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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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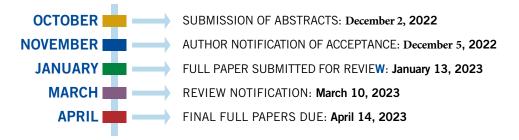
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ABSTRACT DEADLINE: NOW December 2, 2022



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*. For authors who are unable to travel to NURETH 20 because of country-wide travel restrictions, we will provide a remote-presentation opportunity. Further details will be forthcoming.

ABOUT THE MEETING

NURETH is <u>the</u> 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
- 31. General

4. VERIFICATION, VALIDATION AND UNCERTAINTY QUANTIFICATION (VVIIO)

- 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
- 71. 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
- 81. 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
- 80. 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

