18th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-18)

August 18-22, 2019 | Portland, OR, USA | Marriott Portland Downtown Waterfront

CALL FOR PAPERS

EXECUTIVE CHAIRS

General Chairs
Jose Reyes, NuScale Power
Mark Peters, Idaho National Laboratory
Kurshad Muftuoglu, GE Hitachi

Technical Program Chairs
Brian Woods, Oregon State University

Technical Program Chairs Continued
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Akira Yamaguchi, Osaka University

Assistant Technical Program Chairs
Wade Marcum, Oregon State University
Brian Jackson, TerraPower
Donald Todd, Pacific Northwest National Laboratory
Jeff Luitjens, NuScale Power

ABSTRACT DEADLINE: MONDAY, DECEMBER 31, 2018

DECEMBER → SUBMISSION OF ABSTRACTS: **Monday, December 31, 2018**
JANUARY → NOTIFICATION OF ABSTRACT ACCEPTANCE: **Friday, January 4, 2019**
MARCH → DRAFT FULL-LENGTH PAPER SUBMISSION: **Friday, March 1, 2019**
MAY → FULL-PAPER REVIEW NOTIFICATION: **Wednesday, May 1, 2019**
JUNE → REVISED FULL-PAPER SUBMISSION: **Monday, June 3, 2019**

GUIDELINES

The limit for abstract submissions is 250 words. The limit for full-paper submissions is 14 pages. The conference proceedings will be distributed on a flash drive. Selected papers will be published in the Special issues of Nuclear Technology, Nuclear Science and Engineering, and Nuclear Engineering and Design.

ABOUT THE MEETING

NURETH is one of the premier gatherings for experts in nuclear reactor thermal hydraulics and related topical areas. The meeting is held every two years. The Eastern Washington Section of the American Nuclear Society (ANS) is pleased to host NURETH-18 in Portland, Oregon, USA. There are few destinations as beautiful as the Pacific Northwest in the summer. Portland is an ideal destination location that has large-scale amenities while providing local flair. Summer in Portland provides a warm and comfortable location for meeting participants and a sunny getaway for spouses and guests.

SUBMIT A SUMMARY
ans.org/meetings

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TECHNICAL TRACKS

HIGH-QUALITY PAPERS (14 PAGE MAXIMUM) ARE SOLICITED IN THE FOLLOWING AREAS

1. 1. FUNDAMENTAL THERMAL HYDRAULICS
   - 1.a. Fundamental Thermal Hydraulics: General
   - 1.b. Fundamental Thermal Hydraulics: Experiments
   - 1.c. Boiling and Condensation Fundamentals
   - 1.d. Experimental Measurement Techniques and Flow Visualization
   - 1.e. Interfacial Area Transport
   - 1.f. Multifield Two-Phase Flow Modeling
   - 1.g. Natural Circulation, Passive Safety Systems and Related Phenomena
   - 1.h. Subchannel Fluid Dynamics and Heat Transfer
   - 1.i. Two-Phase Flow and Heat Transfer Fundamentals

2. COMPUTATIONAL THERMAL HYDRAULICS
   - 2.a. Computational Thermal-Hydraulics: General
   - 2.b. Accuracy and Uncertainty Analysis
   - 2.c. Computational Fluid Dynamics
   - 2.d. Computational Multi-Fluid Dynamics
   - 2.e. Containment Analysis
   - 2.f. Core Thermal Hydraulics and Subchannel Analysis
   - 2.g. Multiscale Multiphysics Applications in Thermal Hydraulics
   - 2.h. Plant System Code Analysis and Development

3. VERIFICATION AND VALIDATION
   - 3.a. Verification and Validation: General
   - 3.b. Boiling and Condensation Heat Transfer
   - 3.c. CHF and Post CHF Heat Transfer, Flooding and CCFL
   - 3.d. Computational Fluid Dynamics VandV
   - 3.e. Experiments and Data Bases for Assessment and Validation
   - 3.f. Plant System Code Validation

4. OPERATION AND SAFETY OF EXISTING REACTORS
   - 4.a. Operation and Safety of Existing Reactors: General
   - 4.b. Addressing Scaling Issues
   - 4.c. BEPU Analysis and Challenges in Licensing
   - 4.d. Instabilities and Nonlinear Dynamics
   - 4.e. NPP Transient and Accident Analysis

5. SEVERE ACCIDENTS
   - 5.a. Severe Accidents: General
   - 5.b. Advanced Design Features for Severe Accidents Mitigation
   - 5.c. Debris Bed Cooling
   - 5.d. Fuel Coolant Interaction, Modeling and Experiments
   - 5.e. Hydrogen and Fission Product Behavior
   - 5.f. Modeling and Experiments of Severe Accidents
   - 5.g. Natural Convection and Mixing Phenomena, Modeling and Experiments

6. THERMAL HYDRAULICS IN ADVANCED REACTORS
   - 6.a. Thermal Hydraulics in Advanced Reactors: General
   - 6.b. Thermal Hydraulics in High-Temperature Gas-Cooled Reactors
   - 6.c. Thermal Hydraulics in Lead-Cooled and Lead-Bismuth-Cooled Fast Reactors
   - 6.d. Thermal Hydraulics in Salt-Cooled High-Temperature Reactors
   - 6.e. Thermal Hydraulics in Small Modular Reactors
   - 6.f. Thermal Hydraulics in Supercritical Water Reactors
   - 6.g. Thermal Hydraulics in Sodium-Cooled Fast Reactors: Steady Analysis
   - 6.h. Thermal Hydraulics in Sodium-Cooled Fast Reactors: Transient Analysis
   - 6.i. Thermal Hydraulics in Sodium-Cooled Fast Reactors: Severe Accident Analysis

7. SPECIAL TOPICS

   Topics to be identified based on current relevant areas within the field.

8. PANELS

   Topics to be identified based on current relevant areas within the field.

Paper acceptance will be based upon originality of the work, strictly implemented methods or models, quality of results, impact of the scientific advances to the field of thermal hydraulics, conclusions supported by data, proper citing of references, and use of correct grammar and spelling.