RECENT IAEA ACTIVITIES TO SUPPORT ADVANCED WATER COOLED REACTOR TECHNOLOGY DEVELOPMENT

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Abstract

The International Atomic Energy Agency (IAEA) is the world's center of cooperation in the nuclear field. The IAEA works with its Member States and multiple partners worldwide to promote safe, secure and peaceful nuclear technologies. To catalyse innovation in nuclear power technology in Member States, the IAEA coordinates cooperative research, promotes information exchange, and analyses technical data and results, with a focus on reducing capital costs and construction periods while further improving performance, safety and proliferation resistance. This paper summarizes the recent major IAEA activities to support technology development for water cooled reactors, which is the most common type of reactor design at present and will probably still be in the near future.

1. Introduction

In the past 50 years, nuclear power has grown from a new scientific development to become a major part of the energy mix in many countries. After a decade of low interest in nuclear power, the nuclear field is now experiencing a renaissance and many countries are planning to increase their nuclear capacity or to introduce their first nuclear power plant. Water cooled reactors represent more than 90% of the current world fleet, and it is foreseen that in the near term most new nuclear power plants (NPPs) will be evolutionary water cooled reactors.

Achieving economic competitiveness with other energy sources and assuring very high safety levels are key goals of new plant development. The historically high capital cost of nuclear power plants presents a significant challenge to the addition of new nuclear power capacity. Design organizations are challenged to develop advanced nuclear power plants with lower capital costs and shorter construction periods, in sizes suitable for various grid capacities and owner investment capabilities. New nuclear power plant designs are being developed to meet stringent safety requirements. While there are differences in safety requirements among countries developing new designs, the stringent requirements are generally reflected in the IAEA's safety standard series [1].

The mission of the IAEA Division of Nuclear Power is to increase the capability of interested Member States to establish/develop, implement and maintain competitive and sustainable nuclear power programmes and to develop and apply advanced nuclear technologies. The IAEA frames to support the development and the application of the advanced nuclear power technologies includes:

- Coordinated Research Projects (CRP)
- International Collaborative Standard Problems (ICSP)

- Technical Meetings (TM)
- International Collaborative Assessments (ICA)
- Training Courses and Workshops.

This paper summarizes the major recent and ongoing IAEA activities and publications in the area of water cooled reactor technology development.

2. Water cooled reactor technology development publications

The recently released Nuclear Energy Series publications of the IAEA Department of Nuclear Energy support the various needs of all Member States. Member States without developed nuclear programmes will find guidance for the appropriate development of their infrastructure and for the establishment of their entire programmes. On the other hand, Member States with mature nuclear programs will find publications for the continuous improvement of their existing nuclear power programmes and for their expansion.

Many design organizations have been developing advanced water cooled reactors. To provide IAEA Member States with the state-of-the-art information on advanced reactor designs, the IAEA has been periodically issuing status reports based on the technical information collected from the designers [2-5]. These documents present an overview of development trends and goals, as well as detailed descriptions of advanced water cooled reactor designs according to a common outline. The descriptions include summaries of measures taken by the designers to enhance economics and maintainability.

Advanced applications of nuclear energy include seawater desalination, district heating, heat for industrial processes, and electricity and heat for hydrogen production. In addition, in the transportation sector, since nuclear electricity is generally produced in a base load mode at stable prices, there is considerable near-term potential for nuclear power to contribute as a carbon-free source of electricity for charging electric and plug-in hybrid vehicles. IAEA-TECDOC-1584 [6] examines the potential of nuclear energy to expand into these markets by presenting an overview of sample applications, their opportunities, challenges and solutions.

IAEA-TECDOC-1487 [7] defines, collates, and presents the state-of-the-art in design features and approaches used to protect nuclear plants from external event impacts, making a focus on nuclear power plants (NPPs) with evolutionary and innovative designs. This document reflects best practices in Member States and provides technical background to assist designers of advanced NPPs in defining consistent strategies regarding selected design and site evaluation issues in relation to extreme external events. The issues addressed include siting, hazard definition and event combination criteria, reactor and plant design including inherent and passive safety features and active and passive systems, component qualification, and external event PSA.

With increasingly liberalized electricity markets around the world, incentives have increased for identifying means to achieve better NPP economics. The task on “Optimizing Technology, Safety and Economics of Water-Cooled Reactors” was carried out during 1999-2002. Its objective was to emphasize the need, and to identify approaches, for new nuclear plants with water-cooled reactors to achieve competitiveness while maintaining high levels of safety. To achieve the largest possible cost reductions, proven means for reducing costs must be fully utilized, and new approaches (such as improved technologies, risk informed methods for evaluating the safety benefit of design features, and international consensus regarding safety requirements so that standardized designs can be built in several countries without major re-design efforts) should be developed and implemented [8].

3. CRP on natural circulation phenomena, modelling and reliability of passive systems

The use of passive safety systems such as accumulators, condensation and evaporative heat exchangers, and gravity driven safety injection systems eliminate the costs associated with the installation, maintenance and operation of active safety systems that require multiple pumps with independent and redundant electric power supplies. Another motivation for the use of passive safety systems is the potential for enhanced safety through increased safety system reliability. As a result, passive safety systems are being considered for numerous advanced reactor concepts.

The IAEA CRP, entitled “Natural Circulation Phenomena, Modelling and Reliability of Passive Safety Systems that Utilize Natural Circulation”, was started in 2004 to provide international coordination for the work currently underway at the national level in several IAEA Member States. Specific objectives of the CRP are:

- to establish the status of knowledge: passive system initiation & operation; flow stability, 3-D effects and scaling laws
- to investigate phenomena influencing reliability of passive natural circulation systems
- to review experimental databases for the phenomena
- to examine the ability of computer codes to predict natural circulation and related phenomena
- to apply methodologies for examining the reliability of passive systems.

Sixteen institutes have been participated in the CRP: CNEA (Bariloche, Argentina), CEA (France), FZ (Dresden, Germany), BARC (India), Univ. of Pisa (Italy), ENEA (Italy), IVS (Slovakia), JAEA (Japan), KAERI (Rep. of Korea), Gidropress (Russia), University of Valencia (Spain), PSI (Switzerland), Idaho State University (USA), Oregon State University (USA), Purdue University (USA) and European Commission (JRC-Petten, Netherlands).

The IAEA training course on natural circulation phenomena and modeling in water cooled nuclear power plants is one of the outcomes from this CRP (Figure 1). This course provides participants with a comprehensive instruction on natural circulation phenomena and modeling in nuclear power plants. The lecture material was published as an IAEA TECDOC [9]. This course has been held at the International Center of Theoretical Physics (Trieste, Italy) and other locations worldwide almost annually since 2004. Furthermore, this course can be deployed at any new location upon request.

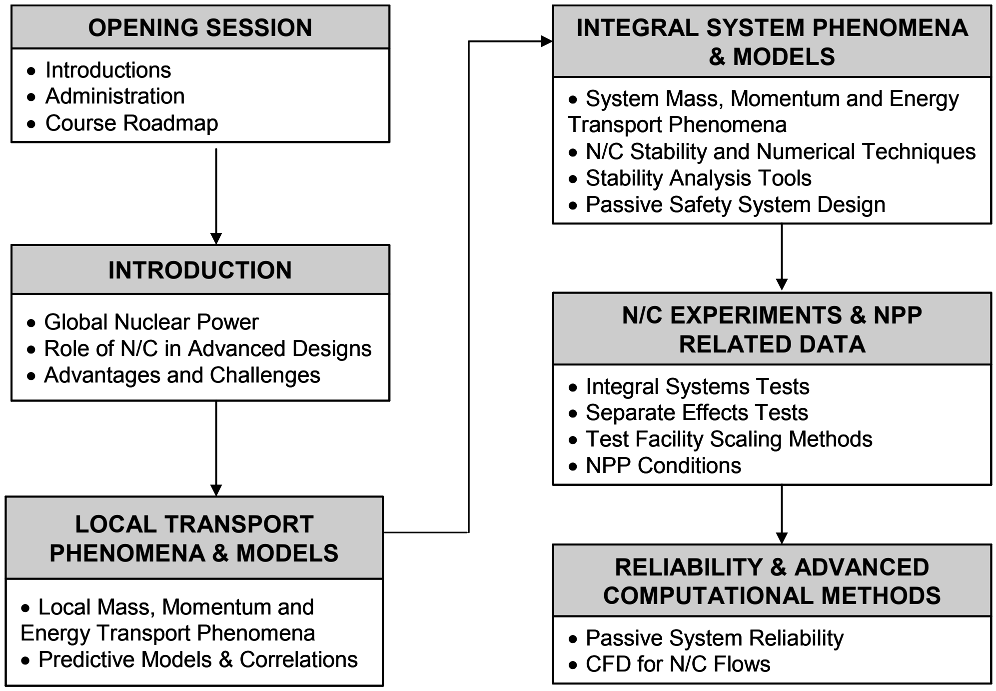


Figure 1 Natural circulation course roadmap

Phenomena influencing natural circulation have been identified and classified into two categories: (a) phenomena occurring during interaction between primary system and containment; and (b) phenomena originated by the presence of new components and systems or special reactor configurations. These phenomena include:

- behaviour in large pools of liquid
- effect of non-condensable gasses on condensation heat transfer
- condensation on the containment structures
- behaviour of containment emergency systems
- thermo-fluid dynamics and pressure drops in various geometrical configurations
- natural circulation in closed loop
- steam-liquid interaction
- gravity driven cooling and accumulator behaviour
- liquid temperature stratification
- behaviour of emergency heat exchangers and isolation condensers
- stratification and mixing of boron
- core make-up tank behaviour.

As shown in Table 1, four categories in different degrees of passivity are defined and used in IAEA [10]. Passive safety systems in Category D are used in many advanced designs and they can be characterized as having active initiation and passive execution. A second outcome of this CRP is a document, currently under publication, that examines passive safety systems adopted by 20 reference designs including evolutionary and innovative concepts to identify the thermo-hydraulic phenomena involved in each passive safety system.

Table 1 Classification of passivity

	Category A	Category B	Category C	Category D
Signal inputs of intelligence	No	No	No	Yes
External power sources or forces	No	No	No	No
Moving mechanical parts	No	No	Yes	Limited
Moving working fluid	No	Yes	Yes	Limited

The third outcome is a TECDOC currently under preparation that will include the improvement in the understanding of each phenomenon, with sample analyses for some integral tests and NPPs, and sample applications of the methodology to examine the passive system reliability.

4. CRP on heat transfer behaviour and thermo-hydraulic code testing for super critical water cooled reactors

There is high interest internationally in both developing and industrialized countries in innovative super-critical water-cooled reactors (SCWRs), primarily because such concepts will achieve high thermal efficiencies (44-45%) and promise improved economic competitiveness utilizing and building on the recent developments for highly efficient super critical fossil power plants. The SCWR has been selected as one of the promising concepts for development by the Generation-IV International Forum.

The higher coolant temperatures proposed for SCWR systems imply fuel cladding temperatures greater than current nuclear reactor operating experience. Because of enhanced heat transfer for supercritical flows and the use of new cladding materials with low corrosion rates, it is necessary to have precise information for establishing both the neutronic and the thermal limits. Consequently, in developing SCWR designs, experimental data for the convective heat transfer from fuel to coolant, covering a range of flow rate, pressure and temperature conditions, are required. Collection, evaluation and assimilation of existing data as well as deployment of new experiments for needed data are necessary to establish accurate techniques for predicting heat transfer in SCWR cores.

Validated thermo-hydraulic codes are required for design and safety analyses of SCWR concepts. Existing codes for water-cooled reactors need to be extended in their application and improved to model phenomena such as pressure drop, critical flow, flow instability behaviour, and transition from super-critical to two-phase conditions.

The IAEA CRP on SCWRs promotes international collaboration among IAEA Member States for the development of SCWRs in the areas of heat transfer behaviour and testing of thermo-hydraulic computer methods. Specific objectives of the CRP are:

- to establish a base of accurate data for heat transfer, pressure drop, blowdown, natural circulation and stability for conditions relevant to super-critical fluids,

- to test analysis methods for SCWR thermo-hydraulic behaviour, and to identify code development needs.

Thirteen institutes have been participated in the CRP: Korea Atomic Energy Research Institute (Korea), University of Wisconsin (USA), China Institute for Atomic Energy (China), Shanghai Jiao Tong University (China), Atomic Energy of Canada, Ltd. (Canada), Bhabha Atomic Research Centre (India), VTT Technical Research Centre (Finland), University of Pisa (Italy), Gidropress (Russia), Institure For Physics and Power Eng. (Russia), Institute for Energy (EC-JRC, Netherlands), OECD-NEA and University of Manchester (UK).

The OECD-NEA has agreed to establish the data base at the NEA, and that the data base will be open to institutes participating in the CRP. There exist many experimental data for heat transfer and pressure drop for supercritical working fluids such as CO_2, Freon, He, CF_2Cl_3, etc. Many institutes are conducting experiments using surrogate fluids or water for supercritical conditions to produce data on heat transfer, pressure drop, critical flow, power-flow instability and natural circulation. All participating institutes committed to supply their experimental data that is already available or will be produced. It is expected that the CRP participants could develop more reliable new correlations based on the experimental data collected. It is planned to define benchmark or standard problems to test thermo-hydraulic codes for SCWR conditions.

5. CRP on establishment of a thermo-physical properties data base of materials for LWRs and HWRs

Improving the technology for nuclear reactors through better computer codes and more accurate material properties data can contribute to improved economics of future plants by helping to remove the need for large design margins, which are currently used to account for limitations of data and methods. Accurate representations of thermo-physical properties under relevant temperature and neutron fluence conditions are necessary for evaluating reactor performance under normal operation and accident conditions.

From 1999 to 2005 the IAEA carried out a CRP on establishment of a thermo-physical properties data base of materials for Light Water Reactor (LWR) and Heavy Water Reactors (HWR). The objective of this CRP was to collect and systematize a thermo-physical properties database for light and heavy water reactor materials under normal operating, transient and accident conditions and to foster the exchange of non-proprietary information on thermo-physical properties of LWR and HWR materials. An internationally available, peer reviewed database of properties at normal and severe accident conditions (THERPRO: http://therpro.hanyang.ac.kr) has been established at Hanyang University (Republic of Korea), and now provides various material properties data and an interactively accessible information resource and communications medium for researchers and engineers.

Institutes participating in this CRP were; Atomic Energy of Canada Ltd. (Canada), Nuclear Power Institute of China (China), University of West Bohemia (Czech Republic), Institute of Physics and Power Engineering (Russia), Institute of High Temperatures of the Russian Academy of Sciences (Russia), Bhabha Atomic Research Centre (India), Commissariat a l'Energie Atomique, Cadarache (France), Hanyang University (Republic of Korea), and Seoul National University (Republic of Korea).

TECDOC-1496 [11] is the output of the CRP and includes:

- thermo-physical properties for nuclear fuel materials, cladding and pressure tube materials, absorber materials and their oxides, structural materials
- thermo-physical properties for coolants (light and heavy water)
- thermo-physical properties for corium under severe accident conditions
- explanation on THERPRO database.

Registering to use freely the THERPRO database is easy by visiting the THERPRO website. In addition, the managers of THERPRO are very interested in continuously enhancing the database, and as such they welcome organizations interested in contributing new data to be added to the database or in participating in the peer review process for new and existing data.

6. CRP on benchmarking severe accident computer codes for HWR applications

Currently different countries follow different regulatory requirements for severe accident considerations in HWRs. It is expected that the new reactor projects will explicitly and systematically consider severe accidents during the design phase to minimize the likelihood of severe core damage and large radioactivity releases.

Computer codes used for the analysis of design basis events have been validated against integral and/or separate effects tests, whereas in the case of severe accident computer codes it is rather impossible, or at least quite expensive, to carryout a validation exercise against integrated experiments. Consequently, the code capabilities have to be assessed based on benchmarking against other severe accident computer codes. In view of this, a benchmarking exercise becomes necessary to assess the results from various computer codes to provide an improved understanding of modelling approaches, strengths and limitations. The exercise could also suggest ways to overcome code limitations and thereby increase the confidence in severe accident code predictions. A benchmarking exercise encompassing the various severe accident codes in use within the HWR community is important not only for providing confidence in the overall performance of the codes but also for the reduction of uncertainties in their predictions.

The IAEA started a CRP in 2009 on benchmarking severe accident computer codes for HWR applications to improve the safety for currently operating plants and to facilitate more economic and safe designs for future plants. The expected outcomes from this CRP are:

- improved understanding of the importance of various phenomena contributing to event timing and consequences of a severe accident,
- improvement of emergency operating procedures or severe accident management strategies,
- advanced information on computer code capabilities to enable the analysis of advanced HWR designs.

Institutes participating in this CRP are: Korea Atomic Energy Research Institute (Rep. of Korea), Shanghai Jiao Tong University (China), Politehnica University of Bucharest (Romania), Atomic

Energy of Canada, Ltd. (Canada), Bhabha Atomic Research Centre (India), and Nuclear Power Corp. of India Ltd. (India).

Planned activities within the CRP include:

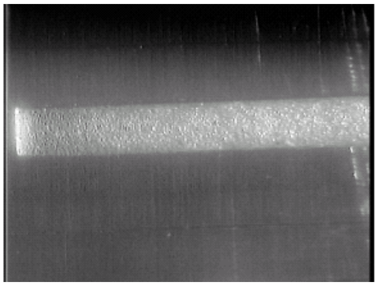
- collection and evaluation of existing models, correlations, experiments, and computer codes applicable to HWR severe accident analysis
- determination of reference design and severe accident scenario for benchmarking analysis considering operating HWRs and available computer codes in Member States
- establishment of criteria for fuel failure, fuel channel failure, fuel channel disassembly, core collapse, calandria vessel failure and containment failure, and reactor vault failure
- benchmark analysis for Phase 1 (accident initiation to fuel channel dryout), Phase 2 (fuel channel dryout to core collapse), Phase 3 (core collapse to calandria vessel failure), and Phase 4 (calandria vessel failure to containment failure)
- benchmark analysis for experiment.

7. CRP on intercomparison of techniques for inspection and diagnostics of HWR pressure tubes

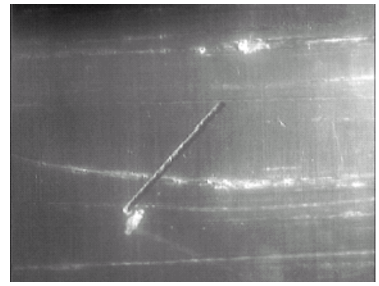
The Pressure Tubes (PTs) of HWRs operate in a high-temperature high-pressure aqueous environment and are subjected to fast neutron irradiation. These conditions lead to the degradation of the PTs with respect to i) dimensional changes (creep and growth), ii) deterioration of mechanical properties (hardening and embrittlement) thereby reducing their flaw tolerance, iii) the growth of existing flaws, which were too small or insignificant at the time of installation, and iv) initiation and growth of new flaws. In order to ensure the PT integrity at all times during their service, they are periodically examined by Non-Destructive Examination (NDE) techniques.

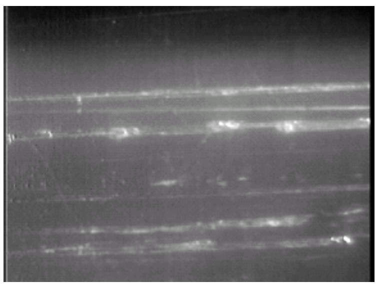
The IAEA conducted a CRP on inter-comparison of techniques for HWR pressure tube inspection and diagnostics. The intent was to identify the most effective pressure tube inspection and diagnostic methods, and to identify further development needs. The CRP was conducted in a round-robin manner. The participating laboratories prepared PT samples containing artificial flaws/blisters/hydrogen resembling real defects of concern. The outside surface of sample was covered to facilitate blind testing. The samples, after examination by participating laboratories, were returned to the originating laboratory, which determined ‘defect truth’ in its sample. The originating laboratory analysed the sample inspection reports from investigating laboratories and compared the defect estimates with their true values (Figure 2). The CRP was conducted in two phases, the first one focused on flaw detection and characterization and the second one dedicated to the ddetermination of hydrogen concentration and blister characterization [12].

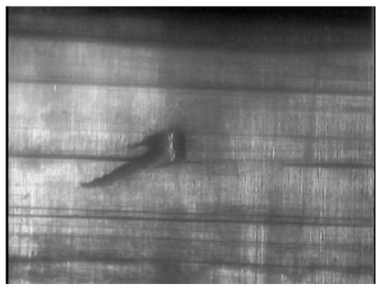
The institutes participated in this CRP were: Comisión Nacional de Energía Atómica (Argentina), Atomic Energy of Canada Ltd. (Canada), Research Institute of Nuclear Power Operations (China), Bhabha Atomic Research Centre (India), Korea Electric Power Research Institute (Republic of Korea), Korea Atomic Energy Research Institute (Republic of Korea), National Institute for Research and Development for Technical Physics (Romania), and Nuclear Non-Destructive Testing Research and Services (Romania).


ID axial bearing pad fret


ID circumferential reference notch


ID oblique 45º


Unintentional ID scratches


Unintentional ID debris fret with lift-off

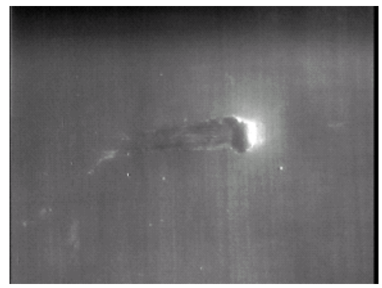

Unintentional ID debris fret with lift-off

Figure 2 Photographs of ID flaws present in the Korean sample

8. International collaborative standard problems

ICSPs provide a structured approach to advance the understanding of neutronic, thermo-hydraulic, fuel or materials behaviour in advanced nuclear power plants, as well as the performance of nuclear plant systems. ICSPs can be established to

- provide a comparison of best-estimate computer code calculations to experimental data under controlled conditions;
- evaluate the capability of computer codes to adequately predict the occurrence of important phenomena, and the corresponding behaviour of nuclear systems during operating, upset and accident conditions, which are represented in experiments.

8.1 ICSP on integral PWR design natural circulation flow stability and thermo-hydraulic coupling of containment and primary system during accidents

A new IAEA ICSP on an integral Pressurized Water Reactor (PWR) design has been prepared as a follow-up to the CRP on natural circulation phenomena, modelling and reliability of passive systems that use natural circulation. Natural circulation flow stability and thermo-hydraulic coupling of primary system and containment during accidents are important phenomena to be examined for integral PWR design. The specific objectives of the ICSP are:

- to compare the best-estimate computer code calculations to the experimental data obtained from the integral test facility representing an integral type reactor

- to improve the understanding of thermal-hydraulic phenomena expected to occur in normal operation and transients in an integral reactor
- to evaluate the capability of computer codes to adequately predict the occurrence of important phenomena, and the corresponding behaviour of nuclear systems during operating, upset and accident conditions, which are represented in experiments.

Oregon State University (OSU) in the USA has offered their experimental facility (Figure 3). The OSU MASLWR test facility models the MASLWR conceptual design including reactor pressure vessel cavity and containment structure. The scope of the ICSP includes two types of experiments: 1) single and two phase natural circulation flow stability tests with stepwise reduction of the primary inventory, and 2) loss of feedwater transient with subsequent ADS (Automatic Depressurization System) blowdown and long term cooling by primary-containment coupling. Participating institutes will perform blind and post-test simulations of the experiments with their own computer codes.

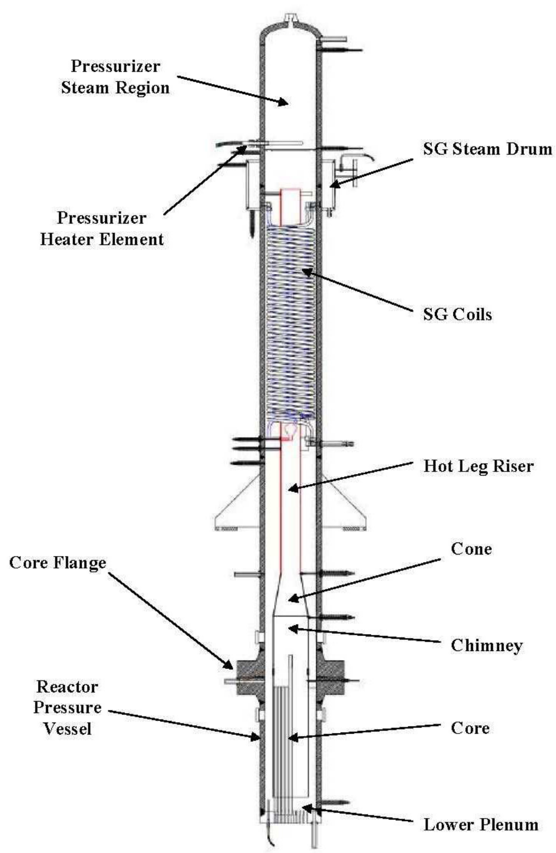


Figure 3 OSU MASLWR test facility: reactor pressure vessel key areas

8.2 ICSP on computer code validation for HWR LBLOCA with RD-14M test

Most internationally recognized codes used for LWR design and safety analysis have been subjected to systematic validation procedures through a number of international programmes. This IAEA ICSP was the first international initiative to compare the performance of codes against experiments for HWR systems.

The reference experiment was performed in the RD-14M test loop located at the AECL Laboratories in Pinawa, Canada. The RD-14M facility is a pressurized water loop with essential features similar to the primary heat transport loop of a typical CANDU 6. A Large Break Loss-of-Coolant Accident (LBLOCA) test, named B9401, was selected as the reference case. This case includes the limited temperature excursion in the core shortly after the LOCA and the demonstration of the performance of the Emergency Core Cooling System (ECCS).

Six different institutes using four different codes and six different idealizations participated in the activity performing the blind and post-test analyses of the B9401 experiment. All codes are two-fluid six-equation codes, except one that is a three-equation code with the drift-flux capability. The strengths and weakness of the codes were identified and the ways to improve the prediction were studied. The participants benefited greatly from the analysis of this experiment due to the exchange of expertise and information that was not available in the open literature [13].

8.3 ICSP on computer code validation for HWR SBLOCA with RD-14M test

Building on the successful completion of the ICSP on HWR LBLOCA, a new IAEA ICSP on a HWR Small Break Loss-of-Coolant Accident (SBLOCA) was started in 2007. The objective of this ICSP is to improve the understanding of important phenomena expected to occur in SBLOCA transients, to evaluate code capabilities to predict these important phenomena, their practicality and efficiency, and to suggest necessary code improvements and/or new experiments to reduce uncertainties. Two RD-14M SBLOCA tests were selected for blind calculations. Eight institutes from six HWR countries are currently participating in this ICSP.

9. Summary and conclusions

Water cooled reactors represent more than 90% of the current worldwide nuclear power plant fleet, and in the near term it is expected that most new nuclear power plants will be evolutionary water cooled reactors. Therefore, high priority should be given to the development of technology to achieve economic competitiveness with other energy sources and to assure high safety levels for water cooled reactors.

As discussed in this paper, the IAEA mission to foster and facilitate technology development in the area of water cooled reactors in Member States is successfully carried out through the organization of coordinated research projects, international collaborative standard problems, technical meetings, international collaborative assessments and training courses and workshops.

10. References

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design Requirements, Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Advanced Light Water Reactor Designs: 2004, IAEA-TECDOC-1391, IAEA, Vienna (2004).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Heavy Water Reactors: Status and Projected Development, TRS-407, IAEA, Vienna (2002).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Small Reactor Designs without On-site Refuelling, IAEA-TECDOC-1536, IAEA, Vienna (2007).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of innovative small and medium sized reactor designs: 2005: Reactors with conventional refuelling schemes, IAEA-TECDOC-1485, IAEA, Vienna (2006).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Advanced Applications of Water Cooled Nuclear Power Plants, IAEA-TECDOC-1584, IAEA, Vienna (2008).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Advanced nuclear plant design options to cope with external events, IAEA-TECDOC-1487, IAEA, Vienna (2006).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Improving Economics and Safety of Water-Cooled Reactors: Proven Means and New Approaches, IAEA-TECDOC-1290, IAEA, Vienna (2002).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Natural circulation in water cooled nuclear power plants: phenomena, models, and methodology for system reliability assessments, IAEA-TECDOC-1474, IAEA, Vienna (2005).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety related terms for advanced nuclear plants, IAEA-TECDOC-626, IAEA, Vienna (1991).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Thermo-physical Properties Database of Materials for Light Water Reactors and Heavy Water Reactors, IAEA-TECDOC-1496, IAEA, Vienna (2006).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Intercomparison of techniques for inspection and diagnostics of heavy water reactor pressure tubes: Flaw detection and characterization, IAEA-TECDOC-1499, IAEA, Vienna (2006).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Intercomparison and validation of computer codes for thermalhydraulic safety analysis of heavy water reactors, IAEA-TECDOC-1395, IAEA, Vienna (2004).