Recent IAEA international cooperation programmes on nuclear reactor thermal-hydraulics and safety

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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 international cooperation programmes related to thermal hydraulics and safety of water cooled reactors, which is the most common type of reactor design at present and will probably still be in the near future.

Keyword: computer code; coordinated research project (CRP); international collaborative standard problem (ICSP); nuclear reactor; safety; simulator; thermal-hydraulics

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 international cooperation programmes in the area of water cooled reactor thermal hydraulics and safety.

2 Coordinated research projects

2.1 Natural circulation phenomena, modeling and reliability of passive safety systems that utilize natural circulation

The use of passive safety systems such as accumulators, condensation and evaporative heat exchangers, and gravity driven safety injection

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systems eliminate the costs associated with the installation, operation and maintenance 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 ongoing 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), Bhabha Atomic Research Centre (India), Univ. of Pisa (Italy), ENEA (Italy), IVS (Slovakia), Japan Atomic Energy Agency (Japan), Korea Atomic Energy Research Institute (Rep. of Korea), Gidropress (Russia), University of Valencia (Spain), Paul Scherrer Institute (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 outputs from this CRP (Fig. 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[2]. This course has been held at the International Center of Theoretical Physics (ICTP, Trieste, Italy) and other locations worldwide almost annually since 2004. The next course will be held at Harbin Engineering University, Harbin, China on 11 - 15 July 2011. Furthermore, this course can be deployed at any new location upon request.

Phenomena influencing the reliability of passive natural circulation systems 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.

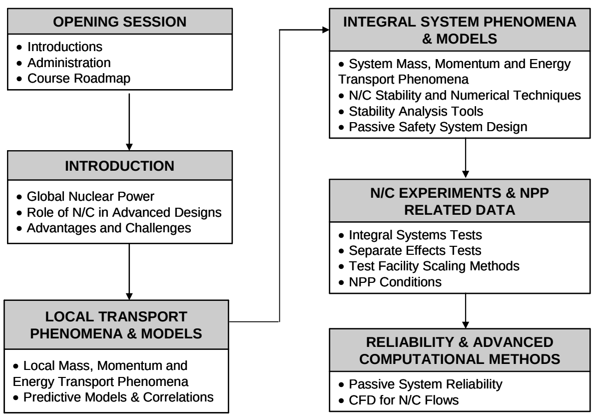

Fig.1 Natural circulation course roadmap.

As shown in Table 1, four categories in different degrees of passivity are defined and used in IAEA[3].

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 output of this CRP is a document, entitled "Passive Safety Systems and Natural Circulation in Water Cooled Nuclear Power Plants"[4], that examines passive safety systems adopted by twenty 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 output is a TECDOC currently under publication process that includes 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.

2.2 Heat transfer behaviour and thermo-hydraulic code testing for supercritical 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. Two benchmark exercises were developed and approved in the 2nd research coordination meeting: 1) steady state flow in a heated pipe: CFD or system codes will be used, and 2) benchmark on stability: system codes or stability models will be used.

2.3 Benchmarking severe accident computer codes for heavy water reactor applications

Currently different countries follow different regulatory requirements for severe accident considerations in heavy water reactors (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 extensively 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 carry out 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.

Table 2 shows the participating institutes and computer codes used in benchmarking analyses.

Table 2 Participating Institutes and computer codes

Institute	Computer code
Atomic Energy of Canada, Ltd. (Canada)	MAAP-CANDU
Bhabha Atomic Research Centre-1 (India)	RELAP5, MELCOOL
Bhabha Atomic Research Centre-2 (India)	RELAP5/SCDAP, ASTEC
Korea Atomic Energy Research Institute (Rep. of Korea)	ISSAC
Nuclear Power Corp. of India Ltd. (India)	ATMIKA.T, CONTACT, SEVAX
Politehnica University of Bucharest (Romania)	RELAP5/SCDAP, COUPLE
Shanghai Jiao Tong University (China)	RELAP5/SCDAP

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.

CANDU 6 was selected as the reference plant for benchmarking analyses since it is the most common HWR design under operation in five countries.
Typical postulated initiating events along with failure of subsequent systems which could lead to severe accident scenarios in HWRs are:
Station blackout
Loss of Coolant Accident (LOCA) with Loss of Emergency Coolant Injection (LOECI)
Steam Generator Tube Rupture (SGTR) (Containment bypass event)
Station blackout is selected as the reference severe accident scenario since:

- Closed form back of the envelope numerical calculations can be made for comparison
- Transients are slow
- Captures most of the severe accident phenomena
- Sequence can be anticipated and tracked: moderator temperature increase, secondary side boiloff, coolant inventory boiloff, primary side liquid relief valve opening, fuel channel heatup, calandria vessel rupture disk opening, core heatup and collapse.

2.4 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.iaea.org) has been established, and now provides various material properties data and an interactively accessible information resource and communications medium for researchers and engineers.

Fig.2 THERPRO website.

Institutes participated 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[5] 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.

3 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 generally 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.

3.1 Integral PWR design natural circulation flow stability and thermo-hydraulic coupling of containment and primary system during accidents

The 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.

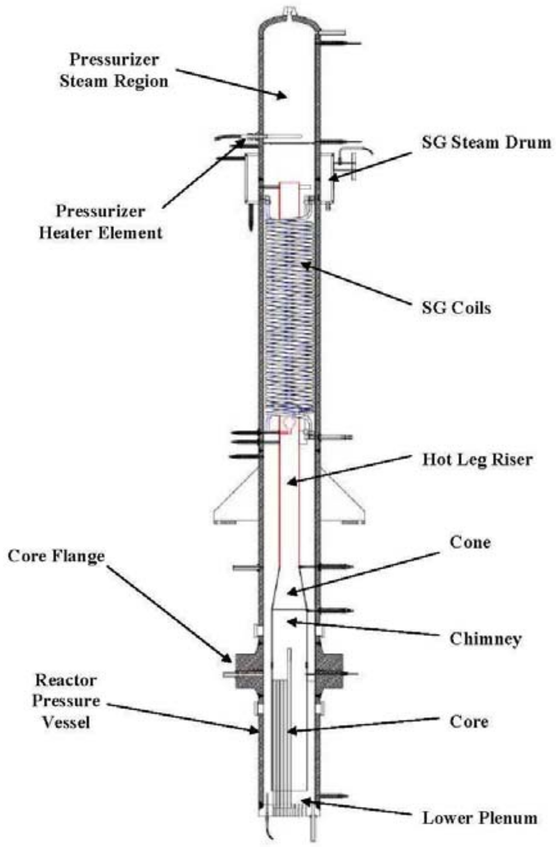

Fig.3 OSU MASLWR test facility: reactor pressure vessel key areas.

Oregon State University (OSU) in the USA has offered their experimental facility (Fig. 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 double-blind, blind and open simulations of the experiments with their own computer codes. The report on MASLWR test facility description[6], ICSP plan[7] and ICSP specification[8] were prepared and distributed to the participants already. The first workshop was held at Oregon State University, USA in March, 2010.

Table 3 shows the participating institutes and computer codes used in the ICSP.

Table 3 Participating Institutes and computer codes

Institute	Computer code
Atomic Energy Regulatory Board (India)	RELAP5/ASTEC
Bhabha Atomic Research Centre (India)	RELAP5/Mod3.2
China Institute for Atomic Energy (China)	RELAP5/Mod3.2/3.3, CFD(CFX/FLUENT)
OKB Gidropress (Russia)	TRAP-97, TRAP-KS, KORSAR/GP
Korea Atomic Energy Research Institute (Rep. of Korea)	TASS/SMR
Korea Institute of Nuclear Safety (Rep. of Korea)	MARS
NuScale (USA)	RELAP5/Mod3.3, N-RELAP5
Oregon State Univ. (USA)	RELAP5-3D
Serco (UK)	RELAP5-3D, TRACE
Shanghai Jiao Tong University (China)	RELAP5/SCDAP, ATHLET
Univ. of Palermo (Italy)	TRACE
Univ. of Pisa (Italy)	RELAP5-3D
US Nuclear Regulatory Commission (USA)	TRACE
Tsinghua Univ. (China)	RELAP5/Mod3.2

3.2 Inter-comparison and validation of computer codes for thermal-hydraulic safety analysis of HWRs

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).

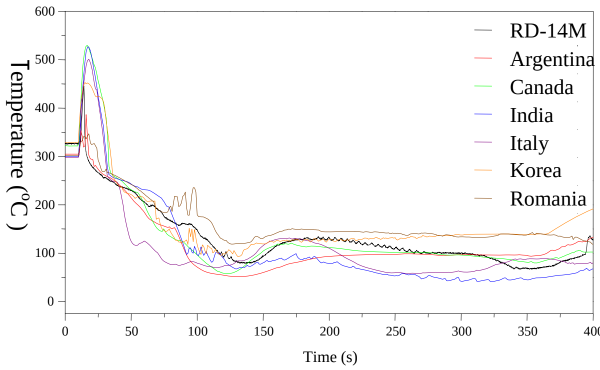

Fig.4 Comparison of results (fuel element temperature).

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[9].

3.3 Comparison of HWR thermal-hydraulic code predictions with SBLOCA experimental data

Building on the successful completion of the ICSP on HWR LBLOCA, the second IAEA ICSP on a HWR Small Break Loss-of-Coolant Accident (SBLOCA) was started in 2007. The objectives of this ICSP are:

- 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 (Table 4).

Table 4 Participating Institutes and computer codes

Institute	Computer code
Atomic Energy of Canada, Ltd. (Canada)	CATHENA
Atomic Energy Regulatory Board (India)	RELAP5
CNEA (Argentina)	CATHENA
CNE-PROD (Romania)	CATHENA
Korea Atomic Energy Research Institute (Rep. of Korea)	CATHENA
Korea Institute of Nuclear Safety (Rep. of Korea)	MARS
Nuclear Power Corp. of India Ltd. (India)	ATMIKA
Tsinghua University (China)	CATHENA

4 Nuclear power plant simulators

Simulator is a very useful tool to understand the thermal hydraulic behaviour of nuclear power plant for normal operation as well as accidents. The IAEA has established a programme in nuclear reactor simulation computer programs to assist Member States in education and training.

The IAEA simulator collection currently includes the following seven simulators:

- A VVER-1000 simulator provided to the IAEA by the Moscow Engineering and Physics Institute in Russia.
- The IAEA generic Pressurized Water Reactor (PWR) simulator has been developed by Micro-Simulation Technology of USA using the PCTRAN software. This simulator is a 600 MWe generic two-loop PWR with inverted U-tube steam generators and dry containment system that could be a Westinghouse, Framatome or KWU design.
- The IAEA advanced PWR simulator has been developed by Cassiopeia Technologies Inc. (CTI) of Canada, and is largely based on a 600 MWe PWR design with passive safety systems, similar to the Westinghouse AP-600.
- The IAEA generic Boiling Water Reactor (BWR) simulator has also been developed by CTI and represents a typical 1300 MWe BWR with internal recirculation pumps and fine motion control rod drives. This simulator underwent a major enhancement effort in 2008 when a containment model based on the ABWR was added.
- The IAEA Pressurized Heavy Water Reactor (PHWR) simulator is also a CTI product and is largely based on the 900 MWe CANDU-9 system.
- The IAEA advanced PHWR simulator by CTI from Canada, which represents the ACR-700 system.
- The IAEA advanced BWR, which largely represents the GE ESBWR design and was also created by CTI.

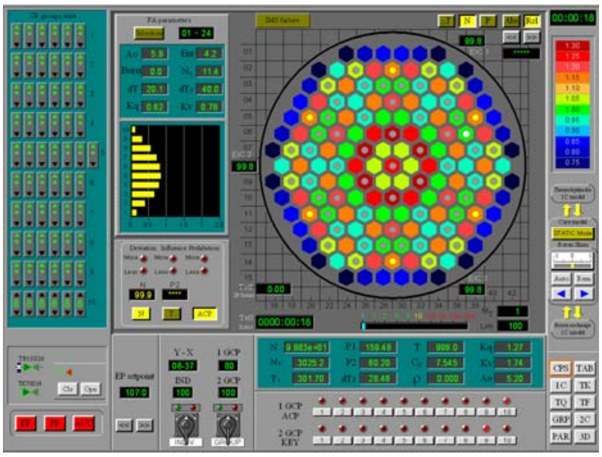

Fig.5 IAEA NPP simulator.

These simulators, including the associated documentation, are distributed at no cost to interested parties in IAEA Member States. Furthermore, the IAEA sponsors training courses and workshops on a regular basis. Since 1997, fourteen workshops have been held, in Egypt, Saudi Arabia, the Republic of Korea, Italy, USA and at the IAEA Headquarters in Vienna, Austria. The workshops are currently a biennial activity sponsored by the ICTP. The most recent workshop took place from 12 October to 23 October 2009 at the ICTP in Trieste, Italy.

5 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 being carried out successfully through the organization of many international cooperation programmes including coordinated research projects and international collaborative standard problems.

Acknowledgement

The IAEA expresses its appreciation to all experts participated in the CRPs and ICSPs.

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