20th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-20)

August 20-25, 2023 | Washington, DC, USA | Washington Hilton

CALL FOR PAPERS

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ABSTRACT DEADLINE: NOW OCTOBER 28, 2022

OCTOBER	SUBMISSION OF ABSTRACTS: **October 28, 2022**
NOVEMBER	AUTHOR NOTIFICATION OF ACCEPTANCE: **November 4, 2022**
JANUARY	FULL PAPER SUBMITTED FOR REVIEW: **January 13, 2023**
MARCH	REVIEW NOTIFICATION: **March 10, 2023**
APRIL	FINAL FULL PAPERS DUE: **April 14, 2023**

GUIDELINES

The limit for abstract submissions is 250 words. The limit for full-paper submissions is 14 pages. Papers exceeding 14 pages will be rejected. If an exception is made and a paper over 14 pages is accepted, page charges are $100/page for p. 15 and above. The conference proceedings will be distributed digitally. Selected papers will be published in the special issues of *Nuclear Technology*, *Nuclear Science and Engineering*, *Fusion Science and Technology*, and *Nuclear Engineering and Design*.

ABOUT THE MEETING

NURETH is the premier gathering for experts in nuclear reactor thermal hydraulics and related topical areas. This meeting is held every two years. The Washington DC ANS Section is pleased to host NURETH-20 in Washington, DC, USA. Washington, DC is more than the capital of the USA– it is the center of both support and regulation of the nuclear industry. Washington DC is not only a center of government but also a tremendous visitor center with cultural attractions for all.

NURETH-20 HOSTS

Raymond Furstenau, USNRC
Yassin Hassan, Texas A&M
Stephen M. Bajorek, USNRC

SUBMIT AN ABSTRACT

epsr.ans.org/meeting/?m=381

PROGRAM SPECIALIST

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TECHNICAL TRACKS

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

1. FUNDAMENTAL THERMAL HYDRAULICS
   - 1A. Two-Phase Flow and Heat Transfer
   - 1B. Boiling and Condensation
   - 1C. Interfacial Area Transport
   - 1D. Critical Heat Flux and DNB
   - 1E. Natural Circulation in Reactor Systems
   - 1F. Thermal Hydraulic Scaling
   - 1G. General

2. COMPUTATIONAL THERMAL HYDRAULICS
   - 2A. Thermal Hydraulics System Code Development and Analysis
   - 2B. Computational Fluid Dynamics (CFD)
   - 2C. Multiphase CFD
   - 2D. Multiphysics Development and Applications
   - 2E. DNS for Model Development
   - 2F. Multiscale CFD and Coupling with System Codes
   - 2G. Subchannel Thermal Hydraulic Analysis
   - 2H. General

3. EXPERIMENTAL THERMAL HYDRAULICS
   - 3A. Experimental Methods and Instrumentation
   - 3B. Integral and Separate Effects Tests
   - 3C. Tests for Assessment of CFD
   - 3D. Experimental Databases and Preservation
   - 3E. Experiments for Advanced and Special Purpose Reactors
   - 3F. Rod Bundle Experiments
   - 3G. Critical Heat Flux and Post-CHF Experiments
   - 3H. UQ Methods and Best Practices for Experiments
   - 3I. General

4. VERIFICATION, VALIDATION AND UNCERTAINTY QUANTIFICATION (VVUQ)
   - 4A. Verification and Validation of Systems Codes
   - 4B. Verification and Validation of Subchannel Codes
   - 4C. Best Practices in CFD
   - 4D. Uncertainty Methodology Development
   - 4E. BEPU Analysis and Challenges in Licensing
   - 4F. General

5. WATER-COOLED REACTOR OPERATIONS AND ANALYSIS
   - 5A. LWR Operation and Safety Analysis
   - 5B. HWR Operation and Safety Analysis
   - 5C. VVER Operation and Safety Analysis
   - 5D. BWR Instabilities and Nonlinear Dynamics
   - 5E. Small Modular LWRs
   - 5F. General

6. SEVERE ACCIDENTS
   - 6A. Severe Accident Scenarios and Source Term
   - 6B. In-Vessel Corium and Debris Bed Coolability
   - 6C. Ex-Vessel Corium Interaction and Coolability
   - 6D. Containment TH, Hydrogen and Fission Product Behavior
   - 6E. Design Features to Prevent Severe Accidents
   - 6F. Uncertainty in Severe Accident Modeling
   - 6G. Severe Accidents in Advanced Reactors and Nuclear Installations
   - 6H. General

7. NEW AND ADVANCED REACTORS
   - 7A. High-Temperature Gas Cooled Reactors
   - 7B. Liquid Metal Cooled Reactors
   - 7C. Molten Salt Reactors
   - 7D. Supercritical Water Cooled Reactors
   - 7E. Microreactors
   - 7F. Reactors for Space Applications
   - 7G. Offshore Nuclear Platforms
   - 7H. Advanced Reactor Fuel
   - 7I. General

8. SPECIAL TOPICS
   - 8A. Hydraulics in Medical Isotope Production
   - 8B. Fluid-Structure Interactions
   - 8C. Accident Tolerant Fuel
   - 8D. Machine Learning and Artificial Intelligence for TH
   - 8E. Test and Prototype Reactors
   - 8F. Thermal Hydraulics of Fusion Reactors
   - 8G. International Benchmarks
   - 8H. Reliability of Passive Systems
   - 8I. Post-Fukushima Thermal Hydraulic Research
   - 8J. Integrated Energy Systems
   - 8K. Decommissioning
   - 8L. NEAMS Thermal-Hydraulics IRP
   - 8M. Memorial Session in Honor of Prof. Peter Griffith
   - 8N. Liquid Metal Heat Transfer
   - 8O. BEPU for Pressurized Thermal Shock
   - 8P. Flow Induced Vibrations in (GO-VIKING)
   - 8Q. OECD/NEA ARC-F Project
   - 8R. Testing and Analysis for Lead Fast Reactor Development