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IAEA Collaborations on Advancing Water Cooled Reactor Technologies

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Abstract

Since the Fukushima Daiichi accident, the International Atomic Energy Agency (IAEA) has been exchanging information on the accident progression and technical lessons learned toward enhanced safety design of water-cooled reactors (WCRs). The lessons have been discussed and updated with new information and utilized to identify the needs in research and development (R&D) in cooperation with the experts from Member States (MSs). The IAEA therefore provided several fora to facilitate the exchange of information on R&D activities and to further strengthen international collaboration among the MSs. One of the main findings was that the Fukushima Daiichi accident highlighted a number of challenges that should be addressed by reconsidering relevant R&D strategies and priorities. Along these the IAEA has organized a number of technical meetings and coordinated research projects.

International collaborations also extend in the heavy water reactor (HWR) area involving technology issues that are specific to HWRs, such as investigations into phenomena associated with pressure tubes, thermal-hydraulics, and core behaviour during severe accidents, or beyond-design-basis-accidents. The goals are to increase the fundamental understanding and improve the modelling tools. Ongoing and recently completed collaborations aim to improve insights into pressure-tube creep behaviour, modelling of the moderator as a passive heat sink in severe accidents, and multi-physics simulations of CANDU core transients. Along with focussed code comparison exercise using large-scale experiments, these collaborations provide participants from R&D, operating, and regulatory bodies valuable data against which analysis methods and codes can be benchmarked.

This paper outlines the outcomes from recently completed or ongoing few IAEA projects and activities pertaining to innovations in safety and technology of the WCRs.

1. Introduction

Since the 2011 Fukushima Daiichi accident, the IAEA is pursuing extensive collaborative projects on the accident progression and technical lessons learned with the goal to apply the new knowledge into enhanced safety of water cooled reactors (WCRs) [1, 2]. After the Fukushima Daiichi accident, new R&D activities have been undertaken by many countries and international organizations in regard to evaluation and understanding of the severe accidents. The IAEA held, in cooperation with the OECD Nuclear Energy Agency (OECD/NEA), the International Experts' Meeting (IEM) on "*Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant*" in February 2015. The objective of this IEM was to facilitate the exchange of information on R&D activities and to further strengthen international collaboration among Member States and international organizations. At the IEM, international experts discussed the effective utilization of R&D to enhance nuclear safety worldwide. From their presentations and discussion, the following lessons learned were identified for strengthening the effectiveness of R&D in the light of the Fukushima Daiichi accident [3]:

1) R&D strategies after the Fukushima Daiichi accident: The Fukushima Daiichi accident revealed the need to reconsider several aspects of existing R&D strategies to ensure that they continue to be effective.
2) Measures to protect nuclear power plants against external and internal events: The Fukushima Daiichi accident highlighted the need for further R&D activities to strengthen measures to protect nuclear power plants against external and internal events as well as to consider the impact of such events, occurring subsequently or simultaneously, on multi-unit NPP sites.
3) Technologies to prevent severe accidents and mitigate their consequences: Continual R&D efforts are necessary to develop robust measures for core cooling and containment venting that can be implemented even under unexpected conditions.
4) Severe accident analysis: Increased attention needs to be given to R&D for severe accidents and their management, both for existing NPPs and for the design of future NPPs.
5) Emergency preparedness and response: R&D efforts are needed to harmonize rapid response strategies to nuclear or radiological emergencies at the national, regional and international levels. Harmonization of R&D will also contribute to improved emergency assessment and prognosis by supporting improved modelling of radionuclide transport and dispersion, and exchange of data and key information.
6) Post-accident recovery: The circumstances resulting from severe accidents are usually unprecedented and unplanned. A number of challenges in the phase of post-accident recovery arise, particularly with regard to decommissioning a damaged reactor. Post-accident recovery activities will need considerable R&D effort, and international collaboration and information exchange in this respect is essential.

One of the main conclusions of the IEM was that the Fukushima Daiichi accident had not identified completely new phenomena to be addressed, but that the existing strategies and priorities for R&D should be reconsidered. Significant R&D activities had been already performed regarding severe accidents in WCRs before the Fukushima Daiichi accident, and the information was very useful for predicting and understanding the accident progression. However, the Fukushima Daiichi accident highlighted several challenges that should be addressed by reconsidering R&D strategies.

As one of the IEM follow-up meetings, the IAEA held a Technical Meeting on "*Post-Fukushima Research and Development Strategies and Priorities*" in December 2015. The objective of this meeting was to provide a platform for experts from MSs and international organizations to exchange perspectives and information on strategies and priorities for R&D regarding the Fukushima Daiichi accident and severe accidents in general. The experts discussed R&D areas that need further attention and the benefits of possible international cooperation, and developed a list of recommendations and suggestions for further R&D activities including [4, 5]:

- Robust measures for reactor core cooling, depressurization and removing heat to the ultimate heat sink;
- Experimentation and analysis of in-vessel melt retention (IVMR) and ex-vessel corium cooling;
- Enhancement of the understanding of the Fukushima Daiichi accident progression including the condition of core debris inside the reactor and containment vessels;
- Severe accident analysis models and codes development including benchmarking and validation; and
- Better understanding of safety issues concerning multiple unit sites and multiple sites.

To support these activities and foster international cooperation the IAEA conducted a number of meetings and initiated cooperative research projects. The overview of some of these activities is provided in the following sections.

2. IAEA Collaborations on in Advancing Safety of WCRs

2.1 IAEA Activities on Technology Development to Cope with Severe Accidents in WCRs

It has been highlighted during the IEM and confirmed at the TM held in 2015 that the R&D area regarding in-vessel melt retention and ex-vessel corium cooling is one of the highest priority areas, and that more phenomenological knowledge should be gained for the strategic and technological development of the countermeasures to cope with WCR severe accidents.

Based on the high interest in Member States, the IAEA organized a TM on "Phenomenology and Technologies Relevant to In-Vessel Melt Retention and Ex-Vessel Corium Cooling" at the Shanghai Nuclear Engineering Research and Design Institute (SNERDI), Shanghai, China, in October 2016. The purpose of the meeting was to provide a platform for detailed presentations and technical discussions on recent progress in R&D activities on in-vessel melt retention and ex-vessel corium cooling during severe accidents at WCRs. More than 60 experts gathered at the meeting and discussed their recent activities related to in-vessel melt retention and ex-vessel corium cooling during severe accidents at water-cooled reactors, and they updated together the scientific and engineering knowledge in this area. The presentations given at the meeting and the meeting summary are found on the IAEA website [6].

2.2 Coordinated Research Project (CRP) on Application of Computational Fluid Dynamics (CFD) Codes for Nuclear Power Plant (NPP) Design

Experts envision the future application of Computational Fluid Dynamics (CFD) codes, once they are properly verified and validated, to substitute for the expensive experimental testing associated with NPP design in general. These codes provide inexpensive qualitative and quantitative information in many key areas for which traditional design tools are limited: i.e. where three-dimensional motions play a significant role. The technology is already well-established in other important industrial design areas, such as in the aerospace, automobile, chemical and turbo-machinery industries. The nuclear industry recognises that CFD codes have reached the desired level of maturity, at least for single-phase applications, to be used in the NPP design process. Here we present the scope of the IAEA CRP in assessing the CFD codes' current capabilities, and how the state-of-the-art codes contribute to the technology advance in respect to their verification and validation.

The use of single-phase CFD codes for nuclear applications has evolved from the sub-channel analysis codes developed in the 1980s. The ability of CFD to simulate the three-dimensional aspects of various NPP phenomena, including pressurized thermal shock (PTS), boron dilution, thermal fatigue, hydrogen distribution in containments, hot-leg temperature homogeneities, etc., has brought the technology to the forefront of NPP safety and design considerations. However, the use of two-phase CFD codes for nuclear applications remains at a lower level of maturity, and may be considered useful for guidance purposes only, but not yet able to produce results of quantitative reliability, except possibly for some dispersed flow situations. Of particular relevance is the use of CFD in the design of advanced nuclear reactors: that is, those of type GEN-III, GEN-III+ and GEN-IV. In particular, CFD codes are useful in understanding and simulating the innovative technologies employed by these new NPP concepts.

The development, verification and validation of CFD codes in respect to NPP design necessitates further modelling work on the complex physical processes involved, and on the development of efficient numerical schemes needed to solve the basic equations in an efficient manner. In parallel, it remains an overriding priority to benchmark the performance of such CFD codes, and for this experimental databases need to be established, both in terms of separate-effect tests and for full-size integral tests. The scaling issue remains paramount to component and, particularly, containment testing, but CFD does provide a means to explore the phenomena in advance of expensive experiments.

The IAEA CRP aims to assess the capabilities of CFD to address these multifarious design aspects, to identify the shortcomings in this technology approach, and to document the state-of-the-art of CFD in respect to being considered essential in the NPP design. Verification and validation (V&V) is a critical factor in assessing the overall capabilities of CFD, and as such is addressed conscientiously within the framework of this CRP. Four "open" CFD benchmark exercises, with a total of seven individual experiments, were chosen for simulation by the CRP participants. Two are based on the ROCOM (Rossendorf Coolant Mixing Model) facility at HZDR (Helmholz Zentrum Dresden Rossendorf): one simulating a Pressurized Thermal Shock (PTS) scenario (under two conditions) and the other with a (simulated) boron injection, and two benchmarks are based on 4x4 PWR fuel bundle tests, which were performed at the Korea Atomic Energy Research Institute (KAERI), investigating the flow and temperature fields with pitch-diameter ratios (P/D = 1.35 and P/D = 1.08) under adiabatic and heated (one rod) conditions. Full geometry and CAD data of the test sections were made available, together with the inlet and boundary conditions, for all four benchmarks. The CFD simulations and the two benchmarks analysis are currently underway at the participant institutes and three publications are planned in 2017 to document the status of CFD use in NPP design.

2.3 Technology Development for HWR

The IAEA fosters international cooperation on technology development for improved safety of the HWRs with the goals to increase fundamental understanding and improve the modelling tools. Ongoing and recently completed collaborations aim to improve insights into pressure-tube creep behaviour, modelling of the moderator as a passive heat sink in severe accidents, and multi-physics simulations of CANDU core transients. Along with focussed code comparison exercises using large-scale experiments, these collaborations provide participants from R&D, operating, and regulatory bodies valuable data against which analysis methods and codes can be benchmarked.

Currently, three international research collaborations related to HWR core technology are ongoing among the MSs that operate HWRs. As described in the following sections, the IAEA currently conducts two International Computational Standard Problems (ICSP), and one CRP related to unique HWR/CANDU safety and ageing issues, as follows:

1) ICSP on *HWR Moderator Subcooling Requirements to Demonstrate Backup Heat Sink Capabilities of Moderator during Accidents*
2) ICSP on *Numerical Benchmarks for Multiphysics Simulation of PHWR Transients*
3) CRP on *Prediction of Axial and Radial Creep in Pressure Tubes*

2.3.1 ICSP on HWR Moderator Subcooling Requirements to Demonstrate Backup Heat Sink Capabilities of Moderator during Accidents

An inherent safety feature of HWRs is the presence of the low-pressure moderator as a passive backup heat sink during emergencies that involve overheating of the fuel. The pressure tube in a CANDU fuel channel is normally separated from the surrounding calandria tube by ~8-mm CO_2-filled annular gap, which thermally isolates the pressure tube (PT) from the calandria tube (CT). Heat transfer occurs primarily by conduction through the gas and by thermal radiation. During abnormal accident conditions, however, the PT can overheat and undergo radial plastic deformation. This permanent dimensional change as a result of the effects of stress and temperature is known as *PT ballooning*. When a PT balloons and comes into a contact with the CT, the resultant contact heat transfer significantly and almost instantaneously increases the rate of heat transfer to the CT, and subsequently, to the moderator. The rate of heat transfer to the CT is determined by the temperature difference between the PT and the CT and by the contact heat transfer coefficient, which depends on the contact pressure. The temperature of the CT is determined by the moderator subcooling and the heat transfer coefficient, or the boiling regime, between the CT and the moderator. At the time of contact, the CT experiences a large increase in heat flux at the contact locations, as stored heat is rejected from the PT to the cooler CT. If the heat flux on the outer surface of the CT exceeds the critical heat flux (CHF), film boiling (dryout) may occur on the surface of the CT. If the area in dryout is sufficiently large and the dryout is prolonged, the pressure-tube/calandria-tube combination can continue to heat up and strain radially, ultimately leading to fuel-channel rupture.

Table 1: Calculated and Measured Time and Temperature at Initial PT-CT Contact

Participants' Model	**Time of first PT/CT contact (s)**	**Bottom PT temperature at time of contact, axial centre**	**Top PT temperature at time of contact, axial centre**
Model 1	71.1	719	719
Model 2	78.6	551	708
Model 3	72.3	801	801
Model 4	76.3	796	803
Model 5	71.2	825	819
Model 6	71.9	811	811
Model 7	77.7	904	904
Model 8	75.2	762	763
Model 9	72.0	804	804
Model 10	76.2	763	763
Model 11	73.9	799	804
Experiment	**71.3**	**890**	**860**

An experiment was performed in 2014 by Canadian Nuclear Laboratories to investigate the fuel channel behaviour under prototypical conditions. This test was similar to the ones performed in the 1980's and 90's [7]. This experiment comprised a transient heat up where the PT experienced an average heatup rate of 21ºC/s at an internal pressure of 3.5MPa with the moderator at 70ºC, corresponding to 30°C subcooling. Important measurements included transient, post-test tube thicknesses, and drypatch areas on the CT outside surface. Nucleate, intermittent, and film boiling regimes after contact, and the development and subsequent rewetting of dry patches, were observed. Table 1 shows the contact time and corresponding PT temperature, respectively, at the bottom and bottom, axially central position. The model results are from blind calculations, but did not change significantly in the open calculations. The initial contact time is predicted with relatively small errors, considering that in the experiment initial contact was at 71.2s and it took about 3.5s to full contact. The contact temperature of the PT is generally underpredicted, in part due to the measurement location being inside the PT material rather than on the outer surface, and partially due to most models using a bulk PT temperature to calculate its deformation rate.

Other results showed that, while the PT average radial creep deformation rate, the time of PT to CT contact and the contact temperatures were generally predicted well, the CT dryout and rewet behavior were not consistent. The large scatter in the post-contact results in the blind calculations, for example the space and time extent of dryout, was caused by not applying best-practice for key phenomena, namely the contact conductance and boiling models. This was the key lesson from this exercise and, after agreeing on the best-practice assumptions and methods/models, the open calculations had considerably less scatter. Open results are still being examined in detail and will be published in the future. The blind simulations achieved their objectives and provided significant insights into analysis codes and user effects. Furthermore, it was clear that both simple and complex codes can adequately capture the behaviour of dryout and rewet, provided a consistent and best-practice approach is taken and boundary conditions are carefully applied.

Nine organisations from five Member States with heavy water reactor technology participated in the blind and open simulations, using 11 different computer codes or methods, ranging from lumped-parameter to 2D and partially 3D methods, to capture the complex physical behaviour and interactions of the various important thermal and mechanical phenomena.

2.3.2 ICSP on Numerical Benchmarks for Multiphysics Simulation of Pressurized Heavy Water Reactor (PHWR) Transients

Several countries operating PHWRs have active research programs to investigate multi-physics code simulation as part of code modernization initiatives. The choice of a coupling algorithm is expected to involve several considerations such as applicability of the proposed multi-physics system to different reactor transients, the state of existing and prototype codes, and the required precision in the results of the analyses. These choices can be expected to impact overall simulation accuracy, computing time and resource demands, and the range of applicability of the system.

Open access to standardized numerical benchmarks, representative of PHWR transients, will facilitate testing of different code coupling methods. The participants are the organizations from Canada, China, India, Republic of Korea, and the USA. One of the main benefits to participants is peer-review of their results. The following transient benchmarks, based on a simplified CANDU six- reactor core configuration, are included:

a) Steady-State (to be used for (a) testing of coupling methods and sensitivities, plus "tuning of models" in order to (b) achieve similar starting conditions for better comparison of subsequent transient benchmarks)
b) Local (asymmetric) Reactivity Insertion
c) Loss of Flow (one pump)
d) Loss of Coolant Accident (~30% reactor outlet header break)

For each benchmark, the main objectives and Figure(s) of Merit (FOM) have been defined, and several proposed modelling methods, issues, simplifications, and user guidelines discussed. The benchmarks will provide a means to test current transient simulation methods, which are generally based on coupling few-group diffusion neutronic codes with thermal-hydraulic codes. In addition, the benchmarks will permit future reactor simulation methods to be tested. For example, while full core transport methods for PHWR transient simulation are not currently widely available, the development of such methods is being actively pursued worldwide. The list of benchmarks may be extended to incorporate new scenarios and solutions as they are developed.

This ICSP started in 2016 and is planned to be completed in 2019. A Technical Committee, consisting of representatives from Canada, the Republic of Korea and India, was nominated to lead the drafting of the *Benchmark Specification Document*.

2.3.3 CRP on Prediction of Axial and Radial Creep in Pressure Tubes

Pressure tube deformation is a critical aging issue in operating PHWRs. Over time, horizontal pressure tubes deform under the influence of pressure, temperature and fast neutron flux in three ways: diametral (or transverse) creep - leading to flow bypass and reduced critical heat flux for fuel rods, longitudinal (or axial) creep - leading to interference between feeder pipes and/or with the fuelling machine, and sag - leading to interference with in-core components. Of these, pressure tube diametral creep is of primary interest, and is schematically shown in **Fig. 1**. A CRP has been established under the auspices of the IAEA and with participation from all HWR operating countries to address this issue with a three-pronged approach:

a) Establishment of the first all-inclusive international database of in-reactor measurements of the pressure tube diametral creep,
b) Microstructure characterization of pressure tube material coupons collected from currently operating HWRs, and
c) Development of prediction models or methods for pressure tube deformation.

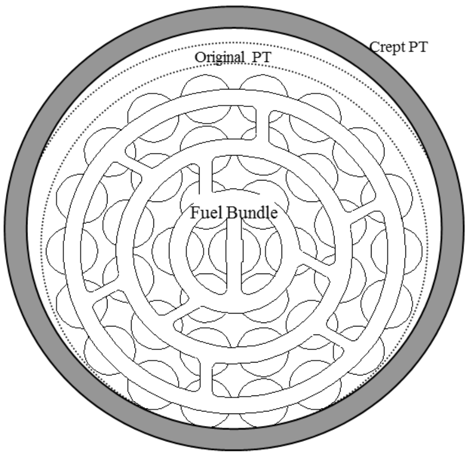

Figure 1. Schematic Presentation of the Diametral Creep Formation in HWRs

Presently, in each individual reactor unit the measured creep data, which is obtained by gauging select tubes during outages, is used to extrapolate future pressure tube performance for the entire core and to determine reactor shutdown trip setpoints. The extrapolation is typically done by applying empirical equations that consider the pressure tube's historical operating conditions of pressure, temperature and neutron flux or fluence, and that are optimized for each plant or unit. Application of one plant's model to other plants, through the examination of resulting prediction biases, has revealed trends, which may point to material-intrinsic factors, i.e. microstructure characteristics that affect pressure tube creep behaviour. An extensive International Database on *Heavy Water Reactor Pressure Tube Diametral Creep* that includes about 600 gauged tubes has been assembled. It presents the possibility for the CRP participants to investigate parametric trends through comparisons between their predictive models and other reactor units' measurements, and gain new insights, in particular because it combines CANDU 6 and Indian HWR data. Another TopSafe 2017 paper provides more details on the pressure tube creep database [8].

2.4 Near-Term Planned CRP on Probabilistic Safety Analysis (PSA) for Multi-Unit, Multi-Reactor Sites

At many nuclear sites, several nuclear power plants (NPPs), either of the same or of different types, designs, or age, are co-located on a single site. While regulations generally recognize the potential for multiunit accidents, probabilistic risk assessments of NPPs have mainly focused on estimating the risk arising from damage to a single NPP. Safety assessments in the past have used deterministic and probabilistic approaches in which the risk at a site with multiple reactors can be represented by summing up the risks of individual units. This simplified approach has several limitations as it ignores potentially complex interactions during a severe event impacting a multiunit site. This CRP is also of interest to other reactor types, including SMRs.

One of the lessons from the Fukushima accident was the need for improving PSA methodologies when applied to multi-unit, multi-reactor-type nuclear sites. Several methods have been proposed and are being developed around the world to extend or "translate" per-unit PSA results to multi-unit site PSA results, such as core damage frequency and large release frequencies. This CRP will bring together experts from LWR and PHWR MSs to consolidate their current (and planned) practices, assumptions and results and, if required, develop a new framework for establishing risk factors, such as core damage frequency and large releases from multi-unit, multi-reactor sites, from existing or new PSA studies. Finally, technology solutions to reduce these risks will be explored.

3. Conclusion

In 2015 the IAEA held, in cooperation with the OECD Nuclear Energy Agency (OECD/NEA), the International Experts' Meeting (IEM) on "*Strengthening Research and Development Effectiveness in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant*." The objective of this IEM was to facilitate the exchange of information on R&D activities and to further strengthen international collaboration among Member States and international organizations. Following the outcomes of this meeting the IAEA pursued international collaborations on recommended topics toward improved and enchased safety aspects of WCRs. In this paper we summarized a few specific projects that the IAEA developed in collaboration with the Member States on the topics pertaining to improved safety of the advanced water cooled reactors pointing at strengthening international R&D strategies in the light of Fukushima Daiichi accident lessons learned and identified R&D needs, such as: phenomenology and technologies relevant to in-vessel melt retention and ex-vessel corium cooling, investigations into HWR phenomena associated with pressure tubes, thermal-hydraulics, and core behaviour during severe accidents, or beyond-design-basis-accidents. The goals of these projects and international cooperation are to exchange information and increase the fundamental understanding of reactor core transients and severe accidents. These presented projects and meeting engaged experts from over 50 Member States participating in execution of the projects, and meeting presentations.

Acknowledgments

The IAEA expresses its appreciation to all experts participated and participating in the technical meetings, CRPS and ICSP presented in this paper.

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