THERMAL HYDRAULICS DIVISION NEWSLETTER

Spring 2005

Message from the Chair

It is my pleasure to report you that the division is in a good state , a financially strong and technically expanding entity. Our division membership has shown continued growth the past several years. THD sponsored NUTHOS-6 (Japan, 2004) with Atomic Energy Society of Japan. The meeting was a great success with 272 technical papers, 6 plenary sessions, and 18 keynote papers presented and 326 registered attendees from 28 countries. After the successful NURETH-10 meeting held in Seoul Korea by KNS in 2003, the NURETH-11 is geared for another successful meeting in France, 2005. NURETH-11 will be held October 2nd to 6th, at Popes' Palace in Avignon in France. The Conference Centre of the Popes' Palace in Avignon is a unique and historical place which hosted Catholic Church Popes in the 14th century for almost a hundred years. THD had 4 technical sessions at the 2004 ANS Winter Meeting which was held at Omni Shoreham Hotel in Washington, D.C. from November 14–18, 2004. We will have 5 technical sessions at the 2005 ANS Annual Meeting which will be held at Town and Country Resort & Convention Center in San Diego, California, from June 5th to 9th 2005. The division is planning 5 technical sessions at the ANS Winter Meeting and Nuclear Technology Expo to be held at Omni Shoreham Hotel in Washington, D.C. from November 13–17, 2005.

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.

Yassin Hassan
Chairperson (2004-2005)
Thermal-Hydraulics Division

2004 ANS THD Awards

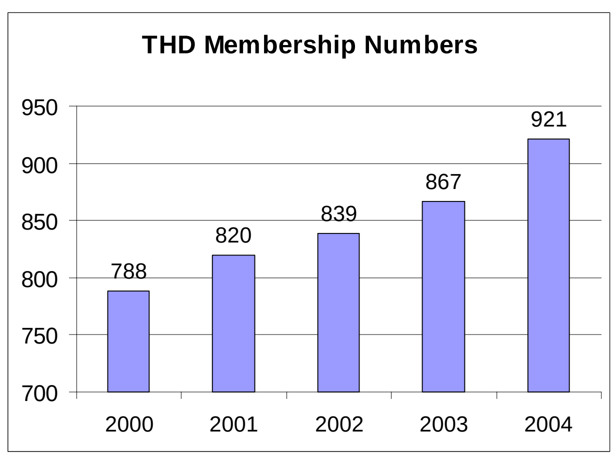
In 2004, the THD gave three major awards: (1) The Technical Achievement Award was presented to Prof. Yassin Hassan of Texas A&M University at the NUTHOS-6 meeting in Japan in October 2004; (2) The Best Paper Award was given to the NURETH-10 paper by Prof. Dr. Horst-Michael Prasser, Research Center Rossendorf, Institute of Safety Research, Dresden Germany; and (3) Distinguished Service Award was presented to Dr. Tom Larson of INL

Technical Achievement Award
Professor Yassin Hassan received the 2004 Technical Achievement Award at the award ceremony during the NUTHOS-6 Conference in Nara, Japan, October 5-9, 2004. The award carries a plaque and a check of $1,000. Professor Hassan gave the award lecture on "The Fantastic World of Bubbles ".

The award was presented to him in recognition of his sustained exceptional contributions to the studies of multiphase flow, turbulence and measurement techniques using Particle Image Velocimetry, 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.

Yassin Hassan is an international authority on multiphase flow and measurement techniques, computational fluid dynamics, large eddy turbulence, turbulence, code modeling, nuclear safety and thermalhydraulics. His contribution in Particle Image Velocimetry (PIV) for two-phase flow is widely recognized. He is currently a professor in the Department of Nuclear Engineering at the Texas A&M University. Hassan received his Ph.D. in nuclear engineering from University of Illinois, and he holds two MS degrees one in Mechanical Engineering from University of Virginia and another in Nuclear Engineering from University of Illinois. Previous to his academic career, he served as Principal Engineer in the Nuclear Power Division at Babcock & Wilcox Company. He is a fellow of both ANS and American Society of Mechanical Engineers (ASME). He received the Glenn Murphy Award of the American Society of Association for Engineering Education (ASEE).

THD Membership



THD membership continues to growth. For the 7th straight year, the number of THD members have increased. This year, we are happy to report a total of 921 members which is an increase of 54 new members over last year, again the biggest one year increase!

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 Officers.

Report on the THD Program Committee

One of the major meetings during 2004-2005 was NUTHOS-6 which was held in Nara-Ken New Public Hall, Nara, Japan. In this conference, 272 technical papers were presented in 57 technical sessions. Highlights of the conferences included 6 plenary sessions, 18 keynote presentations by experts, and an OECD NAE BFBTI workshop that was attended by 30 participants representing 19 organizations from 8 countries. The meeting was attended by 28 participants from USA. The organizing committee was chaired by THD members, Dr. Hiroshi Ninokata and Dr. Jong Kim. During this meeting, it was announced that the NUTHOS-7 meeting will be held in 2007 in Korea.

The ANS Winter Meeting was held in Washington, D.C., at the Omni Shoreham Hotel from November 13 to 17, 2005. THD sponsored four sessions and a total of 33 papers were presented in these sessions. This meeting also had two Embedded Topical Meetings: Best Estimate Twenty-O-Four (BE2004) and Operating Nuclear Facility Safety (2004 ONFS). The THD co-sponsored BE2004 meeting where 48 papers were presented.

This year's Annual Meeting will be held San Diego at the Town and Country Hotel and Conventional Center. THD is sponsoring 4 sessions with 28 summaries and is cosponsoring three sessions in the area of safety. There is an Embedded Topical Meeting: Space Nuclear Conference 2005 (SNC'05) during this meeting.

The next major Thermal-Hydraulics meeting is NURETH-11, which will be held in Avignon, France, on October 2-6, 2005. The conference has received over 500 abstracts and the review process is under completion with full papers acceptance by June 15, 2005. Conference registration has just opened. There is a dedicated website for registration http://nureth11.com/Registration.htm. Dr. Hervé Lemonnier is the Technical Chair for this conference. From our division, Dr. Per Peterson is working as technical Co-chair for this meeting. This meeting offers several technical and sight-seeing tours.

The 2005 ANS Winter Meeting will be held in Washington, D.C. at the Omni Shoreham Hotel on November 14-18, 2005. Our division is sponsoring five sessions. The deadline for summary submission is June 10, 2005. For more information please visit the ANS website www.ans.org/meetings.

It is needless to say that the technical performance of the Division is improving with larger participation at the National Meetings 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 ANS Pittsburgh Section has submitted a proposal to hold NURETH -12 in Pittsburgh, Pa with tentative dates for the meeting October 7-11, 2007.

Professional Development Activities

Our division members are engaged in education, training and attending professional development courses and workshops. The 22nd Zurich Short Course on Modeling and Computation of Multiphase Flows, New Reactor Systems And Methods was offered by Division members Prof. M. Ishii and Prof. M. L. Corradini. Division member, Prof. José N. Reyes, Jr, spent a year and worked as Project Director for *Natural Circulation Coordinated Research Project* at the International Atomic Energy Agency (IAEA), United Nations, Vienna, Austria. Our Division members participate in various professional development workshops. This year's ANS annual meeting in San Diego has two workshops that are of interests to our division members: "Introduction of Thermal Hydraulic RELAP5-3D Code" on June 5, 2005 and "Advanced Gas Reactor Technology Course" on June 9-10, 2005 (2-Day Workshop). Some of our members are attending the American Society of Engineering Education (ASEE) Conference in Portland from June 12-15 and are participating in workshop on Educational Research & Methods. The seminar on "Requirements & Capabilities For CFD Analysis Of Advanced Gas-Cooled Reactors" schedule during the ASME Fluids Engineering Summer Conference (June 22, 2005, http://www.asmeconferences.org/FEDSM05/) and the RELAP5-3D Conference in Jackson Hole, Wyoming, USA, (http://www.inel.gov/relap5/rius/snowking2005.htm) in September 7-9, will also be attended by some of our members.

Newsletter Feature: Advanced Reactor Safety Studies

Shripad Revankar, 2004-5 THD Secretary, shripad@ecn.purdue.edu

With rejuvenated interest in nuclear energy both from industry and government, R&D activities on advanced reactor safety are at a high pace. This year's THD newsletter includes articles on some of the current thermal-hydraulic research and development activities on advanced reactor safety.

Risk Informed Rule Change for Design Basis LOCA

Bert Dunn, Framatome ANP

Over the past three+ decades, many members of ANS THD have been involved in expanding the understanding of the thermal-hydraulic phenomena that characterize the many postulated events analyzed to demonstrate nuclear power plant safety and to support licensing. Simultaneously, the probabilistic risk assessment (PRA) community has developed methods for evaluating event risk. Today, the synergy of these two disciplines has prompted a new industry-wide debate on the application of the "Risk-Informed" approach to changing regulatory rule 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," the NRC regulation regarding loss-of-coolant accidents (LOCAs).

One of the principle insights from the PRA community is that the large break loss-of-coolant accident (LBLOCA) is a relatively small contributor to plant risk. Yet, it continues to be a dominant determinant of the design basis for nuclear power plants. The NRC is considering a rule change to 10CFR50.46 to more appropriately account for LBLOCA risk within the design basis and licensing basis for nuclear power plants. The proposed rule change would establish a new maximum design basis break size, referred to as the transition break size (TBS), along with an alternate set of design criteria for breaks of that size or larger. Licensees will be required to demonstrate mitigation capability for breaks larger than the TBS, but with less restrictive analysis assumptions than those currently applied. The proposed rule change is an appropriate next step in promoting risk informed, performance based regulation.

The nuclear industry, believes that removing LBLOCA from the design basis as an implementation of Option 3 (SECY-98-300) of the NRC risk informed initiative will result in safety benefits and will reduce unnecessary regulatory burden associated with the current design basis. A meeting was held on January 13 2005 between the Westinghouse Owners Group (WOG), the NEI Option 3 Task Force, and the NRC to discuss potential safety benefits of LBLOCA redefinition and to identify example benefits to be quantitatively evaluated. An outcome of that meeting is that two example safety benefits were chosen to be quantitatively evaluated to demonstrate potential safety benefits from implementation of the proposed rule change. The NRC and industry, led by the WOG, are working in parallel to quantify example safety benefits to support the rule change and an implementation Regulatory Guide. The rule should be available for public comment in June, and the final rule could be issued as soon as the end of this year

Condensation Heat Transfer Experiments for Reactor Safety

Karen Vierow, Purdue University, vierow@ecn.purdue.edu

Condensation heat transfer rates are key determinants of the performance of many nuclear power plant safety systems. The condensation phenomena can also contribute towards greater reactor safety in some instances where it is not the main heat transfer mechanism by design. The Laboratory for Nuclear Heat Transfer Systems in Purdue University's School of Nuclear Engineering is performing several experimental and analytical studies into condensation heat transfer at the fundamental science level and the system performance level. These programs are summarized below.

Fundamental Condensation Heat Transfer Studies in Horizontal Tubes

Several current reactor safety systems have vertically-oriented condenser tubes that rely on gravity for condensate drainage and condensation on the inner side of the tube surfaces. Horizontal heat exchangers have the potential for greater heat transfer efficiency because of a comparatively small heat transfer resistance at the top of the tube. Numerous industrial applications of horizontal heat exchangers may also be found, such as for refrigeration and cooling of electronics. However, many aspects of condensation phenomena in horizontal tubes have not been clarified. Further, the three-dimensional nature of the phenomena and the presence of noncondensable gases complicate the problem significantly.

As part of a DOE Nuclear Engineering and Education Research (NEER) project, an experimental facility has been constructed and fundamental data has been obtained for heat transfer coefficients in a single tube when a noncondensable gas is mixed with the incoming condensable gas. Unique features of the experimental program include a new technique to accurately measure the local heat flux through thin-walled tubes and the ability to acquire local heat transfer data at any angle around the condenser tube.

Figure 1 shows a portion of the test section near the steam/noncondensable gas inlet. The condenser tube is a stainless steel tube with a wall thickness of 2.1 mm. Cooling water flows axially along the outside of the tube in the annular cooling jacket. The outer boundary of the cooling jacket consists of polycarbonate plastic blocks that are bolted together with acrylic flanges.

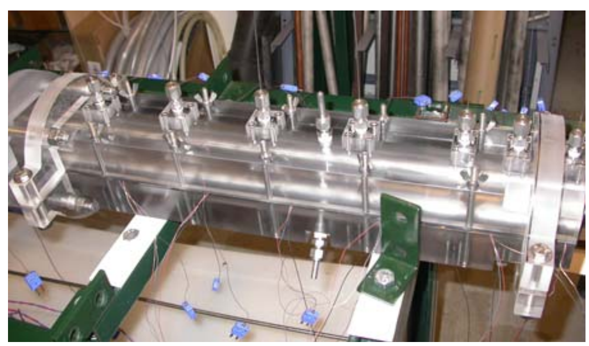


Figure 1 Single Tube Facility Test Section

The local wall heat fluxes on the condenser tube are obtained by measuring the wall inner surface-to-outer surface temperature gradient and performing a conduction calculation based on this temperature drop. A challenge is to measure the inner surface temperature of the condenser tube without altering the phenomena. For the condenser tube inner surface temperatures, a hole was drilled nearly through the tube wall and a thermocouple with a plug design was tapped into the hole. The red and blue wires for these thermocouples are seen emerging from between the top and bottom haves of the secondary side in Figure 1. A technique was devised to calibrate the thermocouple pairs against known heat fluxes and obtain a calibration curve for each thermocouple pair.

Tests have been performed over steam flow rates, noncondensable gas mass fractions and pressures that cover the expected DBA LOCA conditions for a BWR containment. The data show that the local heat transfer coefficients may be over a factor of 2 larger at the top of the tube than at the bottom. Factors affecting the local heat transfer rates include the local flow regime (annular, wavy-annular or stratified), the local noncondensable gas concentration, condensate draining rates and the vapor-liquid interface behavior.

Theoretical models are currently being developed and will be validated against the experimental data. Empirical tools are also being developed for light water reactor safety codes and to support design efforts by numerical simulation of containment cooling processes.

Future issues include acquiring data at peripheral locations other than the top and bottom of the condenser tube, visualization for flow regime identification and investigation of the condensate wave and droplet behaviors.

Design of Horizontal Heat Exchanger for Passive Containment Cooling

As part of the same DOE NEER program, a prototype 6-tube horizontal heat exchanger has been designed for a Passive Containment Cooling System of future LWRs to demonstrate whether it has advantages over the vertical tube bundle heat exchangers. The results of the single-tube experiments are used as the basis for determining local heat transfer rates in the horizontal tube bundle, where such extensive instrumentation is not possible.

The 6-tube facility has been built with a heat transfer region 4 meters in length. Current vertical tube designs have a much larger number of tubes and the horizontal design has been scaled down from the expected actual size. The tube-to-tube effects, the influence of the secondary-side coolant pool heat transfer and the limits of stable operation of the condenser will be evaluated. Currently, the steam supply is connected to just one tube and single-tube testing with conditions as close as possible to those of the single-tube facility is being conducted. Once the applicability of the single-tube facility results have been shown applicable to the tube-bundle facility, the remainder of the condenser tubes will be connected to the steam supply and tube bundle testing will be conducted.

Theoretical models for the tube bundle heat transfer are being developed. The technical goals are to investigate the major factors of horizontal heat exchanger performance in passive containment heat removal from a light water reactor following a design basis accident LOCA and to develop modeling methods for use in nuclear thermal-hydraulic codes.

Future work includes devising a venting system that is prototypic of the current PCCS venting systems and system testing under steady state and transient conditions.

Condensation Heat Transfer in PWR Steam Generator Tubes in the Presence of Noncondensable Gases

Under certain circumstances in a Pressurized Water Reactor (PWR), the coolant system may be in a partially drained state and reflux condensation in the steam generator U-tubes can be the major heat removal mechanism. The situation under consideration in this project corresponds to loss of the Residual Heat Removal System during routine plant shutdown and maintenance. Any noncondensable gases would degrade the heat transfer rate. If heat removal rates are insufficient, this situation could lead to core boil-off, fuel rod heatup, and eventually core damage.

The Institute of Nuclear Safety System, Inc. (INSS), a research institute of Kansai Electric Power Company in Japan, and the Nuclear Heat Transfer Systems Laboratory at Purdue University established a collaborative research program to investigate the effectiveness of reflux condensation in PWR steam generator U-tubes in the presence of noncondensable gases. The final objectives are to provide local heat transfer data for development of methods to analyze reflux condensation in PWR steam generator U-tubes and to investigate the potential for flooding.

The test section, shown in Figure 2, is a double-pipe heat exchanger with the outer pipe cut axially in half and clamped back together. Key features of the experimental data are that they are local data under laminar steam/gas mixture and condensate film flow and that the data are taken from a test section with dimensions similar to an actual steam generator tube. Steady state data were obtained under various steam and air inlet flow rates and pressures. The data show a significant degrading effect of noncondensable gas on heat transfer coefficients. From the data, empirical correlations for the reflux condensation local heat transfer coefficient and the local Nusselt number under laminar conditions were derived.

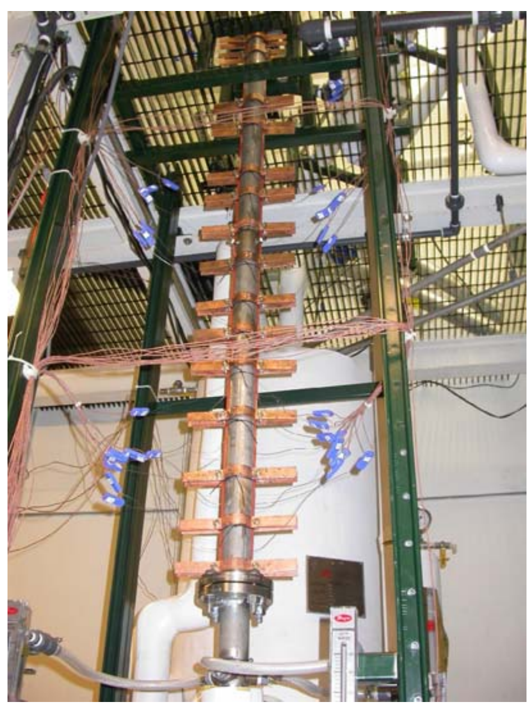


Figure 2 Reflux Condensation Test Section

Future work includes obtaining data under wider Reynolds number ranges for both the steam/gas mixtures and the condensate film and investigating the occurrence of flooding.

In-vessel Retention Strategy for High Power Reactors

J. L. Rempe, Idaho National Laboratory (INL); K. Y Suh, Seoul National University (SNU); F. B. Cheung, Pennsylvania State University (PSU); and S. B. Kim, Korea Atomic Energy Research Institute (KAERI), Joy.Rempe@inl.gov

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. 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 whether currently proposed external reactor vessel cooling (ERVC) of a reactor vessel submerged in a containment cavity filled with water could provide sufficient heat removal for higher power reactors (up to approximately 1500 MWe) without additional enhancements. A three-year, U.S. - Korean International Nuclear Energy Research Initiative (INERI) project was recently completed in which INL, SNU, PSU, and KAERI explored new options, such as enhanced ERVC performance and the use of an in-vessel core catcher (IVCC), that have the potential to ensure that IVR is feasible for high power reactors. State-of-the-art analytical tools and key U.S. and Korean experimental facilities were used to explore each of these options. Although this program focused upon the Korean Advanced Power Reactor 1400 MWe (APR1400) design, recommendations were developed so that they can easily be applied to a wide range of existing and advanced reactor designs.

Key results, conclusions, and recommendations for future evaluations for activities completed in this program are highlighted below.

- *IVR Bounding Scenario Selection* – To obtain representative late-phase melt conditions that could affect the potential for IVR of core melt following a severe accident in the APR1400, the SCDAP/RELAP5-3D© and SCDAP/RELAP5/MOD3.3 codes were applied to the APR1400. Although an extensive series of severe accident calculations is required to identify bounding transients, loss-of-coolant accidents (LOCAs), station blackouts (SBOs), and loss of feedwater (LOFW) sequences were assumed to be major IVR scenarios. Accordingly, a cold leg break (representing the LOCA response) and an SBO with LOFW (to combine remaining dominant IVR scenarios) were selected for analysis. Predicted values for vessel failure time, hydrogen generation, melt relocation masses, melt relocation volumes, decay heat in the relocated corium, and power densities in the relocated corium were compared. For the cases of interest, values predicted by SCDAP/RELAP5/MOD3.3 and SCDAP/RELAP5-3D© were similar. Regardless of the transient considered, results for all calculations led to predictions of large melt masses (~100,000 kg total, or more) relocating at high temperatures (~3,000 K, or higher). These results appear to be consistent with the nature of the "bounding" (e.g., low frequency) transients considered. Specifically, all cases involved complete core dryout and subsequent core heatup in steam. Protracted periods (~1 h, or more) of complete core uncovery occurred in each calculation, leading to large masses of molten core and structural materials relocating.

- *IVCC Design and Evaluation* - Although experimental and analytical investigation results suggest that an IVCC is viable and can significantly reduce the heat loads to the vessel from prototypic core materials, more detailed evaluations are needed to fully assess its merit. For example, more data are needed to confirm that proposed coatings do not degrade over the lifetime of the reactor or when subjected to sustained heat loads. Furthermore, confirmatory thermal-hydraulic tests should be completed to assess the impact of the IVCC on coolant flow.

- *IVCC Narrow Gap Heat Transfer* – The ability to predict heat transfer from a narrow gap has several applications in assessing the potential for IVR. It is needed to predict heat transfer in cracks that develop within the relocated core materials, between relocated core materials and the vessel or the "engineered gap" that may occur if an IVCC is placed within the reactor vessel. Significant progress has been made toward developing a boiling curve for predicting heat transfer in narrow gaps with CCFL. Results indicate that surface angle, gap size, pressure, and dimensional effects must be considered. Data from tests completed in several facilities were used to develop a narrow gap heat transfer model that considers these effects.

- *ERVC Enhancements* – Several investigations were completed that explore methods to enhance the margin associated with ERVC. Results show that using an enhanced insulation configuration placed around the vessel structure can significantly improve the steam venting process (and subsequently increase local critical heat flux (CHF) values). In addition, tests show that coatings can be applied to the vessel external surface and enhance its coolability. Finally, tests were completed to evaluate the combined effect of enhanced insulation and vessel coatings, and results indicate that the combination of these methods further enhances vessel coolability (but that the combined effect is less than the sum of the individual effects). Correlations were developed and applied for predicting CHF with an enhanced vessel/ insulation configuration, vessel coatings, and a coated vessel with an enhanced vessel/insulation configuration. However, it should be noted that these evaluations represent an initial study for possible enhancements. To maximize the benefits of enhanced ERVC, additional studies are suggested to further optimize parameters associated with the vessel/insulation design and vessel coatings.

- *IVR Margin Assessment* – Results from calculations performed with several methods (SCDAP/RELAP5-3D©, VESTA, LILAC, RELAP5, and a lumped parameter model) indicate that proposed enhancements are needed to provide additional margin for IVR when the APR1400 vessel is subjected to bounding heat loads. In the APR1400 cases evaluated, the enhanced cooling associated with a coated reactor vessel or an enhanced vessel/insulation configuration was sufficient to reduce heat fluxes below CHF. Even greater margins for IVR were predicted for cases with both a coated vessel and an enhanced vessel/insulation configuration. Analyses suggest that significant additional cooling is possible with an IVCC. However, it should be noted that these calculations were an initial application of these methods to the extreme conditions proposed in bounding scenarios. In particular, these initial applications indicated that several model parameters applied in previous calculations were not applicable to these severe conditions. Additional evaluations are recommended that consider other debris end states, a broader range of sensitivity studies, more detailed IVCC evaluations, and model improvements for molten pool heat transfer and vessel melting. Furthermore, additional data and evaluations are needed for simulating the heat transfer between the molten pool, its crust, and the vessel for conditions where the heat load is sufficient to cause significant crust thinning.

Evaluations from this K-INERI project indicate that the proposed IVR enhancements significantly increase IVR margins for higher power reactors. Additional evaluations suggested as follow-on tasks are needed to optimize these enhancements, ensure their long-term endurance, and improve methods for simulating their performance.

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: Robert Martin
Email: Robert.Martin@framatome-anp.com

Vice Chair/Chair Elect: Joy Rempe
Email: Joy.Rempe@inl.gov

Secretary: Chang Oh
Email: Chang.Oh@inl.gov

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

Executive Committee (3 year term):
Jong Kim (2008)

2004-2005 THD Officers:

Chair: Yassin A. Hassan

Vice Chair/Chair-Elect: Robert Martin

Secretary: Shripad T. Revankar

Treasurer: Jovica Riznic

Executive Committee Members:

Joy Rempe (2005)
Per Peterson (2005)
Whee G. Choe (2005)
David Bessette (2006)
Martin Bertodano (2006)
Cetin Unal (2006)
Karen M. Vierow (2007)
Chang H. Oh (2007)

Committee Chairs:

Program Committee – Kurshad Muftuoglu
Honors and Awards Committee – Bill Cheung
Nominating Committee – Jong Kim
Membership Committee – Bob Martin