

Richard R. Schultz, Ph.D., P.E.

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Position Title:

- Research Professor: Idaho State University: <https://www.isu.edu/ne/people/faculty/>
 - Joint Appointment: Idaho National Laboratory/Idaho State University
- Consulting Engineer: Schultz Engineering PLLC

Summary of Qualifications

Doctor of philosophy degree in nuclear science and engineering, bachelors and masters degrees in mechanical engineering. Have been teaching advanced engineering courses at graduate level since 2010. Have 40-plus years professional experience in:

- Modeling and analyzing the steady-state and transient behavior of power plants and steam supply systems (PWRs, BWRs, Generation III+ and IV systems and commercial steam supply systems, e.g., Seattle Steam Co.);
- Verification and validation (V&V) of advanced thermal-hydraulic (TH) engineering numerical models and determination of adequacy of software used for licensing purposes;
- Design, scaling, specification (based on phenomena identification and ranking table—PIRT studies), and conduct of thermal-hydraulic experiments; and
- Evaluation of user-effect on application of advanced thermal-hydraulic software and best-estimate plus uncertainty techniques applied to advanced thermal-hydraulic software.

Brief Biography: PIRT, Experiment Design, V&V, Software Adequacy*...

Richard R. Schultz, Ph.D., P.E. has had extensive experience in organizing and participating in PIRT studies on advanced nuclear reactors. The results of these PIRT studies were then used as a basis for identifying phenomena that must be simulated in experiments that are scaled to model the prototype nuclear system. Once completed such experiments were used to generate validation data for the purpose of qualifying systems analysis and computational fluid dynamics software for nuclear plant licensing calculations. The final stage of the process, per the U.S. Nuclear Regulatory Commission Regulatory Guide 1.203 for Transient & Accident Analysis Methods, is the determination of the adequacy of the software. Schultz has participated extensively in all of these activities.

His earliest participation in PIRT studies was focused on the Westinghouse AP600 and AP1000 LWRs during the 1990s. Both of these designs have passive emergency core cooling systems. These studies were performed to provide the basis for scaling studies³ used to design the ROSA-AP600 1/30-scale, full-height test facility—located on site at what is now the Japan Atomic Energy Agency (formerly known as the Japan Atomic Energy Research Institute—JAERI) in Tokai-mura, Japan. Schultz was the USNRC Principal Investigator (PI) of the effort to generate validation data—obtained by conducting 24 experiments at JAERI from 1994 to 1997¹⁵--used for qualification of the RELAP5 software^{12,26}

His more recent involvement in performing PIRT studies was centered on: (i) Very high temperature gas-cooled reactors (VHTR) including studies performed in conjunction with the Korean Atomic Energy Research Institute⁶, USNRC³⁵, and INL⁴⁰. These studies identified and focused on the key phenomena that require analysis and the availability of data that describe the key phenomena. From these PIRT studies a complete experimental program was designed and built^{37,39} to provide data to validate and qualify systems analysis and CFD analysis software.^{16,41,44,45,46,51} Recently he has contributed to similar efforts for the Holtec SMR160 LWR and the Sodium sodium fast reactor.

Education

- Ph.D. Nuclear Science & Engineering, 2010, Idaho State University, Pocatello, ID.
- M. Sc. in Mechanical Engineering, 1971, Rensselaer Polytechnic Institute, Troy, NY.
- B. Sc. in Mechanical Engineering, 1967, University of Florida, Gainesville, FL.

* Relevant references indicated by superscripts—see pages 5 through 7.

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Professional Highlights (ordered chronologically)

Positions:

- Consulting Engineer, Schultz Engineering PLLC (2017 to present)
- Research Professor—Idaho State University, Dept of Nuclear Engineering (2014 – present)
- Adjunct Professor—Oregon State University, Dept of Nuclear Engineering & Radiation Physics. (2014 - present)
- Professor of Practice—Texas A&M University, Dept of Nuclear Engineering (2012 – 2019)
- Distinguished Research Engineer—Idaho National Laboratory (1976-2014).
- Experimentalist—United Technology Research Center: combustor-driven chemical laser experiments (1974-76)
- Engineer—General Electric Co: nuclear safety analyst and responsible for experimental validation data from Two-Loop Test Apparatus (1972-74)
- Test Engineer—Pratt & Whitney Aircraft aircraft engine combustor and afterburner research (1968-72)

Schultz Engineering PLLC: contracts

- Terrapower LLC Sodium Sodium Fast Reactor, 2022-present, PIRT and assist in developing necessary ingredients to licensing methodology including scaling, code assessment, etc.
- Holtec, SMR160 LWR, 2020-2022 (via INL contract) PIRT, Scaling, Modeling Project.

Significant Awards & Recognition:

- Listed in web site: “20 Exceptional Nuclear Engineering Professors,” <https://www.onlineengineeringprograms.com/nuclear/nuclear-engineering-profs-to-know>, 2018 to present.
- Outstanding Engineer Award, 2013, School of Engineering, Idaho State University, Pocatello, ID
- Outstanding Service Award, 2013, Dept of Nuclear Engineering, Texas A&M University, College Station, TX
- Idaho National Laboratory Award for Individual Lifetime Achievement in Science and Technology, 2012: the highest achievement award given by the Idaho National Laboratory; this award is accompanied by a \$20,000 purse.
- George Westinghouse Gold Medal, 2012, for eminent achievement in power engineering.
- Fellow, 2008, American Society of Mechanical Engineers (ASME).
- Outstanding Researcher, 2008, Korea Atomic Energy Research Institute, Daejeon, Korea.
- ASME Dedicated Service Award, 2003—society-level award.
- Outstanding Researcher, 1985, Japan Atomic Energy Research Institute, Tokai-Mura, Japan.

Consulting to Professional Societies, Journals, etc:

- Member: ASME Nuclear Engineering Division Executive Committee. (2000 – present)—Chair 2012-13 & 2021-2022.
- Vice Chair: ASME V&V 30 Standards Committee: V&V in Computational Simulation of Nuclear Systems Thermal Fluids Behavior (Chair 2010 to 2016; Vice Chair 2016 – present)
- Member: Verification & Validation (V&V) in Computational Modeling & Simulation Committee (2011 – present)
- Member: Editorial Advisory Board of *Nuclear Engineering and Design Journal*, Elsevier Publishing (2012 – present)
- Associate Editor, *Journal of Nuclear Engineering and Radiation Science*, ASME (2014 – 2021)
- Series Editor, ASME Monograph Series: *Nuclear Engineering and Technology for the 21st Century* (2013 – present)
- Member, ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events—2011 - 2014 (Note: Full Task Force chaired by Dr. Nils Diaz, former Chairman of U.S. Nuclear Regulatory Commission).
- Member: American Nuclear Society (life member), ASME (life member), and National Society of Professional Engineers

Consulting & Work Overseas:

- PhD Nuclear Engineering Dissertation Examiner, 2019, North-West University, Potchefstroom, South Africa.
- PhD Nuclear Engineering Dissertation Examiner: Idaho State University (2020-present), Oregon State University (2018), City College of New York (2024-2025)
- Lead for Verification and Validation, Center of Excellence Project, Idaho National Laboratory and Argonne National Laboratory, 2018 to 2021; project responsible for development and implementation of NEK5000, Pronghorn, RELAP7, and SAM advanced analysis software.

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- Consultant; Computational Methods, Validation, and Benchmark Committee, Very High Temperature Reactor (VHTR) Generation IV International Forum; meetings held in Beijing, China; Oarai Research Center, Japan; Idaho National Laboratory, USA; NRG, The Netherlands; 2014 to 2019.
- Nuclear Regulatory Commission's Principal Investigator (PI) on:
 - (a) ROSA-AP600 Program (1995-1999) at Tokai, Japan;
 - (b) ROSA-IV Program (1983-1986) at Tokai, Japan; and
 - (c) Marviken Critical Flow Experiments (1979-1980) at Vikbolandet, Sweden[†].
- Department of Energy PI on definition and development of Code of Federal Regulations (CFR), Appendix K (10 CFR §50.46) models in RELAP5-3D for pressurized water reactors and boiling water reactors, Institute of Nuclear Energy Research, Lung-Tan, Republic of China (1998 – 2004).
- Subject Matter Expert for International Atomic Energy Agency, Vienna, Austria (1996 to 2015) in thermal-hydraulics, emergency core cooling (ECC) systems behavior and performance, and V&V.
- Member (2011 - 2015): European Community Scientific Advisory Committee for Thermal-Hydraulics of Innovative Nuclear Systems Project, Karlsruhe, Germany.

Invited Speaker:

- Lecturer in Modeling, Experimentation, and Verification (MeV) Summer School, sponsored by Argonne National Laboratory and Idaho National Laboratory, 2017, 2020 & 2023; lectures on thermal-hydraulics.
- *Verification and Validation for Advanced Numerical Analysis Software*, Thermal Fluids Applications in Nuclear Systems Analysis Seminar, Idaho National Laboratory, April 3-4, 2018.
- *International Workshop on Thermal-Hydraulics of Innovative Reactor and Transmutation Systems – THIRS*, April 14-16, 2008, Forschungszentrum Karlsruhe, Germany;
- International Conference on Nuclear Engineering-18, May, 2010, Xi'an, China;
- 8th International Topical Meeting on Nuclear Thermal-Hydraulics, Operations, and Safety (NUTHOS-8), October 10-14, 2010, Shanghai, China;
- Scaling, Uncertainty, and 3-Dimensional Coupled Code Calculations Seminar, sponsored by the Nuclear Research Group of San Piero a Grado of the University of Pisa (Italy), the Faculty of Electrical Engineering and Computing of Zagreb and the School of Industrial Engineering of Barcelona; 2010, Wilmington, NC; 2012, Daejeon, Korea; 2012, Dubrovnik, Croatia;
- Keynote: The Experimental Validation and Application of CFD and CMFD Codes in Nuclear Reactor Technology, OECD/NEA & IAEA, Daejeon, Korea, Sept 11, 2012.

Funded Research at ISU:

- NEUP[‡] Project: Transient Reactor (TREAT) Experiments to Validate MBM Fuel Performance Simulations (2018-2021)
- NEUP Project: Experimental Investigation of Forced Convection and Natural Circulation Cooling of a VHTR Core Under Normal Operation and Accident Scenarios (2015-2019)
- INL-funded Project: Advanced Data Superposition (2015-2019)
- INL-funded Project: V&V for Center-of-Excellence Program (2018-2021)
- INL-funded Project: Validation Data Matrix for SMRs (2020-2021)
- NEUP Project: Mixing of Gases in VHTR Confinements (2019-2023)
- NEUP Project: Natural Circulation Characteristics in a VHTR (2020-2024)
- NEUP Project: Heat Transfer Characterization in Horizontally Orientated Micro High Temperature Gas Reactors (HTGRs) under Pressurized Conduction Cooldown (PCC) Conditions (2021-2024)

MSc & PhD Student Thesis/Dissertation Advisor:

- 1 MSc student
- 1 PhD student

[†] Also represented Electric Power Research Institute

[‡] NEUP = Department of Energy-funded Nuclear Engineering University Project

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Courses Taught at ISU & TAMU:

- Advanced Fluid Mechanics (NE5599—ISU)
- Advanced Thermodynamics (ME6607—ISU)
- Computational Fluid Dynamics (NE5599—ISU)
- Dissertation (ENGR6660—ISU)
- Fluid Transients (NE5599—ISU)
- Gas Dynamics (NE5599—ISU)
- Heat Transfer (ME4424—ISU)
- Nuclear Engineering Project Design—Senior Design (NE4496—ISU)
- Nuclear Engineering Thermal-Hydraulics (NSEN6603—ISU)
- Nuclear Reactor Safety (NUEN619—TAMU)
- Nuclear Reactor Safety & Economics (NE4499/5599—ISU)
- Transport Phenomena: Theory & Applications (NE5599—ISU)
- Two-Phase Flow (NUEN624—TAMU & NSEN6699—ISU)
- Verification & Validation (NSEN6699—ISU)

Publications:

- Author or co-author of over 140 publications: peer-reviewed journal articles, conference papers, book chapters, reports.
- Publications linked to Major Accomplishments (see next section): references 1 through 36.
- Significant publications from 2010 to date: see references 1, 16 to 20, and 37 to 51.

Patents & Licenses:

- Two patents on passive control rod scram mechanism (with A. Ougouag and W. Terry).
- Licensed professional engineer, State of Idaho, mechanical engineering, License No. 8841.

Major Accomplishments (Relevant references indicated by superscripts—see pages 5 to 7)

- Provided report and recommendations for closure on experiments designed to provide validation data for Department of Energy's methods development to analyze behavior of very high temperature gas-cooled reactors^{46,50}.
- Provided report and recommendations regarding nuclear options to the Governor's office for the state of Utah¹⁷.
- Founding Member, ASME V&V30 Code Committee: V&V in Computational Simulation of Nuclear Systems Thermal Fluids Behavior²⁰
- Demonstrated that condensation-induced water hammer can be mitigated using a stratified layer of saturated water.^{8, 9, 19} For highly subcooled water flowing in the presence of steam in partially-filled horizontal piping, as may occur in passive emergency core cooling systems in advanced reactor systems, a layer of saturated water overlying the subcooled water was shown to prevent condensation-induced water hammer. Schultz dissertation¹⁹ provides first definition of the subcooled water and steam flow envelopes which enable a saturated water layer to segregate subcooled water from steam. This result has opened a new area of scientific research.
- Founding Member, American Society of Mechanical Engineers (ASME) Presidential Task Force on Response to Japan Nuclear Power Plant Events.¹
- Principal Investigator responsible for guidance and direction for development and code assessment of Code of Federal Regulations (CFR), Appendix K (10 CFR §50.46) version of RELAP5-3D for pressurized water reactors and boiling water reactors for Institute of Nuclear Energy Research, Lung-Tan, Republic of China (Taiwan).⁷ Enabled INER to perform fuel reload licensing analyses for ROC LWRs.
- Defined all boundary/initial conditions and responsible for recommending acceptance of completed tests, as US Principal Investigator, of the ROSA-AP600 experiments for the U.S. Nuclear Regulatory Commission (these experimental data were used to validate the systems analysis codes which the NRC ultimately used to perform audit calculations for the Westinghouse AP600 license submittal).^{15,36}
- Lead- or Co-Author of key papers, reports, and other documents that focus on physics and behavior of figures-of-merit and highly ranked phenomena that have dominant role in challenging commercial light water reactor plant scenarios related to plant licensing including: (a) critical flow via the Marviken experiments with measured maximum flow rates of ~14 tons/s;^{10, 32} (b) core liquid level depression during small break loss-of-coolant accidents—ROSA-IV, Bethsy, and Semiscale;^{5,14,30,34} (c) single- and two-phase natural circulation—ROSA-IV and Semiscale;^{26, 31} (d) LOFT core fuel rod thermocouple measurement uncertainties during large break LOCA

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experiments;² (e) passive system behavior including condensation-induced water hammer—ROSA-AP600;^{24,36} (f) code assessment methodologies and protocols for V&V and determining code adequacy in the International Code Assessment and Applications Program for RELAP5 and TRAC-B;^{23,27,28} and (g) best estimate plus uncertainty evaluations of systems analysis codes.^{4,12,26,27,28,33,34}

- Principal Investigator responsible for definition and conduct of the experimental validation experiment program for the Next Generation Nuclear Plant based on NRC historically-accepted code adequacy guidelines.^{16, 18,20, 21,,22,38,39}
 - Co-Inventor of optimally scrambling, passively-driven control rods for gas-cooled reactors.¹³
 - Lead author of complete verification and validation workshop for licensing software.⁵²
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Professional Experience & Career Chronology

2017-present. Consultant, Schultz Engineering PLLC

2014-present Oregon State University, Adjunct Professor, Department of Nuclear Engineering, Corvallis, OR

2012-2019, Texas A&M University, Professor of Practice, Department of Nuclear Engineering, College Station, TX.

2010-present Idaho State University, Pocatello, ID

2014-present. Research Professor, Department of Nuclear Engineering and Health Physics

2010-April, 2014. Adjunct Associate Professor, Graduate Faculty taught advanced thermal-hydraulics and mentored students

1976-April, 2014. Idaho National Laboratory, Idaho Falls, ID—primary projects are listed.

2004-April, 2014. Next Generation Nuclear Plant Technical Lead on NGNP Thermal-Hydraulic Methods and other related projects

1999-2004 Principal Investigator (PI) Project: “Defining an Appendix K Version of RELAP5-3D[®]” Project designed to define the requirements and methodology to create a *Title 10 Code of Federal Regulations*-approved licensing methodology for on-line nuclear power plants. Work performed for the Institute of Nuclear Energy Research (Lung-Tan, Taiwan, Republic of China).

1993-99 PI Project: “ROSA-AP600 Test Planning, Analysis, & Support Project” centered on defining & analyzing 24 experiments at ROSA-AP600, a 1/30-scaled advanced Westinghouse AP600 simulator facility located on Japan Atomic Energy Research Institute campus at Tokai, Japan.

1990-94 PI Project: “BETHSY Experiment Evaluation & Code Assessment”: Analysis of BETHSY experimental data (Commissariat a l’Energie Atomique, Grenoble, France) & application of data to V&V advanced TH codes.

1986-90 PI Project: “International Code Applications and Assessment (ICAP)”: Project centered on defining V&V matrices, analyses, and developmental studies for the RELAP5 and TRAC-BWR advanced TH codes. Work performed by organizations in the member nations: Belgium, European Joint Research Centre, Finland, France, Germany, Italy, Japan, Korea, Republic of China, Slovenia, Spain, Sweden, Switzerland, United Kingdom, and USA. Defined V&V matrices, participated in analyses, and was lead analyst.

1974-76 Research Engineer, United Technology Research Center, E. Hartford, CT—researcher responsible for building, operating experiments, and analyzing the experimental results for combustor-driven chemical lasers.

1972-74 Engineer, General Electric Co (GE), San Jose, CA. Analyzed behavior & licensing envelopes of GE nuclear power plants (boiling water reactors). Responsible for gleaning licensing data from Two Loop Test Apparatus and Vallecitos Core Spray Test Facility.

1968-72 Test Engineer, Pratt & Whitney Aircraft Co, E. Hartford, CT. Gas turbine & Brayton cycle combustion researcher. Experiments focused on characterizing performance of after-burners for TF30 engine (General Dynamics F-111) and main combustors for JT9D engine (Boeing 747). Responsible engineer for full-scale combustion experiments conducted at X-205 test stand at Andrew Wilgoos Gas Turbine Laboratory.

Publications (representative sampling)

1. ASME Presidential Task Force on Response to Japan Nuclear Power Plant Events, 2012, *Forging a New Nuclear Safety Construct*, American Society of Mechanical Engineers, June.
2. Berta, V. T., R. G. Hanson, G. W. Johnsen, and R. R. Schultz, 1993, *Determination of the Bias in LOFT Fuel Peak Cladding Temperature Data from the Blowdown Phase of Large-Break LOCA Experiments*, NUREG/CR-6061.

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3. Boucher, T. J., M. G. Ortiz, D. E. Palmrose, S. M. Modro, R. R. Schultz, D. E. Bessette, and G. S. Rhee, 1995, *Design Modifications of Large Scale Test Facility (LSTF) for AP600 Testing*, INEL-95/0200, July.
4. D'Auria, F., H. Glaeser, S. Lee, J. Mišák, M. Modro, and R. Schultz, 2008, *Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation*, Safety Report Series No. 52, International Atomic Energy Agency, Vienna.
5. Kukita, Y., R. R. Schultz, H. Nakamura, and J. Katayama, 1993, "Quasi-Static Core Liquid Level Depression and Long-Term Core Uncovery During a PWR LOCA," *Nuclear Safety*, 34(1), January-March.
6. Lee, W. J. et al, 2005, *Generation of a Preliminary PIRT for Very High Temperature Gas-Cooled Reactors*, INL/EXT-05-00829, September
7. Liang, T. K. S. and R. R. Schultz, 2001, "Development Program of LOCA Licensing Calculation Capability with RELAP5-3D in Accordance with Appendix K of 10 CFR 50.46," *Nuclear Technology*, 113(3), pp. 355-358.
8. Liou, C. P., D. L. Parks, R. R. Schultz, and B. G. Williams, 2005, "Stratified Flows in Horizontal Piping of Passive Emergency Core Cooling Systems," *Proceedings of International Conference on Nuclear Engineering-15*, Beijing.
9. Liou, C. P., R. R. Schultz, and Y. Kukita, 1997, "Stably Stratified Flows in Closed Conduits," *Proceedings of the 5th International Conference on Nuclear Engineering*, ICONE5-2024, May 25-29, Nice, France.
10. Marviken Multinational Project, 1979, *MXC-301 Summary Report*, MX3-107, Marviken Test Facility, Studsvik, Sweden, December.
11. McCreery, G. E., C. M. Kullberg, R. R. Schultz, T. Yonomoto, and Y. Anoda, 1997, "Heat Transfer Modeling of the LSTF Passive Residual Heat Removal System," *Proceedings of the 1997 International Mechanical Engineering Congress & Exposition*, Dallas, TX, November.
12. Ortiz, M. G., et al, 1995, *Adequacy Evaluation of RELAP5/MOD3 for Simulating AP600 Small Break Loss-of-Coolant Accidents, Volume 2: Horizontal Integrated Analysis of the AP600 1-Inch Diameter Cold Leg Break*, November.
13. Ougouag, A. M., R. R. Schultz, and W. K. Terry, 2001, *Invention Disclosure Record: "Optimally Scramming Control Rod for Gas Cooled Reactors and Self Regulating Mechanism for Devices Requiring the Existence of a Flow of Coolant or Working Fluid*, September.
14. Roth, P. A., and R. R. Schultz, 1991, *Analysis of Reduced Primary and Secondary Coolant Level Experiments in the BETHSY Facility Using RELAP5/MOD3*, EGG-EAST-9251, July.
15. Schultz, R. R., 1994 to 1997, *ROSA-AP600 Test Specification Documents* (specifications of all ROSA-AP600 experiments—set of 24 reports)
16. Sato, H, R. W. Johnson, and R. R. Schultz, 2010, "Computational Fluid Dynamic Analysis of Core Bypass Flow Phenomena in a Prismatic VHTR," *Annals of Nuclear Energy Journal*
17. Schultz, R. R., M. Holbrook, W. Moe, and G. Griffith, 2014, *Nuclear Energy System Options for Utah's Energy Future*, INL-EXT-14-31074, March
18. Schultz, R. R., 2012, "Using CFD to Analyze Nuclear Systems Behavior: Defining the Validation Requirements," *Proceedings of the Experimental Validation and Application of CFD and CMFD Codes in Nuclear Reactor Technology Workshop*, September 10-12, Daejeon, Korea.
19. Schultz, R. R., 2010, *Using Stratified Flow to Mitigate Condensation-Induced Water Hammer*, Idaho State University Ph.D dissertation.
20. Schultz, R. R., E. Harvego, R. Crane, "Development of a Standard for Verification and Validation of Software to Calculate Nuclear System Thermal Fluid Behavior," *Mechanical Engineering*, 2010, 132 (5), 56-
21. Schultz, R. R., Y. Hassan, R. W. Johnson, H. McIlroy, and R. Vilim, 2009, *Experimental and Analytic Studies on the Core Bypass Flow in a Very High Temperature Reactor*, INL/EXT-09-16745, September.
22. Schultz, R. R. et al, 2008, *Next Generation Nuclear Plant Methods Technical Program Plan*, INL/EXT-06-11804, Oct.
23. Schultz, R. R., 2005, *RELAP5/MOD3 Code Manual User's Guidelines*, NUREG/CR-5535, EGG-2596, Vol. 5, January. Original authors C. D. Fletcher and R. R. Schultz
24. Schultz, R. R. , M. Kondo, and Y. Anoda, 2001, "Baseline Study to Model a Typical Condensation-Induced Water Hammer Event Measured at the Two-Phase Flow Test Facility (TPTF) in Japan," *Proceedings of the Pressure Vessel & Piping Conference*, Atlanta, July 22-26.
25. Schultz, R. R. and C. Atwood, 1999, "Using Latin Hypercube Sampling to Improve Best Estimate Plus Uncertainty Techniques," *Proceedings of the American Society of Mechanical Engineers Fluids Engineering Division's Summer Meeting*, San Francisco.

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26. Schultz, R. R., C. M. Kullberg, G. E. McCreery, R. A. Shaw, B. Hanson, N. Newman, C. P. Liou, and J. L. Westcott, 1997, *RELAP5/MOD3 Code Assessment Analyses Based on the ROSA-AP600 Program: Small Break LOCAs and the Station Blackout Transient*, March.
27. Schultz, Richard R., 1994, USRNC AP600-Related Facilities RELAP5 Models Nodalization Consistency, EGG-NRC-11515.
28. Schultz, R. R., 1993, *International Code Assessment and Applications Program: Summary of Code Assessment Studies Concerning RELAP5/MOD2, RELAP5/MOD3, and TRAC-B*, Idaho National Engineering Laboratory, NUREG/IA-0128, EGG-EAST-8719, December.
29. Schultz, R. R., 1992, "Methodology for Quantifying Computational Capability of RELAP5/MOD3 Code for SBLOCAs, LBLOCAs, and Operational Transients," *Proceedings of CAMP Meeting*, Villigen, Switzerland.
30. Schultz, R. R., et al, 1991, *An Investigation of Core Liquid Level Depression in Small Break Loss-of-Coolant Accidents*, NUREG/CR-4063, EGG-2636, August.
31. Schultz, R. R., J. C. Chapman, Y. Kukita, F. E. Motley, H. Stumpf, Y. S. Chen, and K. Tasaka, 1987, "Single and Two-Phase Natural Circulation in Westinghouse Pressurized Water Reactor Simulators: Phenomena, Analysis, and Scaling," *Natural Circulation, Proceedings of the Winter Annual Meeting of the American Society of Mechanical Engineer*, Boston, MA, FED-Vol. 61, HTD-Vol. 92, December.
32. Schultz, R. R., O. Sandervag, and R. G. Hanson, 1984, "Marviken Power Station Critical Flow Data: A Summary of Results and Code Assessment Applications," *Nuclear Safety*, 25(6), pp. 770-783, November-December.
33. Shaw, R. A., R. R. Schultz, and C. M. Kullberg, 1997, *RELAP5/MOD3 Code Assessment Analyses Based on the ROSA-AP600 Program Steam Generator Tube Rupture (SGTR) Tests*, March.
34. Stumpf, H., F. Motley, R. R. Schultz, J. C. Chapman, and Y. Kukita, 1987, "Reverse Primary-Side Flow in Steam Generators During Natural Circulation Cooling," *Natural Circulation, Proceedings of the Winter Annual Meeting of the American Society of Mechanical Engineer*, Boston, MA, FED-Vol. 61, HTD-Vol. 92, December.
35. USNRC, 2008, *Next Generation Nuclear Plant Phenomena and Ranking Tables (PIRTs)*, NUREG/CR-6944, March.
36. Yonomoto, T., H. Nakamura, M. Suzuki, H. Asaka, M. Kondo, I. Ohtsu, Y. Shibamoto, Y. Kukita, R. R. Schultz, G. E. McCreery, J. M. Cozzuol, C. P. Liou, and G. Rhee, 2001, *Summary Report ROSA-AP600 Program*, JAERI-memo 13-009.

Publications from 2010 to present: see references 1, 16 to 20, and 37 to 51

37. Schultz, R. R., et al, 2010, *Next Generation Nuclear Plant Methods Technical Program Plan*—PLN-2498, November
38. Oh, C, E. Kim, R. Schultz, M. Patterson, D. Petti, H. Kang, "Comprehensive Thermal-Hydraulics Research of the Very High Temperature Gas-Cooled Reactor," *Nuclear Engineering and Design*, 2010, 240 (10), 3361-3371.
39. Schultz, R. R., P D Bayless, R W Johnson, W T Taitano, J R Wolf, G E McCreery, 2010, *Studies Related to the Oregon State University High Temperature Test Facility: Scaling, the Validation Matrix, and Similarities to the Modular High Temperature Gas-Cooled Reactor*, INL/EXT-10-19803.
40. NGNP Moisture Ingress Committee, 2011, Assessment of NGNP Moisture Ingress Events, INL/EXT-11-21397, April.
41. Schultz, R. R., *Experimental and Analytic Study on the Core Bypass Flow in a Very High Temperature Reactor*, 2012, INL/EXT-12-24603.
42. Stoots, C., T. Larson, R. R. Schultz, H. Gougar, K. McCarthy, D. Petti, L. Swiler, and M. Corradini, *Verification and Validation Strategy for LWRS Tools*, 2012, INL/EXT-12-27066.
43. Liou, C. P. and R. R. Schultz, Wave Propagation in the Hot Duct of a VHTR, 2013, *Proceedings of the International Conference on Nuclear Engineering—21*, Chengdu, China.
44. Kawaji, M., F. I. Valentin, N. Artoun, S. Banerjee, M. Sohal, R. R. Schultz, and D. M. McEligot, *Experimental Investigation of Convective and Heat Transfer in the Reactor Core for a VHTR*, NEUP Project No: 11-3218, 2015, December.
45. Williams, B., R. R. Schultz, D. M. McEligot, G. E. McCreery, *Studies of Deteriorated Heat Transfer in Prismatic Cores Stemming from Irradiation-Induced Geometry Distortion*, 2015, DOE/NEUP-10-876.
46. Schultz, R. R., H. Gougar, P. Vagendla, A. Obabko, and J. Thomas, 2017, Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Scenarios in Modular High Temperature Gas-Cooled Reactors, INL/EXT-17-43218, Rev 0.
47. Schultz, R. R., "The Role of Thermal-Hydraulics in Nuclear Power Plant Design and Safety," *Thermal-Hydraulics in Water Cooled Thermal Reactors*, Elsevier, 2017.

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48. Harvego, E. and R. R. Schultz, 2017, "Generation IV Reactors," *Nuclear Engineering Handbook*, CRC Press..
49. Schultz, R. R. and K. Kok, 2017, "Small Modular Reactors, *Nuclear Engineering Handbook*, CRC Press.
50. Schultz, R. R., H. Gougar, L. Lommers, 2018, "Identification and Characterization of Thermal Fluid Phenomena Associated with Selected Scenarios in Modular High Temperature Gas-Cooled Reactors, *Proceeding of the High Temperature Reactor Conference*, Warsaw, Poland, October 8-10.
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