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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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
ans.org/meetings

PROGRAM SPECIALIST
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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