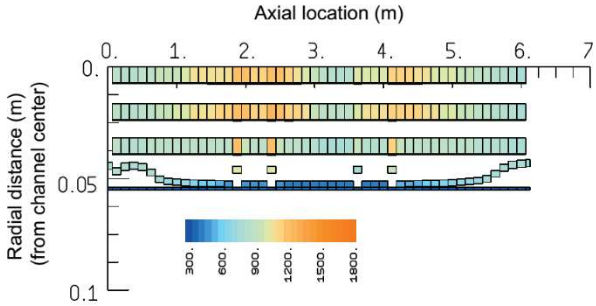


FIG. 57. Nodalization of reactor channel (reproduced from Ref. [148] with permission courtesy of Elsevier).

Two LOCA simulations were performed at internal pressure of 20 bar (case 1) and at low internal pressure of 1 bar (case 2). The dominant mode of PT deformation was predicted to be ballooning creep for the case 1, with formation of full circumferential contact between the PT and the CT. Figure 58 shows the results for the case 1.



(a) Temperature evolution at the centre of channel;



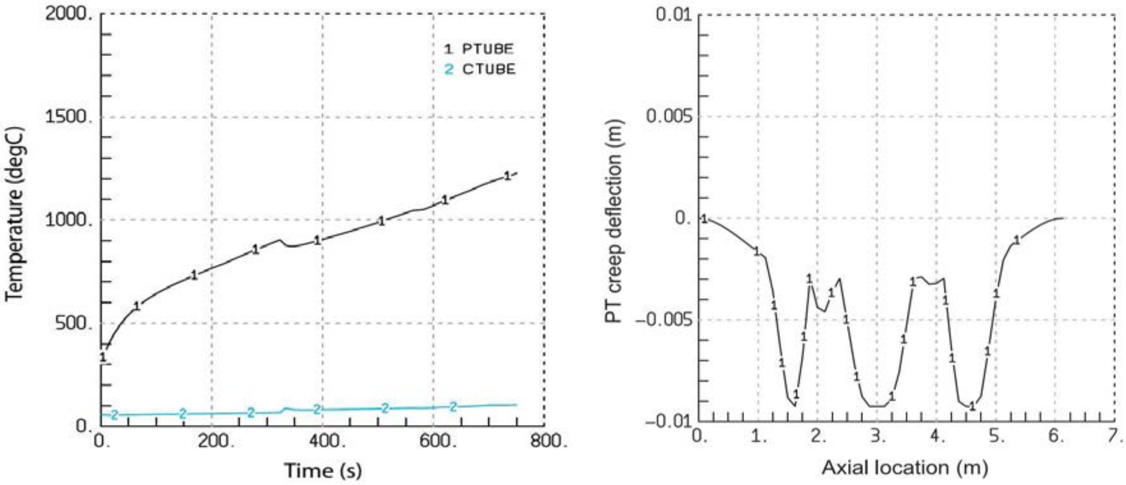
(b) Channel temperature distribution and deformation

FIG. 58. Temperature and deformation for case 1 (reproduced from Ref. [148] with permission courtesy of Elsevier).

Figure 59 shows the results for the case 2. In that case, multiple contacts between the PT and the CT were predicted. Nevertheless, it was observed that the heat transfer through the restricted contact area was insufficient to stop the temperature increase of the PT.

Capability demonstration with sample calculation of the SCDA phase for 540 MWe Indian PHWR

To evaluate the structural integrity of the CV for in-vessel corium retention, an analysis was conducted. The thermal-hydraulic phenomena were simulated using ASTEC-PHWR, while the structural integrity assessment was performed with a finite element methods code (see Section 6.2) [151]. An initial condition of collapsed core post LOCA analysis was assumed.



(a) Temperature evolution at the channel centre *(b) PT deflection at the end of transient.*

FIG. 59. Temperature and deflection for case 2 (reproduced from Ref. [148] with permission courtesy of Elsevier).

The CV was modelled in the ASTEC PHWR specific module using the symmetry option about vertical mid plane. The CV was discretized into 9 nodes in the azimuthal direction and 3 nodes in the radial direction. The debris bed was initialized based on the inventory of collapsed channels taking into consideration the oxidation content in Zr material. The vault was modelled as a single water volume of CESAR. The heat structure component of ASTEC was used to model vault walls. The CSTS, CS balls layer and the fuelling machine side tube sheet were modelled as three layers of end shields. Several other heat structures that are present in the end shields were also modelled, such as octagonal flanges, inner and outer shells, and lattice tubes. Figures 60 and 61 show the heat transfer modes and heat flow paths accounted for.

Figure 62 shows the temperature evolution obtained from the analysis for various structural components. This is then used as an input for performing a structural integrity assessment of the CV.

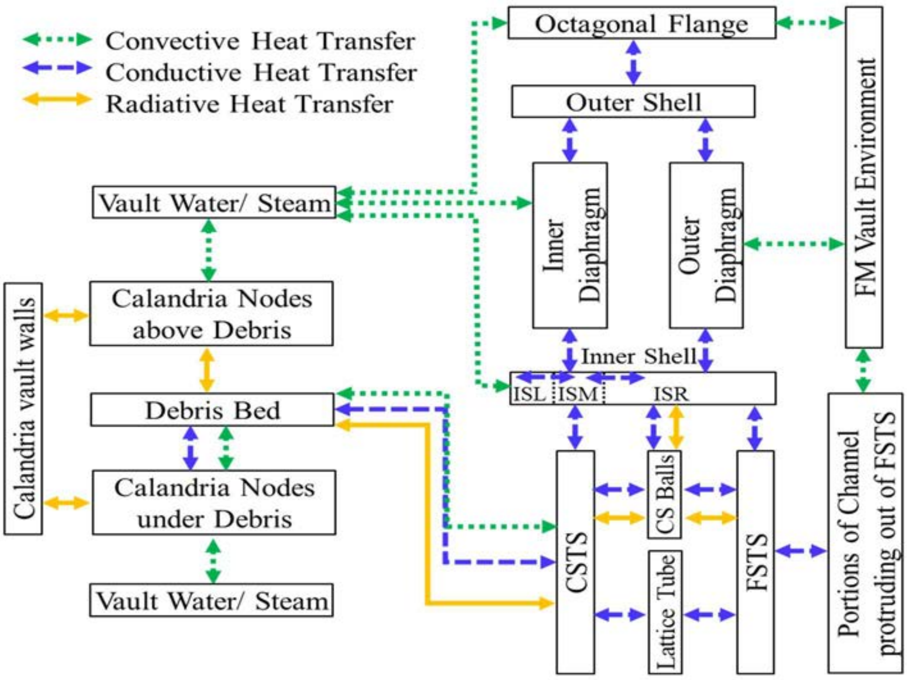


FIG. 60. Heat exchange path between various components of calandria vessel (reproduced from Ref. [151] with permission courtesy of Elsevier).

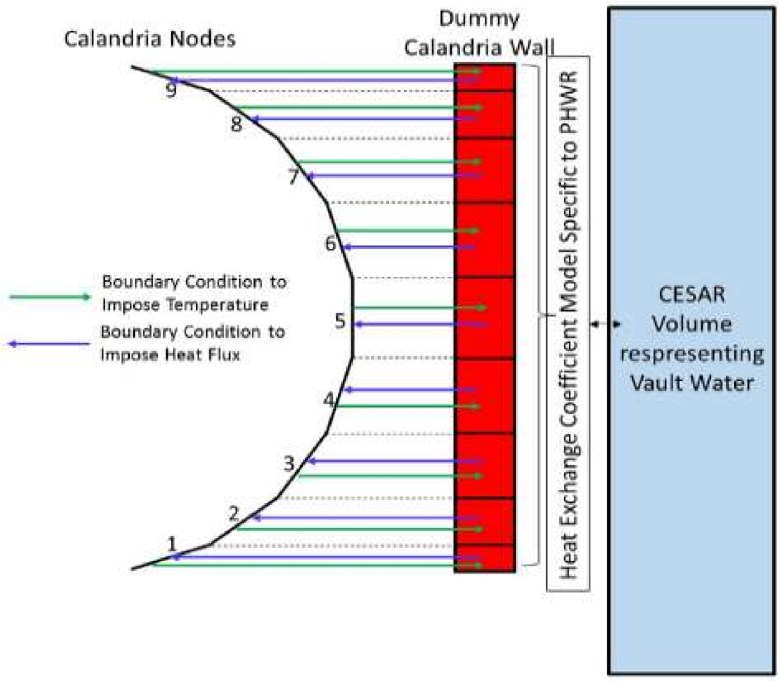


FIG. 61. Boundary conditions between various components of calandria vessel (reproduced from Ref. [151] with permission courtesy of Elsevier).

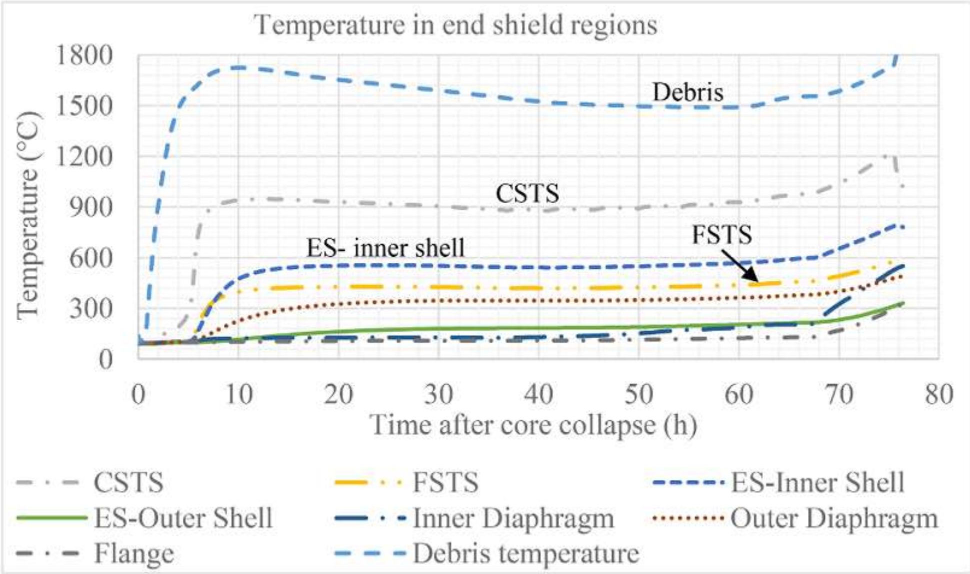


FIG. 62. Temperature evolution of end shield structures and debris (reproduced from Ref. [151] with permission courtesy of Elsevier).

5.5. CONCLUSIONS

The recent years have been characterized by an interest in expanding knowledge and simulation capabilities for severe accidents for all nuclear industry stakeholders around the world, including in PHWR MS. Along with more comprehensive regulations (Section 3) and expanded severe accident experimental programmes (Section 4), extensive code development, application, and validation has occurred.

The PHWR-dedicated analysis codes, such as MAAP5-CANDU and CAISER, have incorporate new R&D results to support mechanistic models and/or refine existing empirical correlations and criteria related to severe accident progression. Improved nodalization details have been implemented to allow for more accurate modelling of the core collapse progression (channel failures, suspended, and terminal debris beds). This approach is also being adopted in the new technology-neutral Indian code PRABHAVINI. In parallel, LWR analysis codes such as RELAP5/SCDAPSIM and ASTEC have been adapted for use in PHWR severe accident simulations by the implementation of PHWR core characteristics and relevant empirical correlations to allow the simulation of core collapse progression, generally with 2D detail. Furthermore, integration of parallel code versions, or previous loosely coupled or multi-step code systems, towards technology-independent codes or tightly coupled code systems have also been part of the recent development in various codes (MAAP6-CANDU, PRABHAVINI, CAISER, RELAP/SCDAPSIM).

Specific improvements have been made in PHWR severe accident codes with respect to phenomena identified as requiring further work in IAEA-TECDOC-1727, more specifically in the modelling of the reactor core and fuel channels, core collapse and disassembly, suspended and terminal debris beds and fission products. Significant investment has been made in developing experimental facilities and conducting experiments, as presented in Section 4. The progress in incorporating the latest experimental knowledge into computer remains ongoing and requiring continuous effort.

6. DEVELOPMENTS IN SUPPORTING MODELS AND TOOLS

Integral lumped-parameter codes were developed to model complex phenomena encountered during severe accidents in PHWRs, such as PT deformation, core degradation and collapse, corium behaviour and retention in the CV, and various ex-vessel processes, utilizing mostly embedded empirical and occasionally mechanistic models and approximations.

Section 5 has highlighted the advances in those integral codes that are the result of recent knowledge obtained from the experiments, presented in Section 4, and advances in computational capabilities. Supporting computational models and tool such as Computational Fluid Dynamics (CFD), Finite Element Analysis (FEA) methods or other three-dimensional tools can also help in gaining more insight on specific phenomena related to PHWR core and containment safety during severe accident conditions and help in increasing the accuracy of integral codes where three-dimensional phenomena are important.

This section presents recent developments in PHWR MS in those supporting models and tools to gain additional insights on phenomena such as disassembled channels coolability, corium behaviour, in-vessel retention, and hydrogen distribution and mitigation.

6.1. DEBRIS BED AND CORIUM BEHAVIOUR

The following subsections describe some supporting models that have been developed to provide additional insights on the behaviour of disassembled channels, the corium, and the corium interaction with the CV.

6.1.1. Estimation of Zr oxidation by steam in heavy water reactors disassembled channels

The disassembly of channels from their original location and relocation to the bottom of the CV allows ingress of the steam present in the CV into the disassembled channels or segments. The high temperature within the channel and the presence of un-oxidised Zr surfaces may lead to further oxidation and generation of hydrogen. The main parameters that affect this oxidation reaction are the temperature of the Zr surface and the availability of steam, of which the latter depends on the ingress of steam into the channel openings, which in turn may strongly depend on the orientation of the channel or channel segment.

To estimate the extent of Zr oxidation, simulations were carried out at BARC using the ANSYS/FLUENT commercial CFD code, as reported in References [90-91]. The Zr-steam oxidation reaction was modelled using user defined functions in ANSYS.

Figure 63 shows the temperature contours obtained for a longitudinal cross-section of a 1-m long channel segment and the temperature at the centre of the segment for different inclinations. It is observed that for a horizontal segment, the ingress of steam is limited to regions near the channel openings at the ends. Associated Zr-steam reactions and hydrogen generation are minimal in this case. With an increase in the inclination of the channel segment, more steam flows into the channel, driven by buoyancy created from decay heat inside the fuel bundles.

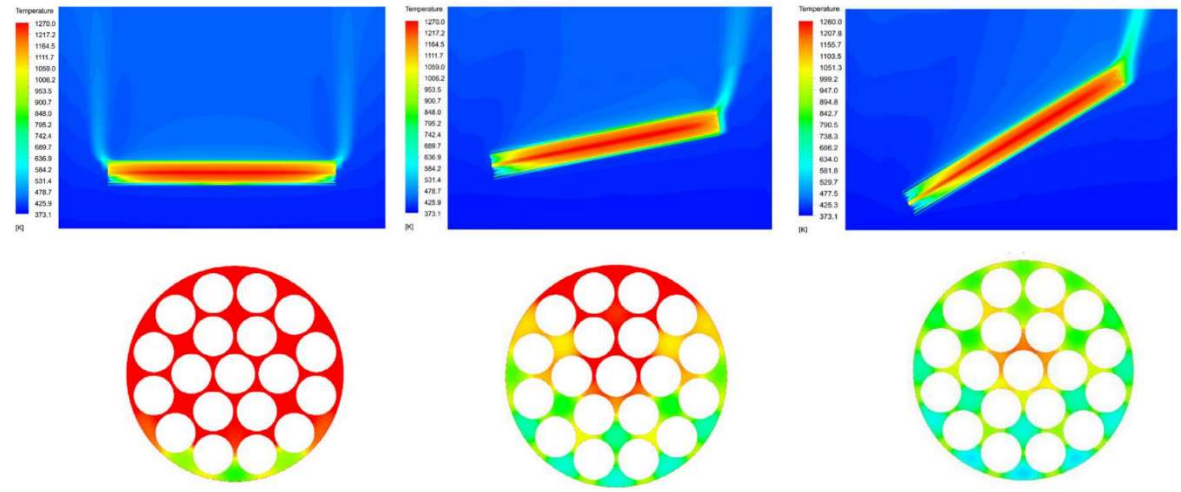


FIG. 63. Temperature fields for a disassembled 1-m long channel segment with different inclination under steam ingress (reproduced from Ref. [91] with permission courtesy of Elsevier).

The flow of steam through the channel produces two effects:

(1) It increases the ingress and concentration of steam within the channel, increasing the Zr-steam oxidation reaction;

(2) Being relatively cooler, the steam flowing through the channel cools the bundle surfaces, decreasing the Zr-steam oxidation reaction [90].

The effect of increased steam ingress into the channel was found to be dominant over the cooling effect produced by the steam flow for channel inclination of up to 20 degrees with respect to horizontal. Beyond 20 degrees of inclination, the cooling effect due to steam flow became more dominant [91]. The predictions of these CFD calculations agreed well with the experiments discussed in section 4.3.1.3. The calculated values of Zr-steam oxidation can be used by integral codes to improve the estimate of Zr-steam oxidation as a function of channel orientation, provided the distribution of probable channel segment lengths and inclinations can be established.

6.1.2. Corium behaviour

Supporting models have been developed in PHWR MS to gain additional insights on corium behaviour. This section highlights a few examples.

6.1.2.1.Corium crust behaviour

Some recent modelling research performed by CNL has focussed on the corium crust that is expected to surround the pool of molten corium during the late in-vessel stage of a CANDU severe accident. The crust would protect the calandria wall from attack by molten corium, prevent molten corium from escaping the calandria, and contribute to the heat flux exiting the calandria to the surrounding water.

The corium crust would form during melting, due to decay heat of the terminal debris bed at the bottom of the calandria [152]. Debris would begin to melt in the interior of the bed, trickle down through the porous bed, and freeze on or near the cooled calandria walls. Analysis has indicated that heat transfer from a small stream of molten corium contacting the thin calandria wall might lead to a local dry-out on the outer surface of the wall. However, preliminary calculations using a one-dimensional LWR model for melt flow in porous debris [153] suggest that corium would freeze into a crust well before reaching the calandria wall.

With a fully formed molten pool and corium crust, the in-vessel retention of corium would require that the heat flux exiting the calandria outer surface be lower than the critical heat flux for boiling water. In the last decade, progress in both experiments and modelling has improved the understanding of the exiting heat flux.

6.1.2.2. Distribution of heat flux exiting the calandria vessel

At each CV location the local heat flux exiting the CV is a result of both heat entering the corium crust from the molten pool it surrounds, and the heat produced within the crust. Thus, estimation of the exiting heat flux goes hand in hand with estimation of the local crust thickness.

A set of one-dimensional analytical formulae and correlations for use in integral codes has been derived at CNL for steady state crust thicknesses above and below the corium pool, temperature profiles in the corium crust and the CV wall, and heat fluxes exiting the CV outer surface. The solutions (yet to be published) are in terms of dimensionless parameters and account for the mushy zone of crust where the corium temperature is between solidus and liquidus.

Exiting heat fluxes were also calculated by a two-dimensional computational fluid dynamics model of the corium pool and crust, as documented in an analysis from the Politehnica University of Bucharest, Romania [154]. The solution included both the melting transient and the ensuing quasi-steady state. The exiting heat fluxes were found to be lower than the anticipated critical heat flux.

A method of calculating the exiting heat fluxes in three dimensions was developed at CNL as documented in [155]. Steady state heat transfer in the corium crust and CV wall, but not the corium pool, was solved using a finite element model. The profile of heat flux entering the crust from the pool was required as an input to the model and was provided by CFD simulations of the liquid corium pool. The crust shape was then adjusted iteratively until the temperature of the inner surface of the crust was the melting point of corium (a mushy zone was not modelled). The converged crust shape is then consistent with the heat flux profile exiting the outer surfaces of the CV walls. The three-dimensional model revealed strong spatial variations in the exiting heat flux, with concentrated heat flux at the junction between the subshell and the annular plate [156]. The model has been used to calculate heat fluxes exiting various locations on the CV surface, under conditions reached during various severe accident scenarios as determined by MAAP4-CANDU. The calculated exiting heat fluxes were compared to measured critical heat fluxes to assess the feasibility of in-vessel retention of corium in each accident scenario. Key findings included that the maximum exiting heat flux is very sensitivity to the weld profile. Significant uncertainties persist, arising from factors including boiling heat transfer (and heterogeneous surface temperature) in regions with complex geometries and significant exiting heat flux gradients.

6.1.2.3. Corium convection

The results of a subset of the CNL's corium convection experiments (see Section 4.2.2.2) have been reproduced with good agreement, using computational fluid dynamics (Unsteady Reynolds-Averaged Navier-Stokes) [76]. These simulations have been extended to the reactor case, with a key prediction being that most of the melt would fall below the liquidus temperature, so long as nucleate boiling is sustained on the outer surfaces of the CV [157]. Given the conditions represented in this simulation (including a decay heat corresponding to a level expected relatively early in the IVMR phase of a severe accident), it can be inferred that so long as nucleate boiling is maintained on the outer surfaces of the CV, and that vessel integrity is maintained, this 'snapshot' of the corium pool represents approximately the maximum corium pool enthalpy achieved during IVMR. Following core collapse, the terminal debris bed would partially liquefy (with only a modest fraction exceeding the liquidus temperature), reach a state represented by this simulation, and eventually solidify (due to diminishing heat generation).

6.1.2.4. Corium ingression into calandria vessel penetrations

Simulations of the ingress of corium into a CANDU end fitting have been performed using CFD at CNL [83]. The results indicate that ingress of corium above the liquidus temperature would likely not result in plug formation; however, a creeping flow of corium closer to the solidus temperature could yield a different result. Further effort is needed to determine the behaviour of the system in less bounding cases, including ingression of corium below the liquidus temperature, and ingression via small cracks in the penetration-bridging crusts. These results are therefore considered preliminary and are not yet reflected in integral codes.

6.2. STRUCTURAL INTEGRITY OF THE CALANDRIA VESSEL

Supporting computational models and tool such as Finite Element Analysis (FEA) methods can help in gaining more insight on the structural integrity in of the CV under severe accident conditions.

6.2.1. Background and calandria failure mechanisms

The in-vessel accident management strategies in PHWRs are based on the ability of CV to withstand the large thermal and mechanical loads for a required duration of time. During a severe accident scenario involving core collapse, the reactor core would relocate into the CV bottom and form a terminal debris bed. The CV would act as a barrier in limiting the progression and physical spread of the accident. It would retain the core debris, facilitating its cooling by the calandria vault water for some time. However, if no accident management actions are taken, it could eventually undergo structural failure.

Various events unfolding during the accident progression such as over-pressurization of the CV, impingement of core debris on its walls, corium-vessel interaction leading to local melting and thinning of the vessel wall, local overheating, etc. may adversely affect the capability of the CV to fully contain the core debris and cause local loss of material strength and load bearing capacity, leading to the redistribution of stresses on the wall material. If the local failure is not arrested, it may lead to the gross failure of the CV and the invalidation of in-vessel accident management strategies. The CV is considered as failed if any of the failure modes impairs its ability to retain the core debris further. The IAEA-TECDOC-1594 [1] has specified CV these failure criteria. These include failure due to creep, high pressure, debris impingement, molten metal layer attack, drain line failure due to hot molten debris at the bottom and reduced external cooling.

CV failure can result in the ejection of the core debris outside the CV and molten corium-concrete interaction with large release of hydrogen and fission gases. Therefore, information on the structural integrity of the CV and its degradation with time, leading to its failure, is crucial for formulating and assessing accident management guidelines. However, unlike the lower head failure event in light water reactor accident scenarios, the knowledge base on the structural failure of the CV under severe accident conditions in PHWRs is rather limited. Moreover, many of these numerical and experimental investigations are based on simplified geometric and loading approximations and thus represent the CV response in a very limited manner. Detailed studies, based on thermo-mechanical analyses, can assess the structural integrity of the CV under the postulated accident scenario and determine the duration of the in-calandria core debris retention.

The structural assessments need modelling of the actual stiffness of the structural elements, realistic boundary conditions, temperature distribution as well as temperature dependent plastic and creep properties of the CV material. These cannot be met with simplified geometric and loading assumptions, or empirical equations or failure assessment diagrams. Dedicated and specialized simulation tools are more suited to deal with such complex problems. Finite element methods are well-established and reliable tools for conducting such analysis with reasonable engineering approximations.

6.2.2. 540 MWe Indian heavy water reactors calandria failure assessment based on finite elements analysis results

A detailed 3D elastic-viscoplastic finite element analysis of 540 MWe Indian PHWR CV has been conducted at BARC. The sequentially coupled thermo-mechanical analysis was performed to assess the structural integrity of the CV under a postulated accident scenario and to determine the duration of the in-calandria core debris retention. The details of the study may be found in Reference [151] and are summarized in this section.

6.2.2.1. Considerations for the structural analysis

Severe accidents are complex, multi-physics phenomena. For this realistic, best-estimate assessment, it was crucial to account for the important aspects that affect the structural behaviour of the CV, including:

- Thermal analysis aspects: the thermal analysis portion of the study was carried out using ASTEC, with suitable adaptations made for PHWRs as covered in Section 5.4.3. The analysis accounts for:
  - The heat input from decay heat and metal-water reaction (Zr oxidation), the primary causs of high temperature loads and associated degradation of the CV;
  - The ultimate heat sink: the calandria vault, moderator and end shield water inventories that provide cooling to the CV and its internals;
  - The complete heat transfer path to the surroundings (see Figure 61 in Section 5.4.3), while maintaining the total heat balance;

  - The suitable discretization of the model geometry to capture the relevant aspects needed for the structural assessment, such as the temperature time histories of different parts and regions including the end shield.
- Structural analysis aspects:
  - As the accident progresses, the temperature loads become more prominent, leading to high thermal stresses. These stresses are affected by displacement compatibility constrains. It is important to correctly account for the external displacement constraints (i.e., boundary conditions), flexibility provided by the geometric features such as diaphragm and stepped shell geometry, and realistic temperature distribution. It is necessary to model the end shields in addition to the CV;
  - Both geometric and material non-linearity have been considered in the analysis along with the temperature dependent tensile and creep deformation properties of the CV material;
  - Apart from the fixed load due to debris dead-weight and self-weight, transient loads, temperatures in different regions and parts of CV-end shield assembly, as well as the pressure exerted by gradually depleting calandria vault water have been considered.
- Initial conditions:
  - The instant of core collapse and the formation of terminal debris bed were selected as suitable event/stage during the accident progression as the initial condition for structural analysis.

Severe accident simulation needs extensive modelling and computational resources and has inherent uncertainty. Since the structural integrity of the CV is threatened during the later phase of the accident, structural modelling of the CV throughout the accident progression starting from the initiating even is impractical. In view of this, a suitable phase during the accident progression, when onwards the structural assessment of the CV is warranted, may be selected as the initial condition for the structural analysis. The state of the reactor at this instant, such as temperature, decay heat, water levels etc. may be evaluated from separate analyses. The determination of initial conditions for the SCDA phase of the analysis depends on various factors such as the postulated accident scenario, the time after the initial event, whether LCDA to SCDA transition is completed, if any accident management actions have been taken, etc. For the present study, the initial conditions were considered at the instant of core collapse and the formation of terminal the debris bed.

6.2.2.2. Scenario and modelling assumptions

The accident scenario postulated for the study was an unmitigated total loss of heat sinks, i.e., simultaneous occurrence of large break LOCA, loss of ECCS, loss of end shield cooling system, loss of moderator cooling system and loss of calandria vault water cooling system. In addition, no accident management actions were considered. Figure 64 shows the FE model which compromises of an integrated ¼ CV and end shield assembly with vertical symmetry planes.

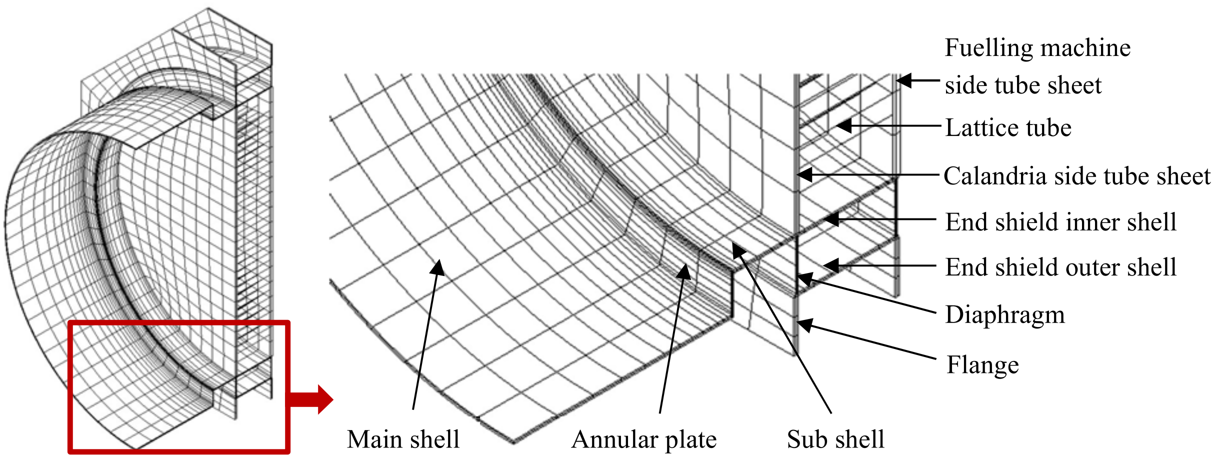


FIG. 64. Finite element model of the calandria - end shield assembly (courtesy of K. Mohta, BARC).

Tensile and creep stresses rupture properties of CV material, SS 304L, were generated in-house for temperatures up to 1100°C and were used in the analysis [158]. Both material and geometric non-linearity were considered. Steady mechanical loads due to dead-weight of core debris, carbon steel balls in end shield and self-weight were considered. Transient loads included thermal stresses due to evolving temperature conditions and hydrostatic pressure exerted by the calandria vault water and the end shield water. The information of transient temperature loading was obtained from a thermal analysis.

6.2.2.3. Results

The FEA results were post-processed to assess the failure of the CV and end shield assembly. The study assessed gross failure of the CV due to creep and plasticity per the following criteria:

- Plastic instability: it arises when the CV is not able to further withstand the dead loads. In the present case, the plastic instability has set in at approximately 74.4 h after the core collapse;
- Large inelastic strains: Failure is assumed when the through-thickness inelastic (plastic and creep) strains in a local region exceeds the 10% strain threshold. In the present case, it was observed at around 71 h to 73 h at different locations;
- Creep stress rupture: Failure due to creep stress rupture was assessed using Larson-Miller parameter and Robinson's damage accumulation rule based on linear summation of individual creep damages (partial life fractions) [159]. The Larson-Miller parameter correlations for the material were determined from the in-house creep-stress rupture tests. The criteria indicated failure at around 71 h to 73 h at different locations.

The CV failure was estimated at roughly 71 h after the core collapse due to plasticity and creep induced failure. Core debris bed melting, and formation of corium pool was not observed by this time instant. Therefore, failure due to molten metal layer attack or failure of drain line is unlikely.

6.2.3. Conclusions

The structural analysis performed by BARC has provided significant insights about the 540 MWe Indian PHWR CV response under severe accident conditions. It has shown that a reasonable time frame of about 3 days is available for operators to take suitable accident management actions before the gross failure of the CV takes place. It has also confirmed that the inherent design features of PHWRs such as the large water inventories and the low power density delay the accident progression and aid in managing the accident.

However, the study has limitations due to its specificity to reactor size (geometry) and postulated accident scenario. The observations and insights obtained from the present analysis are therefore only qualitatively applicable to PHWRs with similar configuration but having different reactor power and core geometry, and the estimation of the CV failure timeline cannot be generalized. Thus, separate analyses are needed for different PHWR designs and postulated accident scenarios. Such analyses could serve in the future, to develop a simplified procedure for assessment of structural integrity of the CV, based on few parameters or variables that reflects the state of reactor (such as decay heat status, available water inventory etc.).

In addition, assessment of the CV failure due to local overheating from exceeding CHF at geometric features such as nozzle junctions etc., dynamic loading arising from falling of the core debris on the CV bottom etc. was not covered in the present study.

6.3. CONTAINMENT HYDROGEN DISTRIBUTION AND HYDROGEN MITIGATION SYSTEMS

Three-dimensional codes are capable of conducting detailed calculations for hydrogen distribution in the containment and assessing the effectiveness of hydrogen risk mitigation systems. Those codes include general purpose thermalhydraulics software and CFD methods. The phenomena, tools and methods discussed in this section are not specific to PHWRs but applicable to most LWRs. They are however still included in this document for completeness.

6.3.1. General purpose three dimensional thermalhydraulics codes

Canada has adopted GOTHIC as part of their Industry Standard Tool Set (IST) for nuclear plant analysis of containments, which is therefore reflected in most CANDU-6 safety cases worldwide.

GOTHIC is a computer program for containment thermalhydraulics, designed to handle multi-phase flows and multi-component. It resolves the conservation equations for mass, momentum, and energy. The area of interest can be represented using a combination of multidimensional and lumped volumes. Empirical correlations are used to calculate the heat transfer convection, condensation, and evaporation between the fluid and the structures. Multidimensional analysis can be conducted using models for both molecular and turbulent diffusion. GOTHIC incorporates specialized features for modelling and controlling equipment commonly found in NPP containment systems, such as pumps, valves, fans, heat exchangers, etc. Its versatile set of equations enables GOTHIC to address multi-physics cases, and its adaptable nodalization options offer efficient solutions suitable for multi-scale applications [160-161].

The qualification report of GOTHIC, detailing V&V across various single and two-phase flow scenarios, undergoes regular updates and is released alongside each version of the software [162].

In CANDU safety analysis, GOTHIC typically uses as inputs the releases into the containment computed by a system code for a given accident scenario. The code then models the containment response, including temperature and pressure, hydrogen concentration, activation of containment sprays, recombiners, etc. Evaluation of flammability limits and radioactive releases to the environment can be calculated through coupling with dedicated codes. GOTHIC is traditionally used for analysis of design basis accidents in CANDU application, but its applicability can be extended into the severe accident analysis domain. Analytical simulations have been performed by CNL that confirmed GOTHIC-IST code's capability to capture hydrogen stratification in safety analysis and confirmed the implementation of risk mitigation provisions at CANDU NPPs during the ex-vessel phase of an accident (see References [163-165]).

6.3.2. Usage of computational fluid dynamics

In a severe accident, the preservation of the reactor containment is required to avoid significant release of radioactive products into the environment. To this end, SAMGs and safety systems aim at reducing the containment pressure (spray or filtered containment venting system) or the hydrogen content (recombiners and igniters) and prevent fast hydrogen deflagration.

To assess the performance of these systems and processes, numerical simulations of representative and extreme scenarios are generally carried out using lumped parameter codes. 3D and CFD codes are used in complementary way to study in depth the hydrogen dispersion and the possibility of gas stratification, which cannot easily be simulated with the lumped parameter codes. The main advantage of CFD codes is that they are applicable to a wider range of conditions than lumped parameter codes [166]. Figure 65 presents the main features of both lumped parameter and CFD codes.

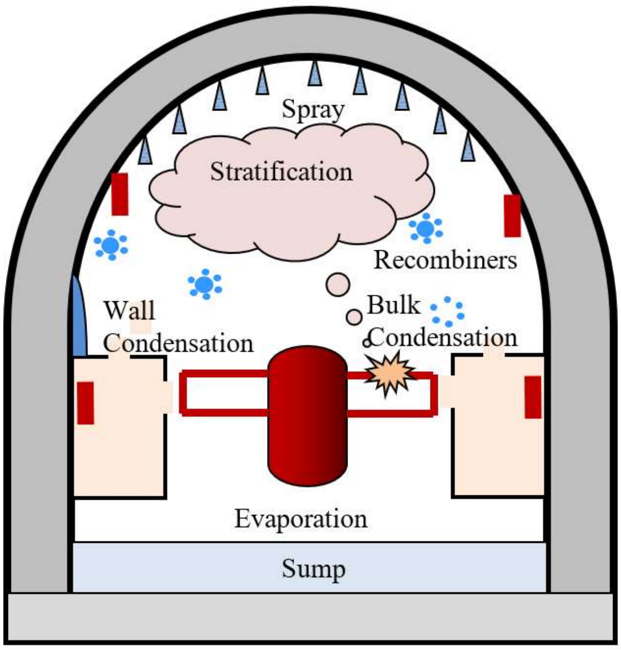


FIG. 65. Representation of various phenomena in the containment.

It is crucial to accurately predict hydrogen concentrations in NPP containments by considering local instantaneous atmosphere mixing and stratification. The usage of CFD codes enables the simulation of such scenarios. Commercial CFD codes however do not have all necessary models needed for performing hydrogen distribution simulations. Figure 65 schematically represents these models that need be implemented into the CFD code before performing hydrogen distribution calculations. This includes the appropriate selection of models for turbulence, wall condensation, spray system actuation, hydrogen recombination in addition to bulk condensation, and sump water evaporation. A short overview of the commonly adopted models is given in Appendix IV. More details can be sought in Reference [167].

6.3.2.1.Examples of application of CFD codes to containments

A number of CFD activities for Indian PHWRs containment safety application have been carried out in India in recent past and are discussed in reference [168]. These include hydrogen distribution studies in post-accident containment atmospheres, modelling of hydrogen recombination in PARs under accident conditions and modelling of hydrogen combustion for containment safety applications etc. Further information may be found in references [169-171].

CFD codes for prediction containment behaviour under accident conditions have however not been systematically adopted in PHWR MS. An example, taken from the framework of the French EPR Flamanville to assess the efficiency of the implemented hydrogen control systems in PWRs, is therefore used in this section to reflect a possible methodology that can be extended to PHWRs [172]. The methodology adopted in this example consisted of the following seven steps:

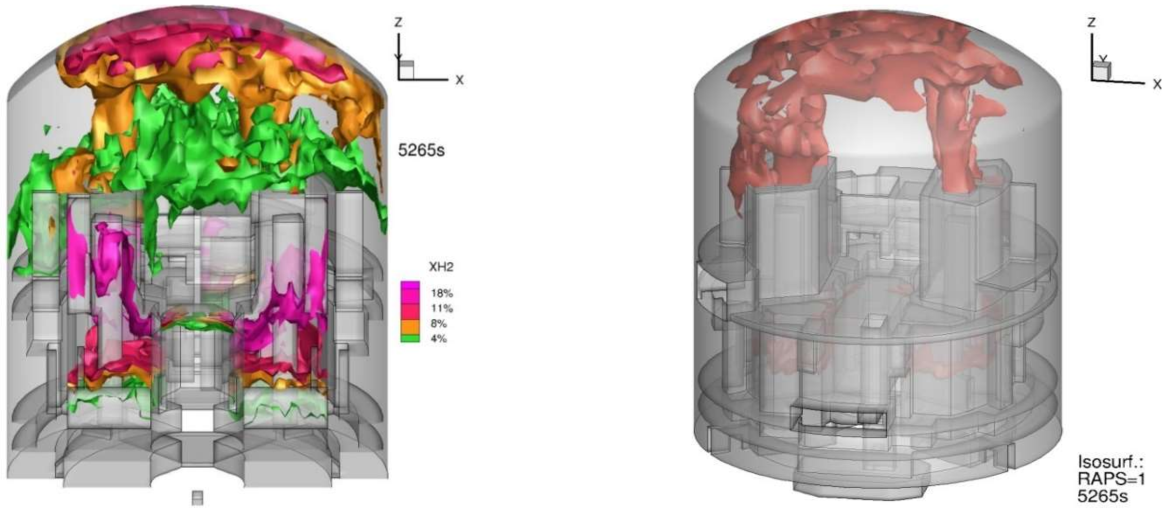
Step 1: Plant design.
The starting point is the geometrical modelling of the containment. This includes defining the containment volumes, walls, and other geometries. The safety systems such as PARs, coolers, spray systems, etc., are also included since those significantly impact the distribution of hydrogen within the reactor containment.

Step 2: Select the relevant scenarios.
The choice of pertinent scenarios is determined by PSA results, encompassing both representative and bounding severe accident scenarios.

Step 3: Evaluate the atmosphere condition in the containment.
The temperature, pressure, and gas composition in various containments regions are established throughout the transient while considering the action of mitigation systems. CFD codes allow for deeper analysis in short-term temporal windows, e.g., hydrogen release phase and scenarios with local accumulation and stratification (see Figure 66-a).

Step 4: Evaluate the time evolution of flammable cloud based on the well-established flammability limits considering the representative conditions of a severe accident.

Step 5: Evaluate the propensity of a premixed flame to propagate inside the reactor containment using adequate flame acceleration criteria (see Figure 66-b).



(a) Hydrogen distribution inside reactor containment;

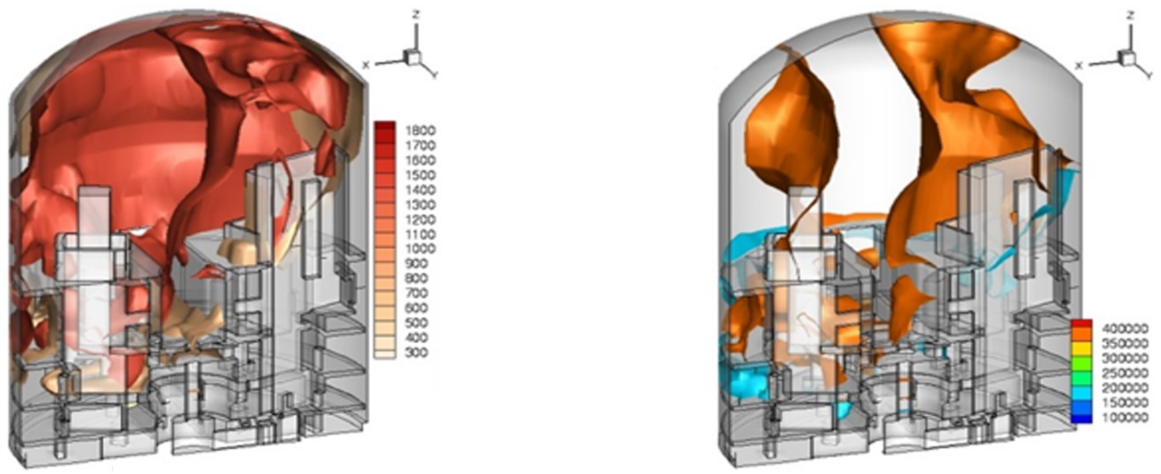
(b) Gas mixture cloud satisfying flame acceleration criterion.

FIG. 66. Hydrogen distribution and gas mixture inside the reactor containment (courtesy of IRSN, France).

Step 6: Evaluate the pressure and thermal loads generated by combustion (see Figure 67-b).

There are two configurations, depending on the flame acceleration criterion:

i. If the criterion is not met, the dynamic pressure loads are omitted and assessed by assuming an adiabatic isochoric complete combustion process.
ii. If the criterion is met, the combustion loads are calculated using the most suitable combustion models, primarily CFD codes.



(a) Combustion at 3000s located at (10,0,30), temperature at t=55ms is 800 and 1500K.

(b) Combustion at 3000s located at (10,0,30), pressure at t=55ms is 2 and 4 bars.

FIG. 67. Examples of pressure and temperature loads induced by combustion (courtesy of IRSN, France).

Step 7: Assess the response of the structure and safety components to pressure and thermal loads resulting from combustion. The containment's structural integrity can be examined using specialized dynamic structural response codes.

6.4. CONCLUSIONS

As discussed in Section 5, lumped parameter integral codes have in general achieved the capabilities to simulate the entire transient over several days and through the various phases of the severe accident progression in PHWRs. Many improvements have been implemented, and supported through experimental programmes, discussed in Section 4. While derivation of empirical models and validation of code calculations based on direct experimental data is preferred, this is not always possible or economically achievable in many of the complex severe accident processes, and thus engineering judgement was often used. In such circumstances, alternative modelling techniques have in many cases replaced or refined the previously applied engineering judgements. This section has provided some examples relevant and specific to PHWR severe accidents, including:

- CFD for in-core behaviour of disassembled fuel channel segments, solid/molten corium, and crust;
- CFD and 3D codes for hydrogen behaviour and mitigation in containment;
- FEA for structural components behaviour (e.g., CV).

Specific results and conclusions from the various supporting analyses were summarized in the respective sub sections, with more detail in references, and showed that coupled and/or competing effects can be simulated to avoid overly conservative assumptions or usage of engineering judgement in deriving empirical models or criteria. An additional benefit from parametric support calculations could also be in aiding uncertainty characterization for integral code uncertainty assessments.

An example of resolving competing effects looked at those arising from increased steam ingress into a channel segment, that increases oxidation but also increases cooling of the fuel. The combined CFD and FEA analysis of a 540-MWe Indian PHWR CV structural failure analysis provided an example of resolving failure criteria. In the cases of detailed CFD calculations of corium and containment behaviour, the aim was to spatially resolve the behaviour and the resulting threats to accident progression and failure criteria in such a way that it can be translated into improved phenomenological models and accident progression criteria in the integral codes.

Before implementation of results into integral codes, however, care has to be taken that they are applicable to the reactor design, accident scenario, and application goals of the severe accident analysis performed. It is also important that all relevant phenomena involved in the detailed behaviour are properly implemented in these special purpose models, which are typically based on commercially available general-purpose tools that may lack some of the phenomenological models.

7. CONCLUSIONS

This publication captures the recent initiatives in advancing, validating, and implementing codes for severe accident simulation and modelling in PHWRs, along with the associated research and development activities. The content is based on the information shared during a Technical Meeting organized by the IAEA in 2022 on PHWR severe accident simulation and modelling and brings updates to the information documented in past IAEA key publications on PHWR severe accident, namely the IAEA-TECDOC-1594 "Analysis of Severe Accidents in Pressurized Heavy Water Reactors" [1] and the IAEA-TECDOC-1727 “Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications” [2].

Over the past decade since the Fukushima Daiichi NPP accident, the scope and application of severe accident analysis have expanded within deterministic and probabilistic safety analysis frameworks. Severe accidents analyses and enhanced tools and methods are also increasingly contributing to the technical basis for severe accident management and supporting emergency response. In parallel, plant simulators have enhanced their capabilities to cover the severe accident domain for training and visualization. The regulatory framework for those applications has also evolved in the recent years. Although slight differences exist between PHWR MS in their formulation of regulatory requirements for severe accidents, they agree in essence.

This publication highlights the advancements in the severe accident experimental database for PHWRs. Significant improvements have been made in the understanding the CT properties at high temperatures, the fuel channel heat-up and breakup and in the behaviour of the suspended debris, core collapse, terminal debris and corium. The understanding of the CV failure regarding the critical heat flux, and load and thermal stress distributions has also increased. In the end shields, several studies have improved the predictive capabilities of the critical heat flux in a porous medium abutted by a tubesheet. Furthermore, significant progress has been made towards managing combustible gases under accident conditions, including assessing the effect of the different properties of H_2 and D_2 on the performance behaviour of hydrogen mitigation systems.

All these advances have increased the knowledge base and reduced the uncertainty level of specific severe accident phenomena. However, some phenomena identified by the PIRTs as requiring additional R&D to refine their existing knowledge and understanding remain to be investigated. The Canadian and Indian experimental programmes incorporate a portion of those in their medium to long-term planning, after which a consolidation of the existing PIRTs and their reconciliation with the updated knowledge level could be performed.

The advances in knowledge for severe accident phenomena, combined with the increases in computational capabilities and more stringent requirements have motivated advances in simulation codes for severe accidents in PHWR MSs. For instance, this publication has documented that PHWR-dedicated analysis codes, such as MAAP5-CANDU and CAISER, have benefited from refined mechanistic models, empirical correlations and criteria as well as improved nodalization details for more accurate modelling of the core collapse progression, including channel failures, suspended, and terminal debris beds. This approach is also being adopted in the new technology-neutral Indian code PRABHAVINI. In parallel, LWR analysis codes such as RELAP5/SCDAPSIM and ASTEC have been adapted for use in PHWR severe accident simulations. The incorporation into simulation codes of the continuously increasing knowledge level on severe accident phenomena will continue to require persistent and systematic efforts.

While derivation of empirical models and validation of code calculations based on direct experimental data is preferred, this is not always possible or economically achievable in many of the complex severe accident processes. This publication provides examples of alternative modelling techniques that have the potential to replace engineering judgements when use of direct experimental validation data is not achievable. With time and additional refinements, those alternate methods could complete some of the remaining gaps in the understanding of specific severe accident phenomena.

APPENDIX I – INPUTS FOR THE DEVELOPMENT OF GUIDELINES AND PROCEDURES

Safety analyses provide various inputs for development of procedures and guidelines. Examples are provided below.

(a) Criteria for entering and exiting accident management guidelines;

(i) Conditions for entering and exiting SAGs and SCGs as applicable include;

Entry Criteria:
- Low sub-cooling margin in inlet/outlet headers;
- Moderator low level;
- Calandria vault water level;
- High radiation level/dose rate in containment (which may be inferred from measurements outside containment);
- Steam generator low level;
- Hydrogen concentration;
- Reactor building/containment pressure;
- Reactor building/containment water level (ECC recovery sump tank);
- Spent/irradiated fuel bay low level;
- Spent/irradiated fuel bay measured dose rate at various locations;
- Events that may cause breach of dry fuel or solid radioactive waste management structures.

Exit Criteria:
- Reactor core temperature < X* AND stable or decreasing;
- Dose rate < site emergency levels AND stable or decreasing;
- Pressure inside containment < X* AND stable or decreasing;
- Hydrogen concentration inside containment < 4% in dry air AND stable or decreasing;
- Availability of ultimate heat sink (for example moderator level above upper-most fuel channels and stable).

*X = Certain value of a given parameter

(ii) Specification of set-points to initiate and to exit individual strategies.
Operation of containment spray and Containment Filtered Vent System when hydrogen concentration in the containment is below limits for any combustion risk issue.

(b) Positive and negative impacts of accident management actions;

(i) Positive impacts

- Operation of hook-up water injection to CV removing the heat;
- Injection of water into SG removes heat through natural circulation;
- Containment spray reducing the containment pressure;
- Operation of building air coolers reducing the containment pressure;
- Recombiners operation reduces the hydrogen concentration in the containment;
- Operation of Containment Filtered Vent System reducing the containment pressure.

(ii) Negative impacts

— Containment spray system activation may increase the flammability mixture concentration;
— Injection of water into the core at high temperatures may lead to enhanced hydrogen generation;
— Operation of containment filtered venting system considering design specification, design constraints and potential sheltering requirements that may affect emergency response actions.

(c) Time windows & symptoms for diagnosis and monitoring;

(i) Time windows available for performing the actions;

— Time available for injection of water to CV;
— Time available for injection of water to calandria vault;
— Time available for water injection to steam generator for maintaining thermo-syphoning as a preventive measure;

(ii) Choice of symptoms (i.e., parameters and their values) for diagnosis and monitoring the course of the accidents (i.e., to determine the reactor core condition, state of protective barriers, etc.).

— Water level in decay heat removal condenser;
— Primary Heat Transport System (PHTS) temperature, reactor headers temperature difference or sub-cooling margin;
— Water level in the containment (sump/suppression pool level);
— Containment pressure;
— Water level in the CV;
— Water level in the calandria vault.

(d) Parameters related to systems and barriers;

(i) Identification of the key challenges and vulnerable plant systems and barriers

— Challenge to containment integrity due to high hydrogen concentration or containment over pressure and temperature;
— Safety systems or engineered safety features designed for DBAs but operated beyond capacity during severe accidents.

(ii) Prioritization and optimization of strategies with respect to achieving safety functions;

If multiple strategies are available for the same function. Adopting the best-suited strategy for a particular scenario.

(iii)Evaluation of capability of systems to perform intended functions.

Capabilities of the systems intended for use in accident management may be evaluated:
— Availability/unavailability of systems intended for use in accident management;
— Effectiveness of selected accident strategies after they have been applied.

(e) Computational aids and its development.

Safety analysis also give inputs for the development of computational aids including accident progression and timings, containment hydrogen concentration and flammability, and rate of water additional for decay heat removal and to maintain or increase the moderator level.

APPENDIX II - SUPPLEMENT TO EXPERIMENTAL PROGRAMME UPDATES IN INDIA – STUDIES SPECIFIC TO LIMITTED CORE DAMAGE PHASE

II.1 FUEL CLAD INTEGRITY STUDY

Assessment of integrity of fuel pins under accident conditions is an important aspect for fuel element design considerations. The fuel cladding acts as an important physical barrier to the release of radioactivity from the fuel to the heat transport system. An experimental investigation of high temperature deformation and rupture behaviour of Indian PHWRs collapsible fuel cladding, made up of Zircaloy-4, was performed under oxidizing steam atmosphere [173]. The experimental setup consisted of a direct electrical heating system for heating of the cladding with controlled power supply, a gas handling system to pressurize the cladding internally to simulate the internal gas pressure, and measurements of cladding surface temperature at multiple locations with K-type thermocouples connected to a data acquisition unit. The transient heating was carried out with various internal pressures and heating rates.

The cladding burst temperature was found to be in the range of 598-1123^{o}C. The cladding bust pressure was found to be higher for the α-phase than the β-phase of Zr and showed an intermediate value for α+β phase. It is observed that the strain limit is affected by the phase transformation, showing a local maximum near the transition region. Based on these observations burst stress correlations are developed factoring in the content of cladding oxidation [173].

II.2 BEHAVIOUR OF IRRADIATED PHWR FUEL PINS

To understand the behaviour of PHWR fuel pins under extended burn up conditions experiments were performed with irradiated PHWR fuel pins with burn up of 15,000 MWd/tU at elevated temperatures of 700-1300°C inside hot cells [174]. A special furnace was used to performed localized heating of 100 mm length of the fuel pin under argon gas environment or in air atmosphere. Ballooning and formation of micro cracks was observed after 10 mins of heating due to release of fission gases from the fuel pins with internal pressure of 2.1–2.7 MPa for temperatures lower than 800°C. For high temperature cases of 1300^{0}C, the fuel clad was found to undergo complete rupture due to oxidation induced embrittlement of the cladding. These clads indicated presence of intra-granular and intergranular bubbles.

Another investigation on fuel pins of a PHWR fuel bundle use in NPP with fuel burnup of 7528 MWd/tU was also performed [175]. The pins were subjected to puncture test creep rate investigation studies. The outer ring fuel pins were found to have fission gas release of about 8% and the gas pressure was found to be 0.55 ± 0.05 MPa at room temperature. The release was found to be very low for the middle ring fuel pins and for the central pin. Creep rate investigation studies evolved into creep rate correlations for the cladding (Zircaloy-2) at different temperatures ranging from 800 °C to 900 °C [175].

II.3 HIGH TEMPERATURE OXIDATION REACTION STUDY FOR PRESSURE TUBE

The present generation of Indian PHWR PTs are manufactured from Zr–2.5%Nb alloy. The properties of this alloy include low pickup of deuterium as compared to Zircaloy-2, lower creep rate, higher strength and better corrosion resistance. The corrosion behaviour of Zr-2.5%Nb is found to be dependent on its microstructure. For analysing the behaviour of PHWR PTs under postulated LOCA conditions, a study was carried out to understand steam oxidation kinetics and evolution of microstructure during oxidation in the temperature range of 500–1050°C [176].

The experimental setup consisted of a resistance furnace for controlled temperature environment containing steam generated from an electric steam generator. The test coupon weights were measured before and after the experiment. A parabolic rate constant fitted in the form of Arrhenius equation was evolved from the studies. The hydrogen pick up was observed to be less in the α-region than the β-phase region. Formation of oxygen stabilised α-Zr layer was found to increase hardness near the oxide-metal interface [176].

II.4 FUEL CHANNEL INTEGRITY STUDY

The PT and CT constitute the fuel channel of PHWR which acts as one of the physical barriers to radioactivity release, subsequent to fuel failure. There are several phenomena that can contribute to the failure of PHWR channels such as reduction in material strength due to oxidation reactions (discussed in the previous sub-section), degradation in the heat transfer from channel external surface with intact or degraded bundle state, ballooning and sagging deformation of PT and effectiveness of heat transfer by contact between the PT and CT. Many of these phenomena take place simultaneously, however, an effort is made towards studying each of these phenomena separately with a dedicated experimental setup that minimizes or altogether eliminates effects of other phenomena. The channel integrity is maintained in most of these phenomena which are discussed in this section. However, some phenomena are found to induce channel failure and hence, they are discussed under SCDA specific studies.

II.4.1 Fuel channel heatup – heat transfer with intact / slumped bundle

The heat transfer from the fuel pins to the PT, from the PT to the CT and from the CT to the moderator determines success of moderator as heat sink in maintaining the fuel channel integrity. The heat transfer from the fuel pins to the PT is affected by the state of fuel bundle; intact (fuel pins are arranged in a form of bundle, as fabricated) or slumped (fuel pins are disassembled from the end plates). Figure II.1 shows intact and slumped bundle configurations.

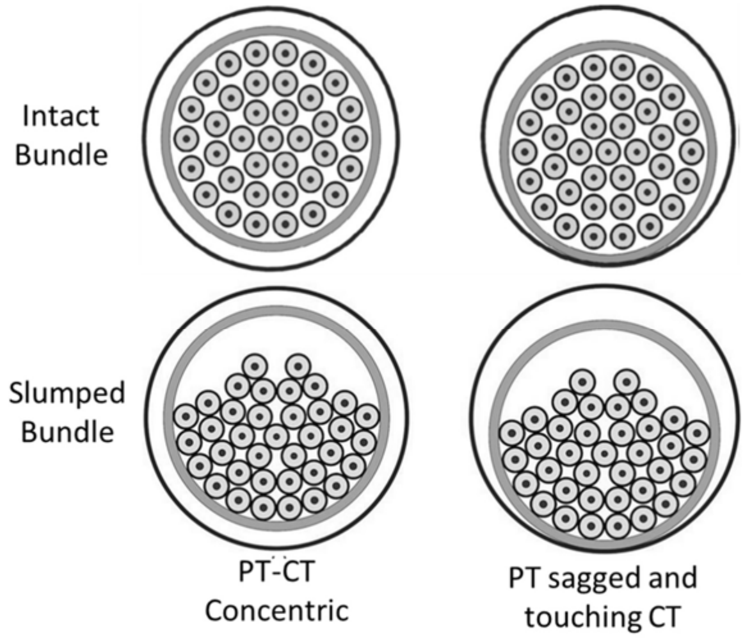


FIG. II-1. Intact and slumped bundle configurations for experiments (reproduced from Ref. [177] with permission courtesy of Elsevier).

In-order to study the heat transfer from the fuel pins under intact [178] and slumped bundle configurations [177] an experimental setup was built consisting of a 37-element fuel bundle simulator (FBS) with intact or slumped geometrical configuration, a PT, a CT and the cooling water tank. The cooling water tank was used to simulate the moderator condition. The PT was arranged either concentrically with the CT or was made to sit on the CT simulating sagged PT conditions.

For intact fuel configuration, the 37-element intact fuel bundle simulator with simulation of radial power profile was arranged concentrically with the PT. The PT was kept concentric to the CT. Temperature of different fuel pins simulators, the PT and the CT at various axial and azimuthal locations was measured with K-type thermocouples. No significant variation was observed for the PT and CT temperatures in the azimuthal direction for the fuel pin simulator. The maximum fuel pin simulator (FPS) cladding and PT temperature was limited to 970°C and 730°C respectively [178].

A similar set of experiments were also performed for an assumed slumped configuration of 37-fuel pin simulators arranged within the PT. The arrangement of PT-CT was varied from concentric to the PT sitting on the CT. A significant temperature gradient in the azimuthal direction was observed for the FPS in contact with the PT as well as for the PT in both arrangements of PT-CT [177].

In all these experiments, the effectiveness of moderator as heat sink was demonstrated to be adequate to limit FBS, PT and CT temperature rise. The integrity of the fuel channel was maintained in all the experiments.

II.4.2 Fuel channel integrity under pressure tube ballooning

The existence of high fuel temperature due to channel voiding and high pressure in the primary heat transport system in initiating events such as stagnation break LOCA with loss of ECCS can lead to ballooning of PT. The area of contact between the PT and the CT is expected to be higher in the ballooning case as compared to the PT sagging case and hence, is likely to affect the PT-CT heat transfer significantly. The behaviour of the PT under deformation conditions is dominated by the structural properties of the PT, which in turn depend on the fabrication route adopted. The Indian PT is manufactured by two-stage hot extrusion followed by annealing and pilgering in two stages to achieve final PT dimensions.

In order to assess PT behaviour under ballooning conditions, an experiment set was performed with short length (1.5 m) PT specimen, electrically heated and internally pressurized with argon gas (4-6 MPa) [179]. The measurement of PT ballooning was performed with a linear variable differential transmitter deflection measurement device. The PT was surrounded by the CT, which was kept submerged in water at 60°C, simulating moderator heat sink. With increase in PT temperature, the PT was observed to undergo sagging at 500°C followed by ballooning. No increase in PT temperature was observed post establishment of PT-CT contact, indicating adequacy of the moderator as heat sink. The transverse creep correlations used for CANDU PT material were found to be applicable for Indian PT material as well [179].

II.4.3 Fuel channel integrity under pressure tube sagging

The deformation route followed by the PT under low internal pressure includes sagging due to its own weight as well as due to the weight of the fuel bundle. The sagging behaviour is governed by the type of supports the PT experiences, namely the supports at the ends and the garter springs that maintain the spacing between the PT and the CT.

The phenomena of PT sagging was studied experimentally for a short length scaled down experimental setup and a full-scale experimental setup [180]. The short length scaled down experimental setup made use of a 2 m length PT scaled down to conserve the structural behaviour of the PT. The end supports for the PT were simulated with one-end fixed and one-end free conditions. To simulate the weight of the fuel bundles inside the PT, cast steel solid cylinders were used. The PT was mounted concentrically with the CT which was kept within a tank such that either an adiabatic-insulated boundary condition or a boundary condition to simulate moderator as heat sink can be used. Garter springs were not used to maintain PT-CT gap in this set of experiments. All other arrangements for PT heating, temperature measurement and deformation measurement were similar to the experiment for PT ballooning. The sagging of the PT was found to initiate for PT temperature near about 450°C and a contact between the PT and the CT was established in the temperature range of 585-610°C. The contact angle was found to be 70^0. The temperature rise in the PT was arrested after a contact with the CT was established due to sagging. The integrity of the PT was maintained throughout the experiment without any occurrence of critical heat flux on the PT surface, indicating efficient cooling from simulated moderator heat sink [180].

Experiments were also performed to study sagging characteristics of a full-scale PT-CT combination with cast steel weight inserts used as fuel bundle weight simulators [181]. The PT, having 5.4 m length was kept concentrically inside CT of length 5 m with the help of garter springs to maintain the PT-CT gap. The CT was kept inside a 5 m long water tank which was used to simulate the moderator as a heat sink. The observations for the full-scale PT-CT sagging tests were similar to the once observed for the scaled down PT-CT test. Sagging initiation was observed to occur at about 460°C PT temperature. PT-CT contact occurred in the temperature range of 665-669°C at a bottom location away from the garter springs. In these tests as well, the integrity of the PT-CT was maintained without occurrence of critical heat flux on PT-CT surface, confirming efficacy of moderator as heat sink [181].

II.4.4 Fuel channel integrity under pressure tube sagging and ballooning

Investigations were also made for the simultaneous effect of PT sagging and ballooning [182]. The experimental setup described in the previous section was modified for this purpose to facilitate internal pressurization of the PT using argon gas up to a pressure of 2.0 MPa. Experiments showed a reduction in PT temperature after establishment of PT-CT contact due to sagging and ballooning. Integrity of the PT was maintained during the entire transient [182].

II.4.5 Effect of thermal contact conductance

In the above-mentioned studies, the integrity of the PT is maintained due to sufficient cooling provided by the moderator as a heat sink. The transfer of heat from the PT to the moderator depends on the thermal contact conductance between the PT and the CT. Thermal contact conductance also plays a significant role in the heat transfer through the contact of fuel pin bundle (intact or slumped condition) with the PT inner surface. The numerical value of contact conductance is crucial for performing numerical simulation involving PT-CT contact heat transfer.

In-order to quantify the thermal contact conductance experiments were performed in a specially designed thermal contact conductance evaluation setup [122] and [183]. The setup consisted of samples of PT and CT material in contact with each other under specific contact pressure and controlled environment. Heat energy was imparted to the copper attachments connected to the PT specimen through radiation heat transfer mode and was removed from the copper attachment connected to the CT specimen. Heat flux through the PT-CT contact was measured with the help of a heat flux meter as well as calculated through temperature measurements at various locations. Special care was taken to reduce heat losses through pathways other than the PT-CT contact path by employing special insulation material and controlling the environment in the setup with either CO_2 or vacuum. The PT-CT contact conductance was evaluated in the form of correlations in terms of contact pressure and contact temperature. A similar exercise was also performed for the contact between fuel cladding and inner surface of the PT.

All these studies described in this section, have helped in quantifying deformations of the PT under different conditions, quantification of thermal contact conductance and demonstrating the efficacy of the moderator as a heat sink in maintaining the integrity of the PT, an important physical barrier against the release of activity.

II.5 MOLTEN FUEL COOLANT INTERACTION STUDIES

In a postulated accident scenario of stagnation channel break, the melt generated within the fuel channels may come out and interact with moderator surrounding the channel. Such a Molten Fuel Coolant Interaction (MFCI) differs from the anticipated MFCI in LWR due to presence of sub-cooled moderator in large quantity and also due to the presence of neighbouring intact channels that can interfere with spreading of molten fuel. The process of MFCI involves multiple, complex and simultaneously occurring phenomena and hence they are studied by dividing the MFCI process in multiple phases. Experiments are being performed on a small scale to study:

- Phase I: film boiling on solid spheres;
- Phase II: hydraulics of cold ball injected into water pool;
- Phase III: thermalhydraulics of hot ball injected into water pool;
- Phase IV: thermalhydraulics of molten material discharged into water pool;
- Phase V: thermalhydraulics of molten material discharged in presence of neighbouring channels in a scaled down PHWR specific geometry.

A dedicated experimental setup is operational with woods metal as simulant molten fuel. Experiments are underway to analyse the fragments generated from the simulant molten fuel due to MFCI. It is also planned to further carry out experiment on a larger scale with 1:3 scaled down CV geometry with internal channels. The molten material used as simulant in this case will be molten alumina, which will be melted in an induction furnace and injected at a channel rupture location within the CV along with a simulated in-core LOCA flow through the ruptured channel.

II.6 STRUCTURAL MATERIAL PROPERTY EVALUATION PROGRAMME

The analysis of severe accident conditions for PHWR requires substantial amount of material property database in a temperature-pressure combination range much beyond the operating ranges of the material generally used in other industrial applications. The results of the analysis are sensitive to the material property data. An effort has been made to compile the mechanical, creep, fracture and fatigue properties of various nuclear materials, tested in India over past several years. The compilation is available in the form of Compendium of Indian NPP Material Properties spanning over eleven volumes, discussing in details of various materials such as ferritic low alloy steel, austenitic and martensitic stainless steel, dissimilar metal welds, zirconium-based alloys to name a few [184].

APPENDIX III - SUPPLEMENT TO MAAP-CANDU DESCRIPTION – FISSION PRODUCT MODEL IN MAAP-CANDU

This appendix documents the fission product model implemented in MAAP-CANDU as well as the upgrades made to it between MAAP4-CANDU and MAAP5-CANDU.

III.1 FISSION PRODUCT MODEL IN MAAP-CANDU

MAAP5-CANDU has the following options for the correlations used to calculate FP and structural material release from the core. The user selects an option by setting parameter FPRAT value. Additional in-vessel FP models are available in MAAP5-CANDU, with the following correlations:

(1) NUREG-0772 [185] for volatiles + Kelly's correlations [186] for non-volatiles (low volatility);
(2) NUREG-0956 [187] for volatiles + Kelly's correlations for non-volatiles (low volatility);
(3) CORSOR-M model [188];
(4) CORSOR-O model [188];
(5) ORNL-BOOTH model [188];
(6) CORSOR-M model for volatile FPs (noble gases, CsI, and CsOH) and CORSOR-O model for non-volatile FPs;
(7) ORNL-BOOTH model for volatile FPs (noble gases, CsI, CsOH, Te, and Sb, plus elemental iodine and organic iodine) and CORSOR-O model for non-volatile FPs;
(8) Cubicciotti Steam Oxidation model [189] for volatiles + Kelly's correlations for non-volatiles.

The first two models (NUREG-0772 + Kelly's and NUREG-0956 + Kelly's) mentioned above were previously available in MAAP4-CANDU, but the remaining six (bullets 3 to 8 above) models are newly introduced in MAAP5-CANDU. Additionally, the user-input flag FFPREL functionality was extended, which now allows user to control in-vessel FP release rates custom to each of fifteen FP groups.

III.2 IODINE CHEMISTRY MODELS

The iodine chemistry models are de-activated in the default input selections for MAAP-CANDU accident simulations, meaning that these models are not used to calculate FP release into the environment. The user can apply the iodine chemistry models to the simulation if desired using logic in the input file.

An assessment of iodine chemistry models in MAAP5-CANDU discussed herein includes a general description of the iodine models, chemical reactions involving iodine, and iodine-related input and output variables in the code.

Iodine releases from damaged fuel into the reactor vessel are usually in gaseous form (e.g., I_2). However, these gaseous releases may rapidly react with other chemical elements released to the reactor vessel at the same time, in particular caesium, rubidium, and silver (found in some control rods, Ag-I-Cd), which are present in large amounts. The compounds formed, such as CsI and RbI, can condense onto droplets, or even solidify, within PHTS or containment. These materials are highly mobile, which is driven by their relatively low melting and boiling points, shown (for I_2, CsI, and RbI) in Table III.1.

TABLE III.1. MELTING AND BOILING POINTS OF IODINE SPECIES (AT ATMOSPHERIC PRESSURE)

Compound	Melting Point (°C)	Boiling Point (°C)
I_2	114	184.3
CsI	626	1,280
RbI	647	1,304

There is a high probability that these iodized aerosols will end up contained as aqueous FPs in the water present in the heat transport system or containment. The aerosols may:

(1) Settle in the water present on the reactor building floor or in the sumps;
(2) Deposit on the cold walls of containment and wash away in the run-off water of condensed steam;
(3) Be trapped directly by droplets from water sprays; or, if they are hygroscopic (as in the case of CsI);
(4) Become active sites for condensation of steam.

Specific iodine compounds, CsI for instance, are also greatly soluble and can dissociate in the aqueous phase and release ions of iodide (I^-). These ions can react in a variety of ways within the aqueous phase, and can form volatile iodine compounds, such as I_2 and CH_3I, which may return to the gaseous phase. Whether in aqueous or gaseous phase, the iodine may react through interactions with paint on the containment walls. These reactions are called "heterogeneous", in opposition to "homogeneous" reactions that occur in the aqueous or gaseous phases.

Under accident conditions, the processes and reactions governing iodine behaviour in the containment can then be separated in five categories:

(1) Homogeneous reactions in the aqueous phase;
(2) Heterogeneous reactions in the aqueous phase;
(3) Mass transfer between phases related to a liquid/vapour equilibrium (e.g., volatile compounds moving from the aqueous phase to the gaseous phase);
(4) Homogeneous reactions in the gaseous phase (not currently included in MAAP and MAAP-CANDU);
(5) Heterogeneous reactions in the gaseous phase.

The iodine chemistry model used in MAAP and MAAP-CANDU is empirical and does not include descriptions of all possible chemical reactions, involving all the intermediate steps and compounds. Rather the model seeks to simulate the global reactions, where only the most stable and significant iodized compounds are involved. This modelling approach will have to be rectified, when more knowledge is gained on the processes related to the iodine chemistry in nuclear reactor systems under severe accident conditions.

While there were no large changes to the iodine modelling from MAAP4-CANDU to MAAP5-CANDU, this recent assessment of the models (and others) has contributed to a better understanding of the treatment of this phenomenon in the code.

III.3. FISSION PRODUCT GROUPINGS

There has been some refinement of the FP modelling in MAAP5-CANDU, focused on the available groups for tracking the FPs. MAAP-CANDU models FP groups (FPGs) [190], where FPs with similar release rates from solid fuel and similar post-release chemistry and transport behaviour are grouped together. MAAP-CANDU organizes individual FPs into different groups based on their chemical species combining the masses of the component elements. These groups also fall into a volatile, semi-volatile, or low-volatile classification. The remainder of FPs are assumed to remain in the UO_2. FPs can be in various physical states (phases: V=vapour; A=aerosol; D=deposited).

There are currently fourteen FPGs in MAAP5-CANDU including two new FPGs that were not previously available in MAAP4-CANDU. These new FPGs for elemental iodine and organic iodine (numbers thirteen and fourteen, respectively). The FPG in MAAP5-CANDU is presented in Table III.2.

TABLE III.2. FISSION PRODUCT GROUPINGS IN MAAP5-CANDU

Fission Product Group #	**Physical State (Phase) V – vapour, A – aerosol D - deposited**	**FP Group Contents**
FPG 1	V	Nobles (Xe + Kr)
FPG 1	A	All non-radioactive inert aerosols (during molten core concrete interaction)
FPG 2	V&A&D	CsI + RbI
FPG 3	V&A&D	TeO_2
FPG 4	V&A&D	SrO
FPG 5	V&A&D	MoO_2 +RuO_2+TcO_2
FPG 6	V&A&D	CsOH + RbOH
FPG 7	V&A&D	BaO
FPG 8	V&A&D	La_2O_3 + Pr_2O_3 + Nd_2O_3 + Sm_2O_2 + Y_2O_3 + ZrO_2 + NbO_2
FPG 9	V&A&D	CeO_2 +NpO_2+PuO_2
FPG 10	V&A&D	Sb
FPG 11	V&A&D	Te_2
FPG 12	V&A&D	UO_2
FPG 13		Reserved for future use
FPG 14	V&A&D	Elemental iodine
FPG 15	V&A&D	Organic iodine

III.4. CHEMICAL REACTIONS INVOLVING IODINE

Table III.3 lists the set of chemical reactions included in the iodine chemistry model implemented in MAAP5-CANDU [190-192].

TABLE III.3. CHEMICAL REACTION IN THE MAAP5 AND MAAP5-CANDU IODINE MODELS

Homogeneous Reactions in Aqueous Phase	
I_2 hydrolysis	$I_2 + H_2O \leftrightarrow I^- + HOI + H^+$
HOI disproportionation	$3HOI \leftrightarrow IO_3^- + 2I^- + 3H^+$
Oxidation of I^- by dissolved O_2	$2I^- + 1/2\ O_2 + 2H^+ \leftrightarrow I_2 + H_2O$
I^- radiolysis	$2I^- + h\nu^* \leftrightarrow I_2 + 2e^-$
IO_3^- radiolysis	$2IO_3^- + h\nu^* \leftrightarrow I_2 + 2e^- + 3O_2$
CH_3I formation	$2CH_3 + I_2 \leftrightarrow 2CH_3I$
Hydrolysis of CH_3I by H_2O or OH^-	$CH_3I + H_2O \leftrightarrow I^- + CH_3OH + H^+$ $CH_3I + OH- \leftrightarrow I^- + CH_3OH$
CH_3I radiolysis	$2CH_3I + h\nu^* \leftrightarrow I_2 + 2CH_3$
Dissociation of water molecules	$H_2O \leftrightarrow H^+ + OH^-$
Heterogeneous Reactions in Aqueous Phase	
Silver iodide formation	$Ag_{aq} \rightarrow Ag_{ox,aq}$ $Ag_{ox,aq} + I^-_{aq} \rightarrow Ag_{ox}\, I_{aq}$
CH_3I formation by reaction of I_2 and I^- with painted surfaces	I_2 (+ paint) $\leftrightarrow 2CH_3I$ I^- (+ paint) $\leftrightarrow CH_3I$
Mass Transfer Between Phases	
Linked to a liquid/vapour equilibrium	I_2 (g) $\leftrightarrow I_2$ (aq) CH_3I (g) $\leftrightarrow CH_3I$ (aq)
Heterogeneous Reactions in Gaseous Phase	
I_2 absorption by painted surfaces	I_2 (g) $\rightarrow I_2$ (ad,g)
I_2 desorption from painted surfaces	I_2 (ad,g) $\rightarrow I_2$ (g)
CH_3I formation	I_2 (ad,g) + paint $\rightarrow$ 2 CH_3I
*hν is related to ionizing radiation	

Reactor safety studies including severe accident analyses are primarily interested in the concentration and chemical form of iodine in the gaseous phase, which is the phase most likely to be released to the environment. From this point of view, the important reactions are as follows:

- Radiolysis of I- in the aqueous phase, causing its transformation into a volatile compound, I2, which transfers into the gaseous phase; the strength of this reaction increases as the pH of the water decreases (i.e., becomes more acidic);
- Formation of organic iodide CH3I (highly volatile) from interaction with paint compounds, mainly occurring in the aqueous phase;
- Exchange of volatile iodized compounds from the aqueous to the gaseous phase;
- Adsorption/desorption of different forms of iodine in the gaseous phase to and from the walls;
- Formation of silver iodide in the aqueous phase; this is an insoluble compound which can be an efficient iodine trap if its stability under radiation can be demonstrated[8].

[8] This reaction is not a major factor in CANDU severe accident scenarios as silver is not present.

MAAP-CANDU does not include a model of the radiolysis reactions in the gaseous phase. It is assumed that in the gaseous phase, the water molecule concentration is insufficient for the effects of radiolysis to be significant.

For the water and gas temperature ranges found in containment under typical severe accident conditions, the chemical kinetics of the reactions described in Table III.3 will determine the eventual distribution of chemical species. Chemical kinetic treatment and equation solutions as applied in MAAP are available in the MAAP User Manuals.

Table III.4 lists the chemical species taken into account in the model. There are eight chemical species with iodine in three different phases and CH_3I in two different phases. These chemical species and their phases are indexed in the code as shown in Table III.5.

TABLE III.4 CHEMICAL SPECIES IN IODINE MODEL

Chemical Species	Aqueous Phase		Gaseous Phase	
	In Suspension	Adsorbed	In Suspension	Adsorbed
I_2	X		X	X
I^-	X			
IO_3^-	X			
CH_3I	X		X	
CH_3	X			
HOI	X			
AgI	X			
Ag	X			

Note: H^+ concentration appears in the rate equations of these species; its concentration is determined by the pH model.

TABLE III.5 INDEX NO. CHEMICAL SPECIES

Index No	Chemical Species	Comment
1	I_2	in aqueous phase
2	I^-	in aqueous phase
3	IO^-_3	in aqueous phase
4	CH_3I	in aqueous phase
5	CH3	in aqueous phase
6	HOI	in aqueous phase
7	Ag	suspended particles in aqueous solution
8	AgI	in aqueous phase
9	AgO_x	in aqueous phase
10	I_2	in gaseous phase
11	CH_3I	in gaseous phase
12	I_2	adsorbed on walls

The transport of these 12 iodine chemistry-related species (see Table III.5) between containment compartments is modelled in MAAP-CANDU.

III.5. ANALYSIS ASSUMPTIONS USED IN IODINE CHEMISTRY MODELS

The following assumptions are included in the iodine chemistry model:

(a) The chemical reactions can take place in any compartment where there is a water pool;
(b) Three phases are considered in a compartment, each phase is assumed to be fully homogeneous i) within the gas, ii) in the liquid phase, and iii) on the walls exposed to air;
(c) The dose rate applied is the one existing in the aqueous phase, where the radiolysis reactions take place;
(d) The pH of the aqueous phase is calculated and provided as an input to the iodine chemistry model;
(e) No chemical reactions can occur before CsI enters the aqueous phase. The I-aq source rate is estimated from the following CsI mass flow rates:
   - From gaseous to aqueous phase (aerosol fall into water);
   - Between submerged wall and aqueous phase (dissolution of deposits);
   - In aqueous phase from primary or secondary system (if any);
   - In aqueous phase from corium debris (if any);
   - In aqueous phase from condensate due to fan coolers (if any).
(d) Agaq (Ag in aqueous phase) source rate is estimated to be the same as the I-aq source rate;
(e) CH3aq (CH3 in aqueous phase) source rate depends on flowrates of water from the reactor primary or secondary system, or containment spray system.

III.6. BRIEF DESCRIPTION OF THE EQUATIONS SOLVED WITHIN THE IODINE CHEMISTRY MODELS

At the temperatures found in the aqueous and gaseous phases in the containment under severe accident conditions, thermodynamic equilibria take a long time to be reached. Thus, the important equations to be solved are the chemical kinetics equations. The treatment of the chemical kinetics in the iodine chemistry model is focused on tracking the amount of the chemical compounds based on the formation and dissolution of the compound due to various chemical reactions.

If $nA_{i\phi}$ is the number of moles of the chemical compound Ai in phase ϕ (aqueous or gaseous) at time t, the variation in the number of moles as a function of time is expressed as:

$$\frac{dnAi_\phi}{dt} = \left(\frac{dnAi_\phi}{dt}\right)_s + \sum_{r(Ai_\phi)} \left(\frac{dnAi_\phi}{dt}\right)_r \tag{III.1}$$

where:

$\left(\frac{dnAi_\phi}{dt}\right)_s$	=	the variation of $nA_{i\phi}$ linked to a source (addition of compound A_i to the system);
$r\,(Ai_\phi)$	=	all the reactions in which the compound A_i is involved in phase ϕ (formation or disappearance);

$\left(\frac{dnAi_{\phi}}{dt}\right)_r$ = variation of $nA_{i\phi}$ linked to reaction r, the form of which depends on the reaction

The following reaction types are considered:

- Homogeneous reaction in phase ϕ;
- Adsorption reaction of Ai on wall in phase ϕ;
- Desorption reaction of Ai from wall in phase ϕ;
- Transfer between aqueous and gaseous phase linked to a liquid/vapor equilibrium.

The following general comments apply to these equations:

(a) At constant pressure, the reaction rate constants vary with temperature in accordance with Arrhenius' law:

$$k(T) = K\exp\left(\frac{-Ea}{RT}\right) \quad \text{(III.2)}$$

Where:

k(T) is the reaction rate constant, K is the pre-exponential factor, a constant for each chemical reaction, Ea is the activation energy of the reaction, R is the ideal gas constant, and T is the absolute temperature.

(b) The orders of reactions relating to various compounds may be whole, fractional or null, and are determined by experiment;

(c) The rate of a reaction may depend on other parameters such as the pH or, in the case of radiolysis, the dose rate in the phase in question.

III.7. MAAP5-CANDU INPUT PARAMETERS RELATED TO IODINE CHEMISTRY MODEL

The most important input parameters related to the iodine chemistry model are listed in Table III.6.

TABLE III.6 MAAP5-CANDU INPUT PARAMENTER RELATED TO THE IODINE CHEMISTTRY MODEL

Input Parameter	Description
IIODIN	Flag; it is used to activate the iodine chemistry model (=1); default value = 0
FELEI	Dimensionless number; the fraction of initial inventory (at the accident initiation time, i.e., at the t=0 s) of iodine that is in the form of molecular (or elemental) iodine, I_2 (FP group #14), see Table III.2.
FORGI	Dimensionless number; the fraction of initial inventory of iodine that is in the form of methyl iodide (or CH_3I, or organic iodine, FP group #15), see Table III.2.
FCSI	The mole fraction of caesium iodine in FP group 2 (CsI+RbI), see Table III.2.

III.8. MAAP5-CANDU OUTPUT PARAMETERS RELATED TO THE IODINE CHEMISTRY MODEL

The most important output parameters related to iodine chemistry model are listed in Table III.7. Note that “node IV” in the table is related to containment (reactor building) control volume (or computational volume) number. The control volume can represent a single containment compartment, or a combination of several containment compartments. MAAP and MAAP-CANDU use lumped parameter containment model, which is described in reference [190].

TABLE III.7 MAAP-CANDU OUTPUT PARAMETERS RELATED TO IODINE CHEMISTRY MODEL

Output Parameter	**Description**
MICHRB(i,IV)	Mass (kg) of chemical species i (see TABLE III.5) in node IV
NICHRB(i,IV)	Molar amount (gram mole) of chemical species i in containment node IV
CICHRB(i,IV)	Concentration (gram mole/litre) of chemical species i in node IV
NITOTW(IV)	Total node IV moles (gram mole) of aqueous-phase iodine (I) from species 1 (I_2), 2 (I^-), 3 (IO_3^-), 4 (CH_3I), 6 (HOI) and 8 (AgI)
NITOTG(IV)	Total node IV moles (gram mole) of gas-phase iodine (I) from species 10 (I_2), 11 (CH_3I) and 12 (I_2 adsorbed on walls)
NITOT(IV)	Total node IV moles (gram mole) of gas and aqueous-phase iodine (I) (sum of NITOTW(IV) and NITOTG(IV))
CI2WM(IV)	Node IV concentration (gram mole/(litre water)) of aqueous-phase iodine (I) from species 1 (I_2, aqueous)
CI2GM(IV)	Node IV concentration (gram mole/(litre gas)) of gas-phase iodine (I) from species 10 (I_2, gas-phase)
CITOTW(IV)	Total node IV concentration (gram mole/(litre water)) of aqueous-phase species 1 (I_2), 2 (I^-), 3 (IO_3^-), 4 (CH_3I), 6 (HOI) and 8 (AgI)
CITOTG(IV)	Total node IV concentration (gram mole/(litre gas)) of gas-phase species 10 (I_2) and 11 (CH_3I)

APPENDIX IV - SHORT OVERVIEW OF THE COMMONLY ADOPTED FLUID DYNAMICS MODELS FOR HYDROGEN DISTRIBUTION AND HYDROGEN MITIGATION SYSTEMS

This Appendix provides a short overview of the commonly adopted fluid dynamics models for hydrogen distribution and mitigation systems. More details can be sought in Reference [167].

IV.1 TURBULENCE MODELS

The turbulence in hydrogen mixing and distribution introduces complexity to the flow. The presence of turbulence complicates the solution of the governing equations for fluid flow. Thus, it is crucial to carefully address each term in these equations and employ suitable approximations.

Numerous turbulence models are available in the literature; however, there is no universally acknowledged superior model applicable to all problem classes. The selection of turbulence models relies on factors like the physics inherent in the flow, established practices for specific problem classes, the desired level of accuracy, available computational resources, and the time allotted for simulation.

The validation exercises carried out by various researchers offer some guidance on selection of a turbulence model. The turbulence models such as k-ε, k-ω, low Reynolds number k-ε and Shear Stress Transport Turbulence are generally used in the simulations related to containment thermal hydraulics and hydrogen distribution studies [167].

IV.2 TURBULENCE MODELS

The selection of the best suited condensation model is important since condensation controls the presence of steam in the containment in the medium and long term which affects the mixing process and the gas mixture compositions.

Numerous researchers have endeavoured to tackle condensation in the presence of non-condensable gasses. An extremely refined computational grid would be essential near the condensation surface to accurately model this process using mechanistic approaches to heat and mass transfer, resulting in prolonged computational times. In order to diminish computational time, researchers have used different correlations for condensation [193-199].

Figure IV.1 illustrates the process of film condensation, depicting the occurrence of condensation at the interface between the gas and the liquid. Steam and non-condensable gasses gradients emerge across the boundary layer of gas [200-201]. Martin-Valdepenas et al. formulated a model that incorporates liquid film resistance [202-203]. The Nusselt condensation model, with correction factors accounting for surface waviness, are used to determine heat transfer coefficient within the liquid film [204]. An alternative model, based on diffusion and formulations from Chilton and Bird, involves solving governing equations through a CFD package. This model simulates wall condensation by applying appropriate boundary conditions. This approach is anticipated to yield improved results, given its theoretical generality, applicability across various conditions, and ability to encompass all factors influencing condensation in the presence of non-condensable gases. Consideration of the film may vary in this approach [167].

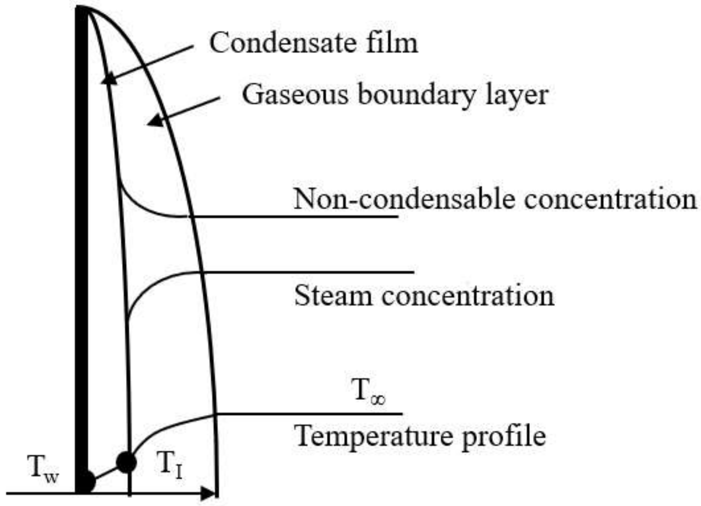


FIG. IV.1 Film condensation with non-condensable on a wall.

IV.3 SUMP EVAPORATION

The sump is a containment feature and large amount of water. Other containment features such as suppression pool, CV, calandria vault are similar to a sump. Under accident conditions, the sump also collects water condensed on walls and from the containment spray system. Sump evaporation and steam condensation on sump surface may affect the steam concentration and gas mixture concentrations in the containment. Hence, sump needs to be modelled for containment hydrogen distribution calculations.

Both sump water evaporation and steam condensation models can be implemented through mass and energy balances using different approaches. Dedicated experimental programme had been carried out in the TOSQAN facility. The obtained results allowed the validation of the heat and mass transfer models [205-206].

IV.4 BULK CONDENSATION

Fog formation within enclosed systems occurs when the pressure of the vapor surpasses the atmosphere saturation pressure. Under these circumstances, steam condenses, giving rise to tiny droplets in the absence of a cold surface. The steam then condenses onto these droplets, causing them to grow. Various models exist to model these phenomena, some requiring experimental constants that are often proprietary. Alternatively, a model based on classical theory of droplet formation for bulk condensation can be implemented in CFD codes.

A bulk condensation phenomenon was modelled based on the classical theory of nucleation. The CFD model for bulk condensation was incorporated into the FLUENT code using user-defined functions for mass, momentum, energy, steam, turbulent kinetic energy, and dissipation rates. A 3D channel geometry resembling the CONAN test facility was chosen for the study. A parametric investigation involving various inlet mass fractions was conducted to assess the efficacy of bulk condensation. Simulations were executed with only wall condensation and also for both bulk and wall condensation. The results indicate that, under the specified conditions, bulk condensation has minimal influence. The model can be applied using user-defined functions for analysing the distribution of hydrogen within a containment system. In the initial phase of blowdown during a large break LOCA scenario, bulk condensation may influence the concentrations of gases within the containment. Integrating this model would therefore be beneficial for conducting simulations of containment hydrogen distribution.

IV.5 SPRAY MODELLING

Most commercial CFD codes deal with multiphase flows. However, the modelling of mass and heat exchange between the gas and the water droplets is generally absent and need to be implemented by specific user functions. For example, the condensation of vapour on the surfaces of water droplets and the evaporation rate of water droplets can be expressed as follows :

$$\dot{m}_{droplet} = k\left(\rho_{vap} - \rho_{sat,droplet}\right)A_{droplet} \tag{IV.1}$$

where ρ_{vap} stands for steam density, $\rho_{sat,droplet}$ steam density at saturation condition, $A_{droplet}$ droplet surface. k is the mass transfer coefficient defined as:

$$\left(k = \frac{S_h \times D}{D_{droplet}}\right) \tag{IV.2}$$

Where $D_{droplet}$ represents the droplet diameter, D the diffusion coefficient and the Sherwood number S_h expressed as function of Reynolds and Schmidt numbers [207].

The gas and water droplet heat exchanges are modelled by considering both condensation (P_{cond}) and convection (P_{conv}):

$$P_{cond} = \dot{m}_{droplet} \times h_{lat} \tag{IV.3}$$

$$P_{conv} = \alpha A_{droplet}\left(T_{gas} - T_{droplet}\right) \tag{IV.4}$$

where α is function of Nusselt number [207]. $T_{droplet}$ and T_{gas} stand respectively to water droplet and gas temperature. H_{lat} represents the latent heat.

IV.6 RECOMBINERS

In most CFD codes, modelling of PAR adopts a lumped parameter approach. It assumes that the flow is steady and the recombiner vessel does not exchange heat with the surrounding gas mixture. The system that characterizes the gas flow within the recombiner incorporates the momentum equation, mass balances for each species, and energy balances for both the gas and the recombiner plates.

The recombination rate can usually be implemented as correlation provided by the manufacturer.

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ANNEX – SUBMITTED TECHNICAL MEETING ABSTRACTS

TECHNICAL MEETING SESSION 1 - UPDATES IN LICENSING REQUIREMENTS AND THEIR IMPLEMENTATION BY THE INDUSTRY

Updates on Regulatory Requirements for Severe Accident Analysis and Practical Elimination of Events for Design of NPPs in India

Dr R. B. Solanki, U.K. Paul and A. Gaikwad (AERB, India)

The Atomic Energy Regulatory Board (AERB) is national regulatory body in India mandated with formulating national regulatory and safety requirements for nuclear and radiation facilities. AERB has well-established systems and process for development of regulatory documents which takes into account of the requirements of relevant IAEA documents, in addition to national legal and regulatory framework. The regulatory requirements are also updated taking into account the regulatory experience as well as feedback from the operating experience, new research & development findings and the current best practices.

The safety reviews performed after the Fukushima Daiichi accident have brought out the inherent strengths found in the PHWR design, regulatory and operational practices and requirements. Further strengthening measures identified are being implemented for the Indian NPPs in a phased manner. These safety enhancements were classified as short term, medium term and long-term measures, taking in to account the feasibility for implementation, need for assessments/analysis/development, engineering & procurement and scheduling of the planned outages for implementation.

A few of these requirements were already introduced in Safety for Design of Light Water Reactor Based NPPs in 2015. These requirements are being updated as a part of the revision of the existing requirements for the design of PHWR based NPPs, wherein additional regulatory requirements are being introduced taking into account of IAEA-SSR-2/1, Rev. 1 (2016). These requirements are associated with increasing plants resilience to cope with extreme external events that exceed the design bases and to improve the provisions for severe accidents mitigation. These includes redefining the ‘plant states’ by introducing Design Extension Conditions and concept of practical elimination of events. The acceptance criteria and safety analysis methods, analysis rules are also being revisited in associated safety guides. The regulatory requirements for ensuring severe accident response capabilities in the design of NPPs is also being introduced with strengthening of DiD provisions by additional safety systems and complementary safety features for handling design extension conditions including severe accidents. Establishing On-Site Emergency Support Centres (OESCs) having the capability to remain functional under radiological conditions following a severe accident and needs to be capable of withstanding extreme external events is also being introduced.

The utilities have developed SAMG for different NPP designs based on the technical bases that have been reviewed by AERB. The SAMG that were developed based on those bases have been implemented at all the operating NPPs. This includes the implementation of hardware enhancements, mock up tests, updates in the training of the operating personnel, and periodic surveillance.

This paper aims to share Indian experience on updating regulatory requirements for severe accident analysis, probabilistic safety analysis and practical elimination of events for nuclear power plants. The paper also brings out certain considerations for establishing regulatory requirements for novel advanced reactors including Small and Modular Reactors (SMRs), which will be deployed in India in future.

Regulatory and Licensing Requirements on Severe Accident Simulation and Modelling and on Applications for SAMG Development and Emergency Planning and Preparedness

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Abstract

Using the lessons learned from the Fukushima Daiichi accident, CNCAN (National Commission for Nuclear Activities Control), the nuclear regulatory authority of Romania, has improved the regulatory framework governing the severe accident simulation and modelling and the associated applications. New regulations and regulatory guides have been issued to cover these aspects and the severe accident analyses have become part of the licensing basis. The paper presents the main developments of the regulatory and licensing framework for severe accident simulation and modelling and for the related applications.

Introduction

After the Fukushima Daiichi accident, taking into account the lessons learned from this event, CNCAN (National Commission for Nuclear Activities Control), the nuclear regulatory authority of Romania, has improved the regulatory framework governing the severe accident simulation and modelling and the associated applications.

The following regulations contain the main requirements relevant for this domain:

- NSN-21 (rev. 1) – Fundamental Nuclear Safety Regulations for Nuclear Installations (2020);
- NSN-22 – Regulation on the licensing of the nuclear installations (2019);
- NSN-24 – Regulation on the deterministic nuclear safety analyses for nuclear installations (2019);
- NSN-07 (rev.1) – Nuclear safety requirements on the response to transients, accident management and on-site emergency preparedness and response for NPPs (first issued in 2014, revised and updated in 2020);
- NSN-23 (rev. 1) – Nuclear safety regulation on the training, qualification and authorization of the personnel of organizations operating nuclear installations (first issued in 2017, revised and updated in 2021).

The following regulatory guides are applicable to this domain:

- GSN-03 - Guide on fulfilling the overall nuclear safety objective set in the fundamental nuclear safety requirements for nuclear installations (2018);
- GSN-04 – Guide on the format and content of the Final Safety Analysis Report for nuclear power plants (2015);

Compliance with the provisions of the above-mentioned regulations is mandatory in the licensing process and for the entire duration of the validity of the NPP license.

Regulatory requirements on severe accident analyses

The regulation NSN-24 contains requirements on the deterministic safety analyses, including severe accident analysis. These requirements include provisions on the numerical and physical models used in the analyses and on the validity, applicability and limitations of the computer codes, taking account of the experimental basis, operating experience and recommendations from the designer of the nuclear installation.

CNCAN has quality management regulations for computer codes for safety analyses since 2003. These are set in the regulation NMC-12 (Specific regulatory requirements on the management systems applied to the production and utilization of computer software for research, design, analyses and calculations dedicated to nuclear installations). These regulations are applicable to the development and utilization of computer codes for all types of deterministic and probabilistic safety analyses. After the severe accident analyses have become part of the licensing basis, CNCAN is verifying compliance with NMC-12 also for the computer codes used for severe accident analysis.

Cernavoda NPP uses an integrated severe accident analysis code, Modular Accident Analysis Program, version 4, for CANDU NPPs (MAAP4-CANDU), for analysing severe core damage accident progression and the resulting release of fission products from the fuel, core debris, and eventually from containment, for representative accident sequences. MAAP4-CANDU is the industry standard toolset for CANDU Level 2 PSA severe accident progression analysis. The accident scenarios are selected based on their contribution to the severe core damage frequency. Examples of representative scenarios include:

- Station Blackout (SBO);
- Small LOCA;
- Feeder Stagnation Break (FSB);
- Steam Generator Tube Rupture (SGTR);
- Multiple Steam Generator Tube Rupture (MSGTR).

Regulatory requirements on the applications of severe accidents analyses

As regards the applications of severe accident simulation and modelling, CNCAN has issued specific requirements for the development, verification and validation of severe accident management guidelines (SAMGs), as part of the NSN-07 regulation. In addition, in accordance with the provisions of the NSN-24 regulation, severe accident analyses are required to be used in the development of emergency preparedness plans and procedures and in the associate training.

Another application of severe accident simulation and modelling is related to the capabilities of the full-scope simulators for NPPs, used in the training for control room operators and shift supervisors, addressed by CNCAN as part of the NSN-23 regulation. In accordance with the NSN-23 regulation, the full-scope simulator needs to have the capability, to the extent practicable, to reproduce severe accident scenarios. In order to comply with this requirement, the full-scope simulator for Cernavoda NPP has been recently modernized and upgraded. While severe accident scenarios are not yet part of the practical part of the licensing examinations, CNCAN is considering this option for the future. Currently, the full-scope simulator's extended capabilities are used for supporting operators' training in severe accident management and for making the emergency response exercises more realistic.

Regulatory reviews and inspections

CNCAN has also developed internal procedures for review and assessment of the severe accident analyses, as well as procedures for the inspection of their applications.

Review and assessment of severe accident analysis is performed as part of the process for license renewal for the operating reactors and also when new or revised analyses are submitted to CNCAN. For new reactors, review and assessment of severe accident analysis will be part of the licensing process from the beginning.

Periodic inspections are performed to verify the following:

— The maintenance, updating and revalidation of severe accident management guidelines to account for design modifications, new analyses, new relevant results of research activities, recommendations from the nuclear industry and regulatory from other countries;
— The use of the computer codes for severe accident analysis and the continuous development of licensee's capabilities in this area;
— The performance of the full-scope simulator, including in simulating severe accident scenarios and the use of these capabilities in operators' training and in emergency preparedness and response training and exercises.

Conclusions

In the last decade, CNCAN has made significant progress in developing and consolidating the regulatory and licensing framework for severe accident simulation and modelling and for the related applications. The challenges we still face relate to developing and maintaining our regulatory capabilities for performing technical reviews of the severe accident analyses and their applications.

Canadian Regulatory Perspective on PHWR Severe Accident Management, Simulation and Modelling
S. Gyepi-Garbrah (CNSC, Canada)

In Canada, Severe Accident Management (SAM) programme for Nuclear Power Plants (NPPs) is enshrined in Canadian Nuclear Safety Commission (CNSC) regulatory requirements. The objective of the programme is to prevent accident progression from becoming a severe accident and terminate its progression as early as possible. If a severe accident to occur at an NPP, the SAM programme enforces several mitigating strategies to preserve the integrity of the fission products barriers and the spent fuel pool, minimize the radioactive materials releases to environment, and achieve long-term safe stable conditions in the reactor core and spent fuel pool. These SAM mitigating strategies, together with their technical basis and enabling instructions, are developed and documented in Severe Accident Management Guidelines (SAMGs) that are station specific and considered part of the SAM programme.

The CNSC assures effective SAM programme at Canadian NPPs through implementing the expectations of regulatory document REGDOC-2.3.2 which is enforced as part of each plant's licence. This presentation will describe SAMG mitigating strategies that are applicable to Canadian NPPs and assessed based on CNSC's regulatory approach. In addition, CNSC regulatory perspective related to SAMG verification and validation will also be discussed.

These activities include reviewing the technical basis of the MAAP-CANDU severe-accident computer code analytical simulations performed to demonstrate the effectiveness of the mitigating strategies and evaluation of SAMG mitigating strategies. This approach is seen as providing valuable evaluation and feedback to operating Pressurized Heavy Water Reactor (PHWRs) to improve and support decision making for increasing confidence in SAM effectiveness.

The results of collaborative research with Canadian National Laboratories (CNL) will also be presented, which addresses the knowledge gaps on H2-CO combustion and recombination and provides data for model development under quiescent and turbulent conditions. The presentation will also highlight SAM activities related to CNSC Emergency Operations Centre (EOC) fast running codes and tools improvement.

To obtain a copy of the abstract's document, please contact us at cnsc.info.ccsn@canada.ca or call 613-995-5894 or 1-800-668-5284 (in Canada). When contacting us, please provide the title and date of the abstract.

Severe Accident Analysis in Canada and its Application

D. Mullin (NB Power, Canada)

Canadian regulatory requirements and standards identify the need to evaluate the ability of nuclear power plant (NPP) design to withstand challenges posed by a beyond design basis accident (BDBA), including a more severe subset of accidents leading to highly unlikely core deformation, core disassembly and the potential radiological consequences. While postulated accident sequences from a probabilistic safety assessment (PSA) can be utilized to identify those that dominate risk metrics and would warrant further analysis in the severe accident domain, it is generally viewed that the outcomes of such severe accident analysis should be utilized to (a) enhance plant design to improve prevention and mitigation measures; (b) assist in emergency response and evaluated exercise planning; (c) verify plant habitability to ensure NPP operators can perform effective mitigating actions; (d) evaluate the likelihood of instrument and equipment survivability under severe conditions; and, (e) evaluate the effectiveness of off-site emergency plans.

The presentation and discussion provide additional detail on how analytical outcomes for postulated severe accidents both from deterministic and probabilistic analyses has been utilized at one Canadian NPP to drive improvements. This includes the selected simulation scenarios applied to determine adequate emergency planning zone sizing in consideration of the need for sheltering, thyroid blocking, evacuation and relocation. Habitability assessments utilizing the source terms from representative and limiting postulated severe accident cases, and the evaluation of emergency mitigating equipment design using science-based external hazard assessments, are described. The key aspects of event progression are also discussed in terms of establishing critical performance objectives for the deployment timing of portable emergency mitigating equipment, which can be evaluated during drills and exercises. Finally, the presentation also discusses how postulated severe accident source terms can be utilized assessing deterministic and stochastic health risk metrics to verify effectiveness of off-site emergency response plans.

Severe Accident Simulation and Modelling at Cernavoda Full-Scope Simulator
B. Tutuianu, (SNN, Romania)

The Fukushima-Daiichi accident has driven the utilities to improve operator's training on severe accidents management. The theoretical understanding of the physical phenomenon taking place during a severe accident and also exercising the practical actions and procedures to mitigate the consequences of such an event are both of a great value when talking about operator's training.

In 2021, Cernavoda NPP completed a Full-Scope Simulator upgrade project with L3MAPPS Canada which included MAAP5-CANDU Severe Accident Code integration into the simulator software configuration.

MAAP5-CANDU is intensively used as an engineering tool for Probabilistic Risk Analysis applications and Severe Accident Analysis and this expertise have to be applied in personnel training. The solution was to keep the classic modelling code for design basis accidents and use MAAP5 CANDU code just for severe accidents and just for the nuclear systems like reactor, NSSS thermalhydraulics and containment. The transition from classic code to MAAP5 is automatically performed when certain parameters are met. The paper describes how a smooth transition is ensured and how this smooth transition was customized to work for Cernavoda simulator. Customizations had to be performed also for site specific accident mitigation plans described in the emergency response procedures.

A Severe Accident evolves over many hours or even days and the operating crew or emergency response team have to practice all the procedures to apply during such an accident. Since the training session is planned over a few hours, the instructor has to prepare the training scenario, using the severe accident code in such a way to condense the entire accident evolution in the planned training time. We will present the tools and strategies used by the instructor to achieve this goal.

Together with the MAAP5-CANDU code, visualization tools were also introduced to help the trainees understand the phenomena which occurs during a severe accident. These tools and the way they are used at Cernavoda for different training purposes will be presented in the paper.

Simulator response at severe accident scenarios was validated by comparing the results obtained in the full-scope simulator versus the results obtained by the stand-alone version of MAAP5-CANDU. Running several LOCA scenarios and Station Blackout and comparing the evolution shows really close behaviour in both integrated and standalone modes.

Introducing the severe accident capabilities to the full-scope training simulator offers a great tool to train plant operators and emergency response teams in an environment as close as possible to the real life. Emergency response procedures as SAGs, SAMGs can now be tested and practiced in a fully integrated fashion. The simulator-driven visualization tools offer the means for the personnel to visualize and to understand the processes taking place in the plant.

TECHNICAL MEETING SESSION II - CODE DEVELOPMENTS AND VALIDATION FOR PHWR SEVERE ACCIDENTS

Issues in PHWR Severe Accidents and Relevant Recent International Studies

M. Krause, (INUEC, Canada)

Introduction

Severe accident analysis is much newer than the nuclear industry itself. While all early reactors, including PHWRs, were designed to various degrees to avoid severe accidents altogether and/or protected against various potential runaway scenarios by their inherent properties, by engineered systems, and/or strict procedures, an earnest effort to analyse in detail the potential progression of a severe accident scenario did not start until they had happened. The major severe accidents were TMI (1979), Chernobyl (1986) and Fukushima (2011). While early attempts at modelling of severe accident phenomena, and to some degree, integrated SA analysis were underway, these real events refocussed the effort in industry, academia, and regulatory regimes. For example, TMI led to huge efforts in hydrogen phenomena, Chernobyl to FP release and dispersion, and Fukushima to long-term SBO initiated progression and unmitigated scenarios.

More recently, the by-now very sophisticated computer models and codes are more widely used to better quantify and reduce risk in support of Probabilistic Safety Assessment (PSA) Level 2 and 3 consequence analyses, to support SAMG technical basis, and as fast-running "simulators" for operator training and as a tool for the technical support center during real or simulated events. Really, these goals are ultimately their main purpose.

From a purely computational effort perspective, PSA support at present represents probably the largest use of SA codes. PSA is a framework for systematic identification and analysis of failures leading to public risk (core damage, large releases, and dose estimate), where typically several representative (enveloping) event sequences are analysed using severe accident codes that incorporate specific plant design features and procedures, postulated safety system failures and human performance errors. PSA can be performed on a per-unit basis (traditional) or a multi-unit site basis ("MUPSA"). These assessments involve many computations to cover the various dominating scenarios and are the foundation for risk quantification (probability x consequence) and are part of the Technical Basis for Severe Accident Management Guidance (SAMG), through event progression and timing of events and failures.

PHWR Specific Needs – Overview

Many severe accident phenomena are similar for various water-cooled reactor designs. However, there are several basic design differences that require special consideration, R&D, and modelling capabilities in horizontal-channel PHWR designs owing to some basic phenomenological differences and potential design vulnerabilities. The main differences arise from the entirely different layout and material composition of the reactor core, affecting the behaviour, phenomenology and main event timing (accident progression timeline) from primary system thermohydraulics, fuel degradation, corium formation and composition, to interactions with, and failure of, the core vessel.

Another difference between PHWR and LWR severe accident treatment in the "limited core damage" regime (now called DEC-A) comes from the historic licensing regime, which has always required to consider LOCA/LOECI as a DBA for PHWRs. Therefore, many SA phenomena have been studied and modelled more extensively than for LWRs, such as hydrogen behaviour, radio-iodine chemistry, and fuel-coolant interaction. However, due to the much larger LWR community, other phenomena are less well studied, such as fuel assembly (bundle) degradation and core collapse, corium properties and behaviour, and IVMR.

For the latter behaviour, IVMA, the CV is considered the "vessel" and is key to halting a severe accident from progressing to the ex-vessel phase. Therefore, it is the current (and recent) focus of PHWR SA research and modelling efforts around the world. Briefly, the CV, which is "pre-flooded" by design, behaviour is driven by a relatively slow boil-off and gradual core "disassembly" with lower power density in first, a suspended and then, a terminal debris bed inside the CV. Main factors that influence whether a progressing to an ex-vessel phase is possible or probable, depend on accurate characterization of the corium behaviour (significantly different composition than in a LWR) and the heat transfer from the corium to the vault water that surrounds the CV. This behaviour and the responsible phenomena are major subjects of study and code development/validation for PHWR SA.

Important initiating events, other than SBO, potentially leading to a SA are LOCAs of all sizes. Here, there is also a significant difference in the thermalhydraulic behaviour of horizontal-channel PHWRs and vertical- reactor pressure vessel LWRs. While the definition of small- and large-break LOCA is similar (not identical) for both reactor types, natural circulation behaviour in certain small-break LOCA scenarios is quite differed, and the PHWR has an additional "critical-break" LOCA category with significant periods of fuel channel flow stagnation. Small- and critical-break behaviour needs to include a detailed understanding of transitions to natural circulation, "thermosyphoning" in the primary system to establish possible progression scenarios into the SA regime. With respect to the TMI accident, and the particularities of that reactor design, the ISES Safety Case Study concluded: "The Babcock and Wilcox 900 PWR design uses 2 steam generators of the once-through type. These steam generators are long, about 28 meters, which induces a specific layout: the bottom of the steam generators is lower than the core inlets. Then the transition to natural convection cooling on the primary side can be difficult in some conditions. Furthermore, they only contain a small amount of secondary cooling water, making the installation rather sensitive during certain kinds of transient."

Relevant Recent International Studies

Brief overviews and summaries of recent IAEA-led benchmarking and activities, completed in the past 10 to 15 years, are given in this Section: coupled neutronic/thermalhydraulic numerical benchmarks (IAEA-TECDOC-1994), Candu-6 unmitigated SBO Benchmark (IAEA-TECDOC-1727), SBLOCA benchmarking against RD-14M facility (IAEA-TECDOC-1688), and the ICSP on moderator subcooling requirements (IAEA-TECDOC-1890).

(a) Numerical Benchmarks for coupled Neutronics-TH Simulations

The motivation for this cooperative project was to support the development/assessment of best estimate,
coupled multi-physics transient simulations for PHWR type reactors with a series of numerical test problems (for which no analytical solutions or real-life data exist) designed to test neutronics and thermalhydraulics feedback effects under extreme conditions. It was based on a simplified CANDU6 at equilibrium core conditions with detailed models of the primary side network only, with no Reactor Regulating System (RRS), Absorber Rods, SDS2, structural components, end shield, or the CV notch. Basic fuelling and thermalhydraulic characteristics

were defined for the participants, who were required to model local TH conditions and feedback behaviours. While the included scenarios would generally not lead to SA conditions, the positive feedback effects that made these stylized problems challenging can be compared to similar positive feedback effects between different “disciplines” in integrated SA analysis, e.g., between TH and FP release/H_2 generation.

Four manufactured problems were designed and simulated with uncoupled and coupled methods, with details of coupling strategies left to each participant. Two sub-problems are defined under each problem: (a) neutronics simulations with postulated TH conditions, and (b) TH simulations with postulated power transient. The purpose of these sub-problems is (a) separate testing of neutronics and TH methods and codes and (b) gather more information for better comparison and analysis of variability in these challenging test problems results. The processing of participants’ results was mainly focused on the decomposition of the variance in results arising from the use of different simulation codes and/or coupling methods. The IAEA-TECDOC-1994 contains detailed specification data, participant model descriptions and results.

(b) Unmitigated Severe Accident Progression Benchmark

The main objective of the IAEA CRP on a generic CANDU-6 Station Blackout (SBO) scenario was to assess the capabilities and compare the predictions from seven participants with different codes simulating the same conditions for a specified, generic CANDU 6 plant undergoing a long-term, unmitigated (i.e., failure of mitigation systems and measures and no SAM actions) SBO scenario.

Good agreement, in terms of overall behaviour and timing of key events, was achieved by most participants in early phase, but poorer agreement was noted in later phases are due to knowledge gaps and accumulation of uncertainties in core disassembly, debris/corium-vessel thermo-chemical interaction and CV integrity models. Detailed results, and suggestions for improving these through subsequent R&D and model improvements, are published IAEA-TECDOC-1727 (2013).

(c) TH for Small-Break LOCA

Following a successful integrated code/experimental comparison exercise on large-break LOCA in PHWR, a second international collaborative intercomparison for validation of computer codes for PHWR thermalhydraulic safety analyses for six participants was conducted about 10 to 15 years ago. The aim was to progress the knowledge of important phenomena expected in a small LOCA in PHWRs, including natural circulation; to evaluate the capabilities of the codes to predict those phenomena.

All the main phenomena were qualitatively captured by the participants. Discrepancies were observed, in particular in void predictions. These did however not significantly affect the overall prediction the parameters that are critical in quantifying the safety margins. Detailed results are published IAEA-TECDOC-1688 (2012).

(d) Moderator as a Heat Sink

The IAEA International Collaborative Standard Problem (ICSP) on HWR moderator subcooling requirements demonstrated the analysis capabilities of member states to calculate the backup heat sink potential of the moderator during accidents. Nine organisations from five member countries with Pressurized Heavy Water Reactor technology participated in the blind simulation and used 10 different computer codes. Blind and open simulations were compared and a heat balance method to calculate PT/CT contact conductance was developed.

The blind simulations achieved their objectives and provided significant insights into analysis codes and user effects. While the calculated pressure-tube deformation characteristics are generally consistent, although the heatup rates were slightly lower than the measured value, the blind post-contact pressure-tube temperatures and calandria tube dryout and rewet behaviour showed significant scatter, likely due to modelling of dryout/rewet behaviour.

The open post-contact pressure-tube temperature and prediction of calandria tube dryout, rewet, and strain were significantly improved. Overall, the code performance observed in the blind simulations indicate that more validation and user training is essential to reduce the scatter in predictions. Detailed results are published IAEA-TECDOC-1688 (2012).

Severe Accident Code Development for CANDU Reactor in KOREA

J.H. Bae, K.H. Kim, D.G. Son, J.Y. Kang, Y.M. Song, B.W. Rhee, Mr J.Y. Jeong, (KAERI, Republic of Korea)

Following the Fukushima accident in 2011, substantial efforts were undertaken worldwide, including South Korea, to enhance measures aimed at preventing and mitigating severe accident scenarios in nuclear power plants. In South Korea, as part of these emergency measures, new severe accident legislation was introduced, resulting in a well-defined criterion for acceptable fission product release during severe accident situations. Specifically, within the context of Probabilistic Safety Assessment, the requirement dictates that the accident frequency should not exceed 10^{-6} when the release rate of cesium-137 reaches 100 TBq.

Furthermore, operators of nuclear power plants (NPP), encompassing both Pressurized Water Reactors (PWR) and Pressurized Heavy Water Reactors (PHWR), are mandated to submit their Accident Management Plans (AMP) to the regulatory body for review.

In the case of PWRs, globally recognized severe accident codes like MAAP, MELCOR, CINEMA, and RELAP-SCDAP are available. This allows the regulatory body to scrutinize AMP reports using independent codes distinct from those utilized by the utility companies, ensuring an impartial review process.

Conversely, for PHWRs, the availability of severe accident codes is limited, with notable options being MAAP-CANDU and MAAP-ISAAC, which share similarities due to their foundation on the MAAP code. Consequently, it presents a challenge for the regulatory body to review AMP reports using separate, highly accurate severe accident codes. Additionally, accurately calculating the severe accident criteria defined within AMPs becomes challenging, primarily because relatively simplified modelling approaches are employed to simulate severe accident phenomena specific to CANDU reactors.

Given the background mentioned earlier, there has arisen a compelling need to develop a highly precise CANDU severe accident code. Consequently, the Korea Atomic Energy Research Institute (KAERI) has recently introduced an intricate CANDU severe accident code named CAISER (CANDU Advanced Integrated SEveRe code). CAISER primarily serves the purpose of replicating the progression of severe accidents within a CV.

The core module within CAISER is dedicated to simulating the core degradation phenomena occurring within the CV. This module comprises two key components: a module for fuel rod degradation and a module for fuel channel degradation. The module for fuel rod degradation emulates severe accident events within a channel, encompassing core uncovery, fuel rod heating, hydrogen generation resulting from oxidation of steam-Zirconia, fuel rod slumping, melting, relocation, and the subsequent thermal interaction of the relocated molten mass with either a PT or a CT. Meanwhile, the module for fuel channel degradation is tasked with replicating the comprehensive severe accident phenomena within the CV, which includes fuel channel sagging, the formation of debris beds due to fuel channel failures, the creation of molten pools, and ultimately, CV failure. These two modules are intricately interlinked to simultaneously model events occurring within both a fuel channel and a CV.

To validate the code's accuracy, the CAISER code was used to simulate the CS28-1 experiment and compared against the experimental data. The results demonstrated that the temperature distribution in the Fuel Enrichment Section (FES) closely matched the experimental data. However, a slight variance was observed in the temperature distribution of the pressure tube along the azimuthal direction. This difference is attributed to the CAISER code's utilization of a one-dimensional coolant temperature distribution in the flow direction. Future efforts should consider accounting for the impact of localized coolant temperature distribution on cross-sectional behaviour.

In order to model a plant accident scenario comprehensively, CAISER's core module has been integrated with the MARS code, originally developed for nuclear power plant circuit analysis. This collaborative setup enables the simulation of solid behaviour within the core using CAISER's core module, while MARS handles the thermalhydraulic aspects. Consequently, within the core section of a nuclear power plant, these two codes establish seamless information exchange at each time step.

Challenges in Modelling of PHWRs for Severe Accident Analysis

R. Srinivasa Rao, (AERB, India)

Computer codes such as MELCOR, ASTEC, MAAP, SCDAP/RELAP5, SOCRAT are being used for the severe accident analysis of boiling water reactors, pressurised water reactors including VVERs etc. The models available in these computer codes are verified and validated to a large extent through multilateral collaborations related to code development, user groups and through experimental programmes. However, these computer codes may not be directly used for the pressurised heavy water reactors (PHWRs) as the severe accident progression is quite different from the light water reactors due to its channel geometry. The main differences are related to core heat-up, contact of pressure and calandria tubes, core dis-assembly, debris size and composition.

At AERB, SCDAP/RELAP5 and other multi-physics computer codes are being used for the independent verification analysis of utility submissions in the severe accident domain. Owing to the reasons mentioned above, there are several challenges encountered in performing the severe accident analysis for PHWRs viz. core heat-up, hydrogen generation, PT-CT contact, heat transfer to moderator, moderator expulsion, PT-CT rupture, core disassembly and collapse, steam ingression, water ingression in the collapsed channels and many others. The challenges faced in the modelling and how they are addressed, will be presented during the meeting.

A highlight on Severe Accident Code Development Activity, Experimental Program for Beyond Design Basis Accident and CFD application for Severe Accidents of PHWRs

O. S. Gokhale, V. Archana, M. Dharmanshu Mittal, P. Majumdar, D. Mukhopadhyay, (BARC, India)

Note: To align with the Technical Meeting agenda, the content of the abstract was separated into tow presentations; "A Highlight on Severe Accident Code Development Activity and CFD Applications" and "Experimental Program for Beyond Design Basis Accidents".

Severe accident analysis code development activity is being pursued over last one decade. The activity is spread out over (i) development of severe accident analysis code PRABHAVINI and (ii) participation in code development activity for European Union code ASTEC, where additional modules were introduced top simulate Limited and Severe Code Damage Accidents and (iii) use of different international codes to address the severe accident phenomena separately. The PRABHAVINI v2.2 code consists of several modules to simulate system thermal hydraulics, core degradation, fission product and actinide release and transportation, containment phenomena, activity dispersion and public dose estimation. All the modules can work in stand-alone as well as in an integrated manner. Currently the code can simulate the DEC-A condition and the detailed severe accident model is in progress. The paper will highlight the above-mentioned activities, application to PHWRs and limitations of usage of international codes for PHWRs.

An experimental programme has been pursued for last one decade to cater to various needs for PHWR safety. Following are the categories on which different kind of studies were made:

(1) Experimental data generation for analysis code validation;
(2) Experiment backed model (e.g., correlation, failure criteria) development for usage in analysis code;
(3) Component integrity (e.g., fuel channel, Calandria) assessment;
(4) Regulatory issue;
(5) Severe Accident Management Guideline validations;

The above-mentioned categories are addressed through different experimental programme for Limited Core Damage Accident (LCDA) and Severe Core Damage Accident (SCDA). The program pursued are as follows:

(1) Channel Heat-up Study Programme (LCDA and SCDA);
(2) Degraded Core Study Programme (SCDA);
(3) Molten Fuel Coolant Interaction Study Programme (LCDA);
(4) Hydrogen Source Term Estimation Programme (SCDA);
(5) SAMG Guidelines Validation Programme.

The paper highlights details of the programme and the insights generated from each programme.

Advantage of commercial CFD code has been taken for developing an understanding of core damage progression. The simulation includes analysis of 2D and 3D model for half core exposed core section (Fuel Channels, Guide tubes and Calandria) to get an overview of realistic core heat up. This activity will generate insight in modelling the core degradation progression for PRABHAVINI code. A highlight on this activity will be presented.

Development of Two-Dimensional Thermal Hydraulics Code for Assessment of Performance of Passive Catalytic Recombiner Devices

A. Agarwal, P.Rahatgaonkar, S.Sharma, N.K Maheshwari, M.Kansal, K.K De (NPCIL, India)

In case of a postulated Unmitigated Severe Accident in nuclear power plant, significant quantities of hydrogen could be produced due to metal-water reaction among high temperature fuel clads/pressure tubes/calandria tubes, when exposed to steam. High concentration of hydrogen if accumulated within the containment might have potential of causing deflagration / detonation imposing potential threat to early containment failure. To mitigate the risk of hydrogen combustion Passive Catalytic Recombiner Devices (PCRDs) are installed in the containment, optimizing numbers and location. In the presence of Hydrogen and Oxygen, a catalytic reaction occurs spontaneously at the catalyst surface even in the presence of steam. Heat of reaction produced favours convection flow through the enclosure and promotes mixing in the containment.

For the assessment of the PCRD performance in terms of maximum temperature of catalyst and hydrogen removal rate of PCRD, an in-house thermalhydraulic model with catalytic chemical reaction and natural convection phenomena is developed. The results of the model are verified against the experiments carried out on REKO-3 experimental facility. The results are also verified against the experiments carried out at Hydrogen Recombiner Test Facility (HRTF), an in-house experimental facility. The developed model is integrated with in-house lumped parameter containment analysis code Post Accident Containment System Response (PACSR) and removal rate obtained from analysis is verified against experimental data.

Overview of the CFD Studies performed at the University Politechnica of Bucharest on the CANDU 6 Calandria Vessel Integrity during Severe Accidents

D. Dupleac, (University Politechnica of Bucharest, Romania)

An important feature of the CANDU reactors is represented by the large water inventories surrounding the fuel. These inventories can play an important role at severe accident prevention and mitigation, by passively removing decay heat from the fuel for several hours after an accident.

Regarding the calandria vessel integrity, the main concern is representing by the thermal loading of the calandria assembly. As long as the reactor vault water level remain above the molten debris level, the key factor to ensure the calandria vessel integrity is the local thermal margin to the CHF on the downward facing calandria vessel wall.

The paper overviews the analytical studies performed at Politehnica University of Bucharest on the CANDU 6 calandria vessel integrity during severe accidents. The analyses start from a debris bed at the bottom of calandria vessel and are performed with ANSYS-FLUENT CFD code.

From the analyses carried out, we may conclude that the calandria vessel will remain well-cooled and intact during the severe accident if the makeup cooling water is supplied to the reactor vault as the predicted maximum heat flux transferred through the calandria vessel wall is well below the CHF limits for all the cases analysed.

An Overview of the Recent MAAP-CANDU Development and Benchmarking Activities to Support CANDU Severe Accident Analyses

S.M. Petoukhov and A.C. Morreale, (CNL, Canada)

Integrated analysis of severe accidents in CANDU reactors is a complicated research area supported by both integrated analysis by simulation codes and by focused modelling and experiments investigating individual phenomena that impact severe accident progression and the radionuclide source term. MAAP-CANDU is a computer code that simulates the integrated response of a CANDU nuclear power plant to a postulated severe accident sequence. The code is currently developed and maintained by joint efforts of the CANDU Owners Group (COG) and the Electrical Power Research Institute (EPRI).

Atomic Energy of Canada Limited (AECL), and later COG, supported the development of a severe-accident-related Phenomena Identification and Ranking Table (PIRT) for CANDU 6 reactors. The CANDU 6 PIRT identified phenomena that are common to CANDUs, ranking them in order of importance and knowledge level. A Prioritization Strategy Document was prepared that identifies knowledge gaps and proposed research activities, to maximize the use of R&D resources.

The prioritization was performed on two important severe accident progression topics:

— In-Vessel Retention of corium;
— Fission product release at the site boundary.

This presentation provides an overview of progress/achievements in CANDU In-Vessel Retention R&D activities since 2018, when a previous update was presented. These activities were directed to reduce knowledge gaps identified by the PIRT.

A brief overview is also provided on the recent MAAP-CANDU benchmarking activities on several severe accident-related phenomena, such as core disassembly/collapse and reflux condensation heat transfer. .

Some insights and preliminary results are provided for the uncertainty and sensitivity analysis of a CANDU 6 plant by means of severe accident codes in the framework of the IAEA CRP I31033 entitled: “Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors”.

Finally, this presentation includes an overview of the MAAP-CANDU GRaphical Animation Package Extension (GRAPE), which is a flexible, efficient, interactive and integrated visualization tool for analysing plant behaviour during postulated accidents including accident management actions for single and multi-unit CANDU plants. GRAPE was developed by FAI in consultation with CNL (AECL) and the Canadian Nuclear Safety Commission (CNSC). CNSC uses MAAP-CANDU and GRAPE as one of the tools in their Emergency Operations Centre.

References:

S.M. Petoukhov, “Status of Severe Accident R&D to Support CANDU In-Vessel Corium Retention”, Proceedings of the International Severe Accident Management (ISAM-2018) Conference held in Ottawa, Ontario, Canada, October 15–18, 2018;

Analysis of PHWR Reactor Block Components under Postulated Severe Accident Loads

K. Mohta, O. S. Gokhale, S. K. Gupta and J. Chattopadhyay, (BARC, India)

During a severe core damage accident scenario, the reactor block components such as Calandria, end shield etc. would be subjected to large thermal and mechanical loads, which are not envisaged during the design. Sustained application of these loads would threaten the structural integrity of the components such as Calandria, which is crucial for successful implementation of in-vessel accident management strategies. For formulating/ assessing the accident management guidelines, detailed investigation is needed to understand the thermalhydraulic and structural behaviour of reactor block components, to determine the thermal and mechanical loads arising under the postulated conditions, and to assess structural integrity of Calandria and its degradation with time leading to its failure. As severe accidents are complex phenomena involving multi-physics aspects, a number of considerations are needed for performing severe accident analysis. In this section, a novel approach based on sequentially coupled thermo-mechanical analysis of Calandria assembly, to assess the structural integrity of Calandria assembly for loads arising under a postulated severe accident scenario has been discussed. Various considerations for the study, the methodology followed, tools used for analysis and post-processing of results have been covered. In addition, the failure criteria, devised as per the International Atomic Energy Agency (IAEA) guidelines, have also been discussed and structural failure assessment of Calandria assembly has been carried out. In the end, a case study on 540 MWe Indian PHWR for a postulated scenario comprising of large Loss of Coolant Accident (LOCA) along with Loss of ECCS, Loss of Moderator cooling, Loss of End Shield water cooling, i.e., unmitigated total loss of heat sink has been presented.

Experiments for PHWR Severe Accident

T. Nitheanandan (CNSC, Canada)

The Reactor Safety Science and Technology has evolved in several PHWR countries over four decades. The R&D investment from PHWR operators is gradually improving plant operations and margins. These investments have improved the understanding of severe accident progression on high-risk phenomena with uncharacterized uncertainty in knowledge.

Among the few Phenomena Identification and Ranking (PIRT) studies completed, three phenomena with “high” importance and “very-limited knowledge level and uncertainty that cannot be characterized” are cracking of crust layer on the inside vessel wall and associated heat transfer, the gap-formation between crust layer and calandria vessel wall, and coolability of corium in the basement – if in-vessel retention fails. The above phenomena aim to retain the molten corium within the calandria vessel and recognize that it significantly minimizes radiological and financial risk. The Critical Heat Flux (CHF) measured from the vessel bottom and the end-shield inner tubesheet surface strengthen the feasibility of in-vessel retention.

Studies on hydrogen mitigation and improving containment behaviour continue on several fronts. The studies on the effect of pressure loads due to the combustion of hydrogen and carbon monoxide mixture on containment and equipment have revealed a wealth of information. Research is progressing in understanding the differences in properties between deuterium and hydrogen on Passive Autocatalytic Recombiners (PARs).

An IAEA benchmarking study, “Benchmarking Severe Accident Computer Codes for Heavy Water Reactor Applications,” compared seven severe accident computer codes and identified areas of improvement required in model development and code enhancement. The study concluded that there was good agreement on the timing of predicted events with less complex, early stages of the sequence. In contrast, the agreement deteriorated with late-stage phenomena with more complex phenomena like core disassembly, debris oxidation, and corium-vessel interactions.

Summary

Although significant improvements in understanding severe accident phenomena have been achieved, the remaining phenomena are more complex and challenging. The PHWR community needs to initiate joint R&D activities to improve the knowledge level of the remaining complex phenomena and reduce code calculation uncertainties.

Phenomena Identification and Ranking Table for Severe Accident Analysis of PHWRs - An Indian Perspective

S. K. Pradhan, (AERB, India)

The mainstay of the Indian nuclear power programme is Pressurized Heavy Water Reactors (PHWRs), starting from Rajasthan Atomic Power Station in 1973. The programme has come a long way with PHWR units of 220MWe, 540MWe and 700MWe. Post Fukushima, the severe accident analysis and experimental studies gained importance and significant efforts are being dedicated to study the involved phenomena for PHWRs also. To consolidate the information on PHWR Safety R&D on Severe Accidents, a State-of- the-art Report (SOAR) was prepared. The report focuses on identification and prioritization of the gap areas for further resolution through analytical and experimental activities with suggested way forward and timelines, taking into account the existing infrastructure and capabilities of various stakeholders. The presentation will highlight the phenomena identification and ranking table for channel integrity, calandria integrity, fuel behaviour, reactions at high temperature, source term estimation, containment integrity, molten fuel coolant interaction (MFCI), molten corium concrete interaction (MCCI) etc. as brought out in the SOAR as well as recent developments in this direction.

An Update on Severe Accident R&D Prioritization Strategy to Support CANDU In-Vessel Corium Retention

S.M. Petoukhov and J. Spencer, (CNL, Canada)

Integrated analysis of severe accidents in CANDU reactors is a complicated research area supported by both integrated analysis by simulation codes and by focused modelling and experiments investigating individual phenomena that impact severe accident progression and the radionuclide source term. MAAP-CANDU is a computer code that simulates the integrated response of a CANDU nuclear power plant to a postulated severe accident sequence. The code is currently developed and maintained by joint efforts of the CANDU Owners Group (COG) and the Electrical Power Research Institute (EPRI).

Atomic Energy of Canada Limited (AECL), and later COG, supported the development of a severe-accident-related Phenomena Identification and Ranking Table (PIRT) for CANDU 6 reactors. The CANDU 6 PIRT identified phenomena that are common to CANDUs, ranking them in order of importance and knowledge level. A Prioritization Strategy Document was prepared that identifies knowledge gaps and proposed research activities, to maximize the use of R&D resources.

The prioritization was performed on two important severe accident progression topics:

- In-Vessel Retention of corium,
- Fission product release at the site boundary.

This presentation provides an overview of progress/achievements in CANDU In-Vessel Retention R&D activities since 2018, when a previous update was presented. These activities were directed to reduce knowledge gaps identified by the PIRT.

A brief overview is also provided on the recent MAAP-CANDU benchmarking activities on several severe accident-related phenomena, such as core disassembly/collapse and reflux condensation heat transfer. .

Some insights and preliminary results are provided for the uncertainty and sensitivity analysis of a CANDU 6 plant by means of severe accident codes in the framework of the IAEA CRP I31033 entitled: “Advancing the State-of-Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water-Cooled Reactors”.

Finally, this presentation includes an overview of the MAAP-CANDU GRaphical Animation Package Extension (GRAPE), which is a flexible, efficient, interactive and integrated visualization tool for analysing plant behaviour during postulated accidents including accident management actions for single and multi-unit CANDU plants. GRAPE was developed by FAI in consultation with CNL (AECL) and the Canadian Nuclear Safety Commission (CNSC). CNSC uses MAAP-CANDU and GRAPE as one of the tools in their Emergency Operations Centre.

Reference:

S.M. Petoukhov, "Status of Severe Accident R&D to Support CANDU In-Vessel Corium Retention", Proceedings of the International Severe Accident Management (ISAM-2018) Conference held in Ottawa, Ontario, Canada, October 15–18, 2018;

Severe Accident Experiments and Phenomenological Analysis at Canadian Nuclear Laboratories in the Past Decade

J. Spencer, (CNL, Canada)

Over the past decade, Canadian Nuclear Laboratories (CNL) has pursued improvements in understanding severe accident phenomena in pressurized heavy water reactors (PHWRs) through simulations and experiments. Given the importance of in-vessel melt retention (IVMR) and core disassembly to accident progression in PHWRs, a high priority has been given to investigating phenomena relevant to these areas.

This presentation will include a short overview of the progression of an accident to the IVMR core damage state and the key phenomena involved. A brief summary of phenomenological investigations performed at CNL will be provided in this presentation, including: the loading limits of the calandria tube–rolled joint system, terminal debris bed (TDB) formation, critical heat flux (CHF) at various locations around the calandria vessel periphery, PHWR corium miscibility, corium ingression into vessel penetrations, and the response of the calandria vessel to heat stresses. Finally, a more detailed description will be provided of experiments and simulations investigating the spatial distribution of exiting heat flux during IVMR.

Generic Approach to Address H2 Risk Evaluation Including Modelling and Simulation Tools

A. Bentaib, (IRSN, France) and S. Gupta, (Becker Technologies, Germany)

During a severe accident in a nuclear water-cooled reactor, large quantities of hydrogen or Deuterium can be generated and released into containment as the reactor core degrades. Additional burnable gases (H_2 and CO) may be released into the containment in the event of corium/molten concrete interaction (MCCI). This could then create a risk of combustion. As was observed during the Fukushima accidents, the combustion of hydrogen can cause high pressure peaks that can undermine the reactor containment and lead to the failure of surrounding buildings. A hydrogen explosion can also pose a safety problem in spent fuel storage areas, where flammable conditions can be reached in the absence of adequate ventilation. In this case, the hydrogen explosion could lead to the dispersion of radioactive products in the environment. To prevent the risk of a hydrogen explosion, most of the mitigation strategies adopted in Western countries are based on the use of passive autocatalytic recombiners (PARs). However, studies of representative accident sequences indicate that, despite the installation of PARs, it is difficult to prevent, at any time and for any location, the formation of a combustible mixture that could lead to a local acceleration of the flame. In order to gain a better understanding of the phenomena associated with the risk of combustion and to resolve the problems highlighted after the events at Fukushima Dai-ichi, such as the risk of explosion inside ventilation systems or the potential migration of a flammable mixture into spaces beyond the primary containment, additional R&D projects have recently been launched.

This paper provides an overview of recent experimental and analytical works that has been conducted in order to improve the knowledge related to hydrogen recombination and combustion and to achieve a high level of confidence in the related simulations.

THAI Containment Safety Research Under Severe Accident Condition and its Use for Code Validation
S. Gupta, (Becker Technologies, Germany)

Safety assessment and accident management in nuclear power plants (NPPs) necessitate investigating complex phenomena and processes with adequate accuracy. In support of such activities, THAI research programme aims at investigating open questions on hydrogen and fission product behaviour in the containment of water-cooled reactors in the frame of national projects sponsored by the German Federal Ministry for the Environment, Nature Conservation, Nuclear Safety and Consumer Protection (BMUV), and international projects which run under the auspices of OECD Nuclear Energy Agency (NEA). Presently, Phase VII of THAI national project that includes water-cooled small modular reactor relevant topics and OECD/NEA THEMIS project including pressurized Heavy Water Reactor Technologies relevant topics like combustion behaviour and mitigation of deuterium containing gas atmosphere are ongoing.

In general THAI experimental research aims to evaluate the combustible gas risk by investigating mixing and distribution phenomena of hydrogen (carbon monoxide)-steam-air mixtures, passive autocatalytic recombines performance behaviour and hydrogen combustion behaviour under typical accident conditions e.g., elevated pressure/temperature, steam, stratified and well mixed atmospheric conditions. Effect of engineered safety systems (e.g., spray operation) on hydrogen mixing and combustion behaviour has also been subject of investigation in the concluded THAI projects. Investigations on source-term relevant phenomena focus to provide in-containment fission product behaviour under representative boundary conditions and includes issues like fission product (aerosols, iodine) resuspension and remobilization (entrainment, pool scrubbing).

The experimental data produced in the frame of various THAI projects have contributed continuously to the validation and further improvement of simulation codes used for reactor application, e.g., by use of THAI data for International Standard Problem ISP-47 on containment thermalhydraulics & ISP-49 on hydrogen combustion, PAR code benchmarks in OECD/NEA THAI, THAI-2 & 3 and EC-SARNET projects. Use of THAI/THAI+ experiments for validation of LP- and CFD-codes as well as for reactor application purpose are illustrated with selected examples in the present paper,

Tentative structure for the presentation is provided below:

- Introduction – National and International (OECD/NEA) THAI projects;
- Experimental activities- Hydrogen risk ▪ Gas distribution and thermalhydraulics;
  - Hydrogen (Deuterium) mitigation – Passive Autocatalytic recombiners:
  - Hydrogen (Deuterium) deflagration;

  - o Hydrogen deflagration & spray interaction;
  - o Hydrogen combustion effect on source-term.
- Experimental activities- Source Term ▪ Fission product distribution and interaction with safety systems:
  - o Fission product deposition and washdown from surfaces and gas atmosphere (subject also relevant for water submerged SMRs);
  - o Fission product remobilization behaviour from water pools and surfaces;
- Use of THAI data for code application – selected examples of national and international code benchmarks and code validation activities - Containment thermalhydraulics/gas distribution
  - o Hydrogen combustion;
  - o Hydrogen mitigation- Passive Autocatalytic Recombiners;
  - o Multi-compartment iodine behaviour;
  - o Effect of Spray operation on hydrogen combustion and gas borne fission product washout behaviour.
- Information about relevant external (joint) projects, e.g., NUGENIA/IPRESCA;
- Conclusions & Perspectives.

ABREVIATIONS

ACR	Advanced CANDU Reactor
AECL	Atomic Energy of Canada Limited
AERB	Atomic Energy Regulatory Board
AMP	Accident Management Plan/Programme
AST	Alternative Source Term
ASTEC	Accident Source Term Estimation Code
BARC	Bhabha Atomic Research Centre
BARCOM	BARC Containment
BDBA	Beyond Design Basis Accident
PRABHAVINI	PRogram for Anwik BHatti Apakarsh and VIkiran NIrdharan
BWR	Boiling Water Reactor
CABS	Catalyst Activity Bench Scale
CAISER	CANDU Advanced Integrated SEveRe code
CANDU	CANada Deuterium Uranium
CCIF	Cold crucible induction facility
CDF	Core Damage Frequency
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CNCAN	National Commission of Nuclear Activities Control
CNL	Chalk River Laboratories
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owner’s Group
CRP	Coordinated Research Project
CSA	Canadian Standards Association
CSTS	Calandria Side Tube Sheet
CV	Calandria Vessel / Calandria

DBA	Design Basis Accident
DEC	Design Extension Condition
DFC	Diagnostic Flow Chart
DiD	Defence in Depth
ECCS	Loss of Emergency Core Cooling System
EOC	Emergency Operations Centre
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
FAI	Fauske and Associates Inc., LLC
FCI	Fuel Coolant Interaction
FEA	Finite Element Analysis
FP	Fission Product
FPG	Fission Product Group
FSAR	Final Safety Analysis Report
GRAPE	GRaphical Animation Package Extension
GUI	Graphical User Interface
IAEA	International Atomic Energy Agency
IRSN	Institute for Radiation Protection and Nuclear Safety
IVMR	In-Vessel Melt Retention
KAERI	Korea Atomic Energy Research Institute
KINS	Korea Institute of Nuclear Safety
LCDA	Limited Core Damage Accidents
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MCCI	Molten Corium Concrete Interaction

MFMI	Molten Fuel Moderator Interaction
MS	Member States
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
NSA	Nuclear Safety Act
NSSC	Nuclear Safety and Security Commission
OECD	Organization for Economic Cooperation and Development
PHTS	Primary Heat Transport System
PHWR	Pressurized Heavy Water Reactor
PIRT	Phenomena Identification and Ranking Table
PRABHAVINI	PRogram for Anwik BHatti Apakarsh and VIkiran NIrdharan
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RB	Reactor Building
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SBO	Station Black-Out
SCDA	Severe Core Damage Accident
SCG	Severe Challenge Guidelines
SCST	Severe Challenge Status Tree
SG	Steam Generator
SRG	Standard Review Guides
US NRC	United States Nuclear Regulatory Commission
VVER	Water-Water Energetic Reactor
WL	Whiteshell Laboratories

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