THERMAL HYDRAULICS DIVISION NEWSLETTER

Spring 2004

Message from the Chair

I want to report to you that the division is in a good state with financially strong and technically expanding entity. THD sponsored NURETH-10 (Korea, 2003) with Korean Nuclear Society. The meeting was a great success with 317 contributed and invited papers presented and 342 registered attendees from 31 countries. The KNS with NURETH-10 experience pledges to support the NURETH-11 (France, 2005) organizing committee for the preparation of the conference. THD is a cosponsor for NUTHOS-6 (Nara, Japan) which will be held October 4th – 8th, 2004 at Nara-Ken New Public Hall. The conference location is one of many historical tour places in Japan. THD had 5 technical sessions at 2003 ANS Winter Meeting which was held at Hyatt Regency Hotel, New Orleans, Louisiana from November 16th to 20th, 2003. We will have 5 technical sessions at 2004 ANS Annual Meeting which will be held at Omni William Penn Hotel from June 13th to 17th, 2004. The Annual Meeting will be celebrating the 50th anniversary of ANS. THD agreed with the headquarters of ANS that THD will not sponsor NTHC if NTHC would be held within a month period of ANS. 2004 NHTC (July 11-15, 2004) falls within ANS's one month meeting restriction period.

As a reminder, please visit the THD website for the most current information pertaining to our division's activities. You can find it at http://thd.ans.org.

Whee Choe
Chairperson (2003-2004)
Thermal-Hydraulics Division

2003 ANS THD Awards

Professor Soon Heung Chang received the 2003 Technical Achievement Award at the award ceremony during NURETH-10 Conference in Seoul, October 5-9, 2003. Professor Chang gave the award lecture on **"Understanding, Predicting, and Enhancing the Critical Heat Flux (CHF)".**

The award was presented to him in recognition of his sustained exceptional contributions to the studies of critical heat flux and other unique scientific contributions to the understanding of thermal hydraulic phenomena of significance to nuclear plant operation and safety, and for his impact on the thermal hydraulics community as a researcher, educator, and leader promoting technical excellence and international scientific exchanges.

Soon Heung Chang is currently a professor in Nuclear and Quantum Engineering Department and Dean of Academic Affairs at the Korea Advanced Institute of Science and Technology. A graduate of Seoul National University, he received his Ph.D. in nuclear engineering from MIT and was a staff member at Bechtel Power Corporation and a visiting scientist at Chalk River Nuclear Laboratory. He has served on the International Nuclear Safety Advisory Group of IAEA and the Korea Presidential Advisory Committee on Science and Technology. He was also a Commissioner of the Korea Nuclear Safety Commission and the Chairman of the Advisory Committee on Nuclear Safety of Korea. Professor Chang was the Technical Program Chair of NURETH-10 and is Technical Program Co-Chair for the forthcoming NUTHOS-6 in Nara, Japan, October 4-8, 2003. He is an international authority on nuclear safety and thermal hydraulics testing and analysis, and his technical and scientific contributions in critical heat flux are widely recognized.

THD Membership

Our division membership continues to sustain strong growth. For the 6th straight year, THD member has increased. This year we are happy to report a total of 839 members. This is an increase of 34 new members over last year, the biggest one year increase in some time!

Year	Number
1999	753
2000	770
2001	782
2002	805
2003	839

THD welcomes anyone who is interested in thermal-hydraulics and related areas to join the Division and participate in the Division activities. The Division activities include paper review, paper presentations, organizing and chairing technical sessions, sponsoring topical meetings, recognizing worthy candidates for honors and awards, overseeing and participating in all aspects of meetings sponsored by the Division, and supporting student conferences. If you are interested in becoming a new member of our Division or if you would like to participate in any of our activities, please contact any of the Division Officiers.

Report on the THD Program Committee

Martin Bertodano, Purdue University
Email: bertodan@purdue.edu.

The highlight of the year was NURETH-10 held October 5-11 in Seoul, Korea. The meeting was well attended with 4 plenary lectures, 13 Keynote Lectures and 267 papers. The work of Dr. Jong Kim and Dr. Won-Pil Baek and countless others who helped organize the sessions and review the papers made this a memorable occasion.

The ANS Winter Meeting in New Orleans was also well attended with 40 summaries, which is an improvement over previous years. I would like to thank the members of the Program Committee who continually help with the review of these summaries.

For the Annual Meeting in Pittsburgh there are only 17 summaries. As usual, there are fewer summaries in the summer. However, this summer there is an ICAPP embedded topical with its own thermal-hydraulics sessions so the meeting promises to be interesting.

The next big Thermal-Hydraulics meeting is NUTHOS-6 in Nara, Japan on October 4-8, 2004. The 6th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation, and Safety, NUTHOS-6 at www.nuthos6.org , is expected to be the largest TH meeting to be held this year. Since 1982, the NUTHOS meeting series has served the international nuclear societies as an open forum where high-quality and state-of-the-art technical information is activity discussed and exchanged among world-class experts. This years meeting is hosted by the Atomic Energy Society of Japan in Nara, Japan, the ancient capital of Japan from 710 to 784 A.D. It is one of the cities in Japan that can provide a perfect academic environment for international conferences and offers an ideal setting for tourists.

Looking into the future, NURETH-11 will be held in Avignon, France, on October 2-6, 2005. The deadline to submit an abstract is May 1, 2004 and more information may be found at the website www.nureth11.com. Dr. Herve Lemonnier has been working diligently with the THD and the planning is on schedule. This promises to be an excellent meeting in beautiful southern France.

The Call for Papers for the 2004 ANS Winter Meeting in Washington, D.C. is already out. The summaries are due on June 11. It will be held November 14-18. This year's meeting includes the BE 2004 embedded topical, which THD co-sponsors. For more information visit the ANS website www.ans.org/meetings.

The technical performance of the Division is improving. Our participation at the National Meetings is one of the highest in terms of the number of summaries. The NURETH meetings continue to be the flagship of the technical program and are presently maintaining their historical level of participation. The next NURETH meeting will be held in the US in 2007 or 2008. At this point we are looking for sponsors, so if any local section is interested we would like to know it.

If you have any comments or suggestions please email me at bertodan@purdue.edu.

Newsletter Feature: Advanced Reactor Activities

Robert Martin, 2004 THD secretary

Every year the THD newsletter includes articles of current thermal-hydraulic research and development activities. Industry-wide interest in new reactor designs has served to expand the scope of existing R&D programs. In this year's newsletter, articles are featured summarizing some of these activities.

Status of New Reactor Licensing Activities in the U.S.

Sandra Sloan, Manager of New Reactor Licensing, AREVA
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The NRC is currently reviewing several new reactor designs. These include the Westinghouse AP1000, a 1000-MWe advanced LWR; the General Electric ESBWR, a 1390 MWe advanced BWR; the AECL ACR-700 (a 700 MWe advanced CANDU which is light-water-cooled and heavy-water-moderated); the AREVA SWR 1000, a 1250 MWe advanced BWR; and the Westinghouse IRIS (a 325 MWe advanced LWR in which the entire RCS, including the core, reactor coolant pumps, and steam generators, is contained in a single pressure vessel). Other designs which have previously been submitted for NRC review but are currently inactive include the General Atomics GT-MHR (a 300 MWt helium-cooled reactor) and the Pebble Bed Modular Reactor (PBMR).

Three utilities have submitted applications to the NRC for an Early Site Permit, which if approved would allow the utility to "bank" a site for future reference in a license application. Dominion has submitted an application for its North Anna site. Exelon has submitted an application for its Clinton site. Entergy has submitted an application for its Grand Gulf site. Early public meetings have already been held regarding the Dominion and Exelon applications. Also, notice has been given that the Atomic Safety and Licensing Board will be convened to begin the review and hearing process for both of these applications.

The U. S. Department of Energy has issued an RFP for utility-led consortiums who are seeking to apply for a combined license (COL) for a new reactor. The COL is a combined construction permit and operating license. No proposals have yet been submitted, but the deadline isn't until the end of 2004.

Generation IV Research Activities at the INEEL

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As the Department of Energy's (DOE's) lead laboratories for nuclear technology development, the INEEL and ANL-West have been organizing and coordinating the Generation IV Initiative, which will develop nuclear technologies that are safe, clean, the most cost-effective and proliferation-resistant.

The Generation IV systems selected in 2002 by ten Generation IV International Forum (GIF) countries are the Very-High

Temperature Reactor (VHTR), Supercritical-Water-Cooled Reactor (SCWR), Sodium-Cooled Fast Reactor (SFR) Lead-Cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), and Gas-Cooled Fast Reactor (GFR).

The VHTGRs are those concepts that have average coolant temperatures above 900^0C or operational fuel temperatures above 1250^0C. These concepts provide the potential for increased energy conversion efficiency and for high-temperature process heat application addition to power generation. While all the High Temperature Gas Cooled Reactor (HTGR) concepts have sufficiently high temperatures to support process heat applications, such as desalination and cogeneration, the VHTGR's higher temperatures are suitable for particular applications such as thermochemical hydrogen production (Fig. 1). However, the high temperature operation can be detrimental to safety following a loss-of-coolant accident (LOCA) initiated by pipe breaks caused by seismic or other events. Following the loss of coolant through the break and coolant depressurization, air from the containment will enter the core by molecular diffusion and ultimately by natural convection, leading to oxidation of the in-core graphite structures and fuel. The oxidation will release heat and accelerate the heatup of the reactor core.

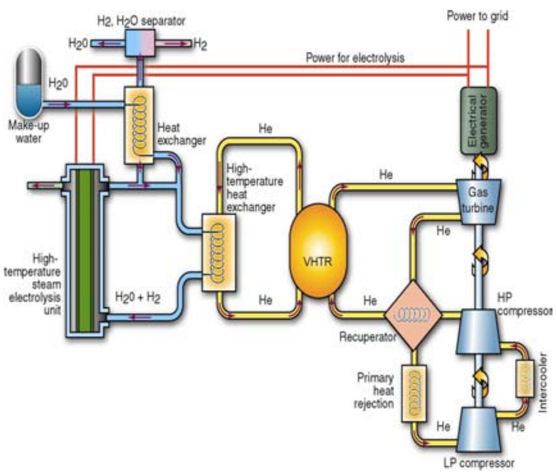

Figure 1. VHTR Design

Thus, without any effective countermeasures, a pipe break may lead to significant fuel damage and fission product release. The Idaho National Engineering and Environmental Laboratory (INEEL) has investigated this event for the past three years for the HTGR. However, the computer codes used, and in fact none of the world's computer codes, have been sufficiently developed and validated to reliably predict this event. New code development, improvement of the existing codes, and experimental validation are imperative to narrow the uncertainty in the predictions of this type of accident.

As part of DOE's NERI and INERI project, a research team led by Chang Oh is working on thermal hydraulic-related tasks on VHTR that include the computer code development, validation, material testing, Brayton cycle efficiency improvement.

SCWR research at INEEL is led by Jacopo Buongiorno. SCWRs are promising advanced nuclear systems because of their high thermal efficiency (i.e., about 45% vs. about 35% efficiency for advanced Light Water Reactors, LWRs) and considerable plant simplification. SCWRs are basically LWRs operating at higher pressure and temperatures with a direct once-through cycle. Operation above the critical pressure eliminates coolant boiling, so the coolant remains single-phase throughout the system. Thus the need for recirculation and jet pumps, pressurizer, steam generators, steam separators and dryers is eliminated (Fig. 2). The main mission of the SCWR is generation of low-cost electricity. It is built upon two proven technologies, LWRs, which are the most commonly deployed power generating reactors in the world, and supercritical fossil-fired boilers, a large number of which is also in use around the world. The SCWR concept is being investigated by 32 organizations (including national laboratories, universities and industry) in 13 countries (including Canada, Japan, Germany, Korea and the U.S. in leading roles). In the U.S. most funding comes from the Generation-IV program, as well as four NERI and two I-NERI projects. The Generation-IV SCWR program is led by INEEL and operates under the following general assumptions, which are consistent with the SCWR's focus on electricity generation at low capital and operating costs: Direct cycle, Thermal spectrum, Light-water coolant and moderator, Low-enriched uranium oxide fuel, and Base load operation.

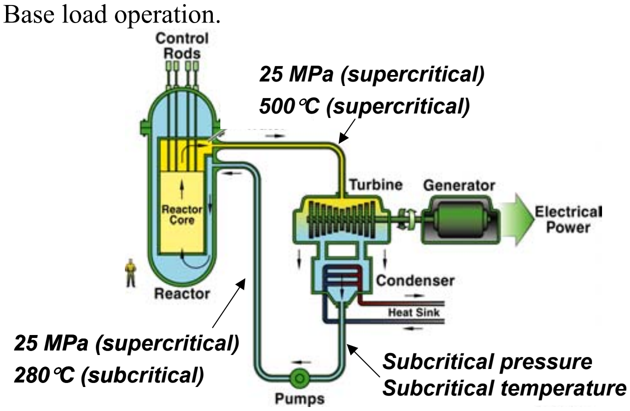

Figure 2 SCWR Design

The key challenges on the path to demonstration of the SCWR viability were identified in the Generation-IV Roadmap Report, and include: Development of materials for in-core service, Design of the safety systems and containment to achieve adequate level of safety, Demonstration of dynamic stability, Development of a credible mechanical, thermal and neutronic design for the fuel assembly, Confirmation of the economic potential of the concept.

The DOE Advanced Fuel Cycle (AFC) Program seeks to develop and demonstrate technologies needed to transmute the long-lived transuranic actinide isotopes contained in spent nuclear fuels into shorter-lived fission products and to close the fuel cycle with increased efficiency utilizing a high temperature gas-cooled reactor. To support this effort, initial scoping-level irradiation tests on a series of fertile and non-fertile fuel forms (i.e. uranium or no uranium and potential materials for use in advanced fuel forms are being irradiated in the Advanced Test Reactor (ATR). The irradiation portion of the project led by Richard Ambrosek at INEEL includes a coordinated effort with ANL-W and Los Alamos National Laboratory (LANL).

Four fuel forms are under consideration for testing by the AFC Program: nitride, metallic, oxide and dispersion fuel forms. Of the four fuel forms, the metallic and nitride forms are the leading candidates and are being tested first. The fuel forms to be tested are four drop-in capsule experiments (hereinafter just called experiments), AFC-1B, AFC-1E and AFC-1F (low burnup) and AFC-1D (high burnup). Burnup is defined to be percent depletion of initial Pu-239 atoms (and U-235 atoms if present), with the nominal low burnup to be at least 5% to 7% for AFC-1B, and 5% to 8% for AFC-1 Æ and 1F, but no greater than 10%, and the nominal high burnup for AFC-1D to be at least 20% but no greater than 25%.

In addition to the four fuel forms, the high temperature gas-cooled fast reactor (GFR) is receiving consideration within the DOE Advanced Fuel Cycle Initiative (AFCI) and Generation IV reactor program.

This experiment (GFR-F1) will test fuel related refractory ceramics, nickel based 800H and MA754 (ODS) alloys, and iron based T122 and MA957 (ODS) ferritic alloys. The GFR-F1 test is a non-fueled experiment and the material candidates are to be irradiated in a drop-in capsule. The irradiation test series will utilize the drop-in positions in the ATR's East Flux Trap (EFT).

Each AFC-1-series experiment's design configuration is a stack of six rodlets within a single, 52-inch long stainless steel outer experiment capsule. Each experiment capsule is approximately 52inches long by 0.354 inches in diameter and weighs approximately 1.1 pounds.

Each capsule is placed in an AFC basket (with an AFC aluminum spacer inserted below the capsule). The AFC basket design functions to both hold the experiment in one of the seven irradiation positions in the EFT as well as filter thermal neutrons to reduce rodlet power. The basket design uses aluminum for the structure and a thin cadmium shroud for the thermal neutron filter.

Advanced Computational Thermal Fluid Physics and its Assessment for Supercritical Reactors

Donald M. McEligot, INEEL, and Jung Yul Yoo, Seoul National University
Email: dm6@inel.gov

The ultimate goal of this *Korean / US / laboratory / university collaboration* of coupled fundamental computational and experimental studies is the improvement of predictive methods for Generation IV reactor systems (e.g., supercritical water reactors). The specific objectives are to develop and to extend direct numerical simulation (DNS), large eddy simulation (LES) and differential second moment closure (DSM) techniques to treat supercritical property variation and complex geometries, thereby providing capabilities to

- assess predictive capabilities of current codes
- provide bases to improve safety and subchannel codes
- provide computational capabilities where current codes and computations are inadequate
- give predictions for Generation IV conceptual and preliminary designs and
- ultimately, handle detailed flow problems for final Generation IV designs.

This basic thermal fluids research applies first principles approaches (DNS and LES) coupled with experimentation (heat transfer and fluid mechanics measurements). Turbulence is one of the most important unresolved problems in engineering and science, particularly for the complex geometries and fluid property variation occurring in these advanced reactor systems and their safety systems. DNS, LES and differential second moment closures (DSM or Reynolds-stress models) are *advanced computational concepts* in turbulence "modeling" whose development is being *extended to treat complex geometries and severe property variation* for designs and safety analyses of supercritical-pressure reactors (SCRs).

Variations of fluid properties along and across heated flows are important in Supercritical-pressure Water Reactors (SCWRs), Very High Temperature gas-cooled Reactors (VHTRs) and Gas-cooled Fast-spectrum Reactors (GFRs), all Generation IV reactor systems concepts. Significant differences and uncertainties have been found between thermal hydraulic correlations for these conditions. Improved computational techniques and supporting measurements are needed to assist the developers of codes for reactor design and system safety analyses to treat the property variations and their effects reliably for some operating and hypothesized accident scenarios of these reactors. The geometries of the reactor cooling channels of some SCWR concepts are demonstrated in Figure 1. Most of these geometries are more complex than those that have been used to generate the empirical correlations employed in current thermal hydraulic codes. Advanced computational techniques may be applied but measurements with realistic geometries are needed to assess the reliability and accuracy of their predictions.

Prof. R. H. Pletcher (Iowa State) is extending LES to generic idealizations of such geometries with property variation; *Prof. J. Y. Yoo (SNU)* supports these studies with DNS. *Prof. S. O. Park (KAIST)* is developing DSM models and is evaluating the suitability of other proposed RANS (Reynolds-averaged Navier-Stokes) models by application of the DNS, LES and experimental results. *INEEL* is obtaining fundamental turbulence and velocity data for generic idealizations of the complex geometries of these advanced reactor systems. *Profs. J. S. Lee, S. T. Ro and J. Y. Yoo (SNU)* are developing experiments on the effects of property variation on turbulence structure in superheated and supercritical flows. *Profs. J. M. Wallace and P. Vukoslavcevic (U. Maryland)* are developing miniaturized multi-sensor probes to measure turbulence components in the supercritical flows of the SNU experiments. *Profs. L. E. Hochreiter (Pennsylvania State) and J. D. Jackson (U. Manchester)* provide industrial insight and thermal-hydraulic data needs and review the results of the studies for application to realistic designs and their predictive safety and design codes.

DNS employs no turbulence modeling; it solves the unsteady governing equations directly. Consequently, along with measurements, it can serve as a benchmark for assessing the capabilities of LES, DSM and general RANS techniques. It also can be applied now for predictions of heat transfer at low flow rates in reduced power operations and transient safety scenarios, such as loss-of-coolant or loss-of-flow accidents, in SCWRs, GFRs and VHTRs. Figure 2 indicates that for SCWRs it can handle sensitive situations which are difficult to treat properly with correlations or with many turbulence models. Once validated, LES and DSM techniques can be applied for predictions at higher flow rates, such as near normal full-power operating conditions, for these Generation IV reactor concepts. The flow facility developed at SNU provides means of measuring heat transfer to supercritical fluids for assessment of the effects of their property variations and the miniaturized multi-sensor probes from U. Maryland will permit measuring the turbulence which is modeled by the codes. The INEEL experiment models the complex geometry of coolant passages in an SCWR concept to provide benchmark data.

INEEL has developed the World's largest Matched-Index-of-Refraction flow system (http://www.inel.gov/env-energyscience/physics/mir/). By using optical techniques, such as laser Doppler velocimetry (LDV), measurements can be obtained in small complex passages without disturbing the flow. The refractive indices of the fluid and the model are matched so that there is no optical distortion. The large size provides good spatial and temporal resolution. This facility provides means to investigate the complex flow features of Generation IV reactor geometries.

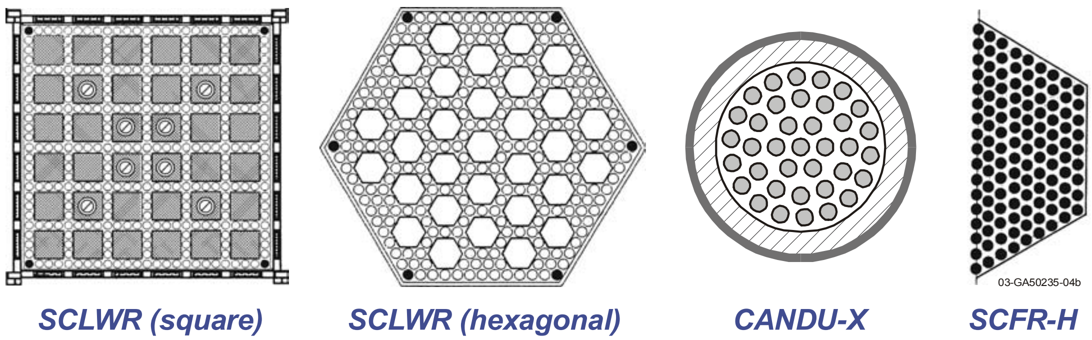

Fig. 1. Proposed designs for fuel assemblies in some SWR concepts.

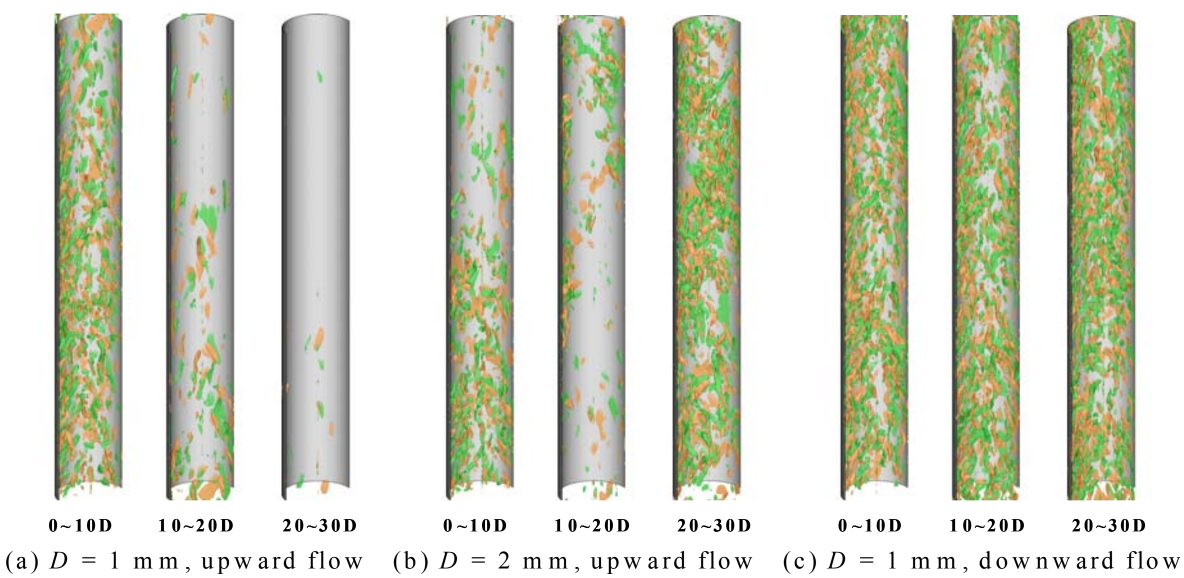

Fig. 2. Turbulence sensitivity to fluid property and buoyancy in DNS of heat transfer to supercritical flow.

In-Vessel Retention Strategies for High Power Reactors

J. L. Rempe, K.G. Condie, and D. L. Knudson, INEEL; K. Y Suh, Seoul National University; F.B. Cheung; Penn State, and S. B. Kim, KAERI

In-vessel retention (IVR) of core melt is a key severe accident management strategy adopted by some operating nuclear power plants and proposed for some advanced light water reactors (ALWRs). If there were inadequate cooling during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of the reactor vessel, as happened in the Three Mile Island Unit 2 (TMI-2) accident. If it is possible to ensure that the lower head remains intact so that relocated core materials are retained within the vessel, the enhanced safety associated with these plants can reduce concerns about containment failure and associated risk. However, it is not clear that currently proposed External Reactor Vessel Cooling (ERVC) without additional enhancements could provide sufficient heat removal for higher-power reactors (up to approximately1500 MWe). Hence, a three-year, U.S. - Korean International Nuclear Energy Research Initiative (INERI) project has been initiated in which INEEL, SNU, PSU, and KAERI are exploring options, such as enhanced ERVC performance and the use of internal core catchers, that have the potential to ensure that IVR is feasible for high power reactors.

The ultimate objective of this project is to develop specific recommendations to improve the safety margin for success of IVR in high power reactors. State-of-the-art analytical tools and key U.S. and Korean experimental facilities are used to evaluate options that could increase the margins associated with the proposed modifications as depicted in Figure 1. This increased margin has the potential to improve plant economics (owing to reduced regulatory requirements) and increase public acceptance (owing to reduced plant risk). Although this program is focusing on the Korean Advanced Power Reactor - 1400 MWe (APR1400) design, recommendations will be developed so that they can easily be applied to a wide range of existing and advanced reactor designs. Selected accomplishments during the first two years of this INERI are highlighted below.

In-Vessel Core Catcher Investigations

A preliminary in-vessel core catcher design was developed that consists of several interlocking sections (see Figure 2). The use of multiple sections reduces cost, and simplifies manufacture and installation. Scoping thermal, flow, structural, and materials assessments were completed to provide insights about the configuration, materials, and thickness for an in-vessel core catcher located approximately 5 mm above the lower head. High temperature materials interaction tests were made using samples of proposed core catcher materials with plasma-sprayed coatings. Results suggest that a stainless steel based material with a 500 μm zirconium dioxide coating over a 100 μm Inconel 718 bond coating will perform better at the high temperature, oxidizing conditions expected during a severe accident. High temperature materials interaction tests to investigate the performance of proposed core catcher materials are underway at INEEL's high temperature test laboratory (HTTL) facility using prototypic materials expected to relocate during a severe accident (e.g, UO_2, ZrO_2).

Four in-vessel core catcher tests were completed in the KAERI LAVA-GAP facility. The LAVA-GAP tests, which use Al_2O_3 to simulate materials relocating from the core, provide insights about the impact of an in-vessel core catcher and candidate coatings on vessel thermal response. Post-test examinations and test instrumentation show that the presence of an insulator coating significantly reduces the thermal loads and attack from relocating materials.

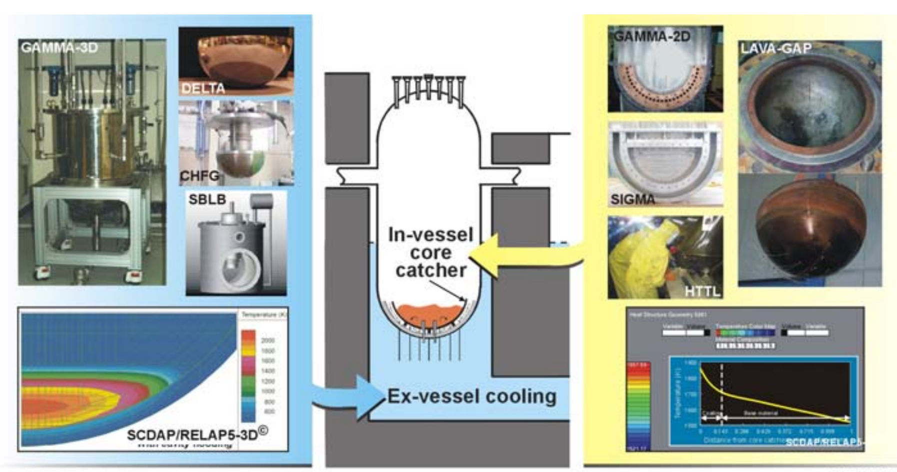

Figure 1. Key U.S. and Korean experimental facilities and state-of-the-art analytical tools are applied to investigate options to enhance ERVC and core catcher performance

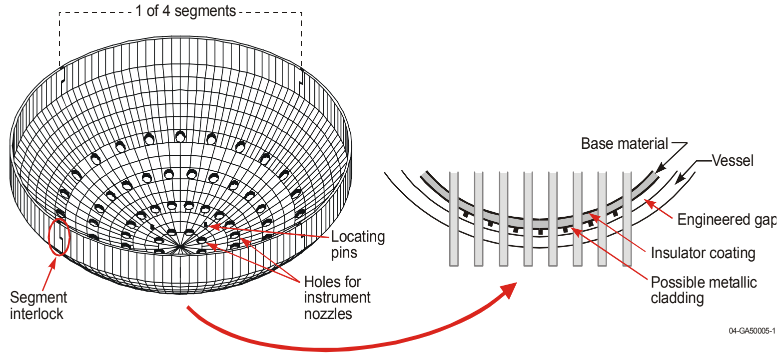

Figure 2. Proposed core catcher design

SNU is leading efforts to develop a narrow gap boiling curve for quantifying heat removal in the engineered space between the in-vessel core catcher and the reactor vessel. To address differences observed in experimental data from one and two -dimensional test facilities, SNU is obtaining data from both types of facilities. Data from these facilities provide new insights about multi-dimensional heat transfer coefficient behavior and allow SNU to develop a narrow gap boiling curve.

External Reactor Vessel Cooling Investigations

ERVC experiments in the U.S. are performed in the PSU SBLB test facility using an improved vessel/insulation design. Two separate types of tests are currently being conducted, one with and the other without vessel coating. For the case without vessel coating, the main focus is on the performance of the improved vessel/insulation design. For the case with vessel coating, the main focus is on the enhancement of the CHF due to vessel coating alone. In both types of experiments, the CHF phenomena including the vapor dynamics and the vapor generation cycle on the vessel outer surface are studied along with the upward co-current two-phase flow induced in the annular channel by the boiling process. In the Republic of Korea, two separate series of boiling tests are being performed using the DELTA and the GAMMA 3D apparatus. The nucleate and film boiling heat transfer coefficients for the various downward facing surfaces are being measured in the DELTA tests. The results of this series of tests are being compared with previous test data and various nucleate and film boiling heat transfer coefficients obtained from numerical studies. In the U.S., transient quenching and steady state boiling experiments are underway in PSU's SBLB facility. Results demonstrate that micro-porous aluminum coatings applied to the exterior surface of a vessel could significantly enhance the CHF for boiling on downward facing curved surfaces (varying from 40 to more than 100% higher). Similar results were obtained for vessels coated with copper micro-porous coatings. However, micro-porous copper coatings were found to be much less durable and tended to degrade after several boiling cycles.

An improved vessel/insulation arrangement design has been developed and installed in the SBLB. Initial test results indicated that the improved vessel/insulation design not only facilitated the steam venting process but also enhanced the downward facing boiling heat transfer on the vessel outer surface.

Results of the ANS THD Elections

The results of the recent ANS elections are in. We have the following new Division Officers and Executive Committee members.

Division Chair: Yassin A. Hassan
Email: Hassan@cedar.ne.tamu.edu

Vice Chair/Chair Elect: Robert Martin
Email: Robert.Martin@framatome-anp.com

Secretary: Jovica R. Riznic
Email: riznic@cnsc-ccsn.gc.ca

Treasurer: Shripad T. Revankar
Email: revankar@ecn.purdue.edu

Executive Committee (3 year term):
Chang H. Oh (2007)
Karen M. Vierow (2007)

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Vice Chair/Chair-Elect: Yassin A. Hassan

Secretary: Robert Martin

Treasurer: Joy Rempe

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Mamoru Ishii (2004)
Yassin Hassan (2004)
Joy Rempe (2005)
Per Peterson (2005)
Whee G. Choe (2005)
David Bessette (2006)
Martin Bertodano (2006)
Cetin Unal (2006)

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