16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics

August 30—September 4, 2015
Hyatt Regency, Chicago, Illinois, USA

CALL FOR PAPERS

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Paper Deadlines
Abstract Submission Deadline: 12/15/2014 (Extended)
Abstract Submission Website: http://epsr.ans.org/meeting/?m=196
Abstract Acceptance: 12/31/2014
Draft Full-Length Paper Submission: 02/28/2015
Review Notification: 04/30/2015
Final Paper: 05/31/2015

The page limit for NURETH-16 papers is 14 pages.

Selected papers will be published in a special edition of Nuclear Technology and Nuclear Science and Engineering journals.

About the Meeting

The Chicago Local Section of the American Nuclear Society (ANS) is pleased to host the 16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-16) at the Hyatt Regency Chicago from August 30 – September 4, 2015.

Chicago is one of America’s greatest cities, and one of the most easily accessible destinations in the U.S. for travelers from around the world due to its central location. Chicago has a proven history of hosting past ANS meetings, including the 2012 Annual Meeting.

NURETH-16 attendees will be a short distance from Chicago attractions, such as Navy Pier, architectural boat tours, theatrical and musical entertainment, and countless dining and shopping options.

Building on the foundation of past meetings, including NURETH-15 in Pisa, Italy, NURETH-16 will invite experts in the field of thermal hydraulics from around the world to share their research and discuss important issues related to nuclear reactor thermal hydraulics.

NURETH-16 will also be a venue for organizations to showcase their expertise in nuclear reactor technology, with several opportunities for meeting sponsors to exhibit their products and services to the nuclear reactor thermal-hydraulics community.

Questions: nureth16@anl.gov

http://nureth16.anl.gov
NURETH 16

TOPICS AND SESSIONS

Basic Thermal Hydraulics
- Two-Phase Flow and Heat Transfer Fundamentals, including Experimental Thermal Hydraulics and Instrumentation
- Boiling and Condensation Fundamentals
- Multifield Two-Phase Flow Modeling
- Flow-Induced Vibration in Reactor Components
- Supercritical Fluids Thermal Hydraulics and Heat Transfer
- Interfacial Area Transport (Database, Modeling, Measurement Techniques)
- Microscale and Nanoscale Phenomena
- Natural Circulation, Passive Safety Systems and Related Phenomena
- Subchannel Fluid Dynamics and Heat Transfer Experimental Measurement Techniques and Flow Visualization

Operation & Safety of Existing Reactors
- Plant Life Extension and Power Up-Rating
- Instabilities and Nonlinear Dynamics
- NPP Transient and Accident Analysis
- Safety of Sodium-Cooled Reactors
- Safety of VVER Reactors
- Full Spectrum LOCA Evaluation Methodology
- BEPU Analysis and Challenges in Licensing
- Addressing Scaling Issues

Severe Accidents
- Molten Core Natural Convection and Physical-Chemical Phenomena
- Modeling and Experiments of Severe Accidents
- Natural Convection and Mixing Phenomena, Modeling and Experiments
- Fuel Coolant Interaction, Modeling and Experiments
- Direct Containment Heating by Dispersed Molten Fuel
- Debris Bed Cooling
- Combustion and Fires, Modeling and Experiments
- Advanced Design Features for Severe Accident Mitigation
- Analytical Activities on the Accident Progression and In-core Status of Fukushima Dai-ichi Units
- Hydrogen and Fission Product Behavior

Code V&V, Design and Operation of SETF Including Instrumentation
- Computational Fluid Dynamics V&V (DNS, LES RANS, etc.)
- Computational Multi-Fluid Dynamics
- Core Thermal Hydraulics and Subchannel Analysis
- Plant System Code Validation
- Experiments and Databases for Assessment and Validation (including of 3D Models)
- Boiling and Condensation Heat Transfer
- CHF and Post CHF Heat Transfer, Flooding and CCFL
- Containment Analysis
- Boron Dilution/Mixing
- Experiment Design for V&V

Others, including Waste Management & Fusion Technology

Code Development Including Numerics (System TH and CFD)
- Computational Fluid Dynamics (DNS, LES, RANS, etc.)
- Computational Multi-Fluid Dynamics
- Core Thermal Hydraulics and Subchannel Analysis
- Plant System Code Development
- Boron Dilution/Mixing
- Steam Generator Thermal Hydraulics
- Pressure Surges in Nuclear Power Plants
- Containment Analysis
- Accuracy and Uncertainty Analysis
- Fast Transient Modeling
- Enhanced Near-Wall Flow and Heat Transfer Modeling
- Fluid and Structures Mechanical Interactions
- Multiscale Multiphysics Applications in Thermal Hydraulics

Thermal Hydraulics in Advanced Reactors
- Thermal Hydraulics in High-Temperature Gas-Cooled Reactors
- Thermal Hydraulics in Salt-Cooled High-Temperature Reactors
- Thermal Hydraulics in Sodium-Cooled Fast Reactors
- Thermal Hydraulics in Supercritical Water Reactors
- Thermal Hydraulics in Lead-Cooled and Lead-Bismuth-Cooled Fast Reactors
- Thermal Hydraulics in Small Modular Reactors

SPECIAL SESSIONS
S1. Advancements in the Prediction of DNB with CFD, E. Baglietto
S2. NEAMS Sponsored Advances in Thermal Hydraulics Modeling and Simulation, J. Thomas
S3. The NURESAFE European Project: Multiscale Thermal Hydraulic Analyses, D. Bestion
S4. CFD Modelling of Fuel Assemblies: From High Fidelity to Low Resolution Models, F. Roelofs
S5. Hydrogen Management after Fukushima, E. Komen
S7. Advancements in SFR Thermal Hydraulics, T. Sofu
S8. OECD/NEA Benchmark Study of the Accident at the Fukushima Dai-ichi Nuclear Power Plant, M. Pellegrini and R. Gauntt
S9. Heat Transfer in Supercritical Flows, M. Rohde and X. Liu
S10. Advances in Instrumentation and Measurement Techniques for V&V, Y. Hassan and S. Lomperski
S11. Addressing the GSI-191: Progress in Methodologies and Technologies, R. Vaghe
S13. Thermal Hydraulic Experiments and Numerical Analyses in Support of the MYRRHA, K. Van Tichelen
S14. CFD Benchmark of NESTOR High Fidelity PWR Rod Bundle Data at In-Core Conditions, D.M. Wells and Y. Hassan
S15. Realistic BWR LOCA Evaluation: Methodology Development and Application, K. Mufuoglu
S16. FHR Integral and Separate Effects Test Experiments, P. Peterson and X. Sun
S17. Issues and Advances for Application of Thermal Hydraulic Codes to FHRs, P. Peterson and X. Sun