NuScale LOCA PIRT

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February 29, 2012

U.S. Nuclear Regulatory Commission
Pre-Application Meeting
Rockville, MD

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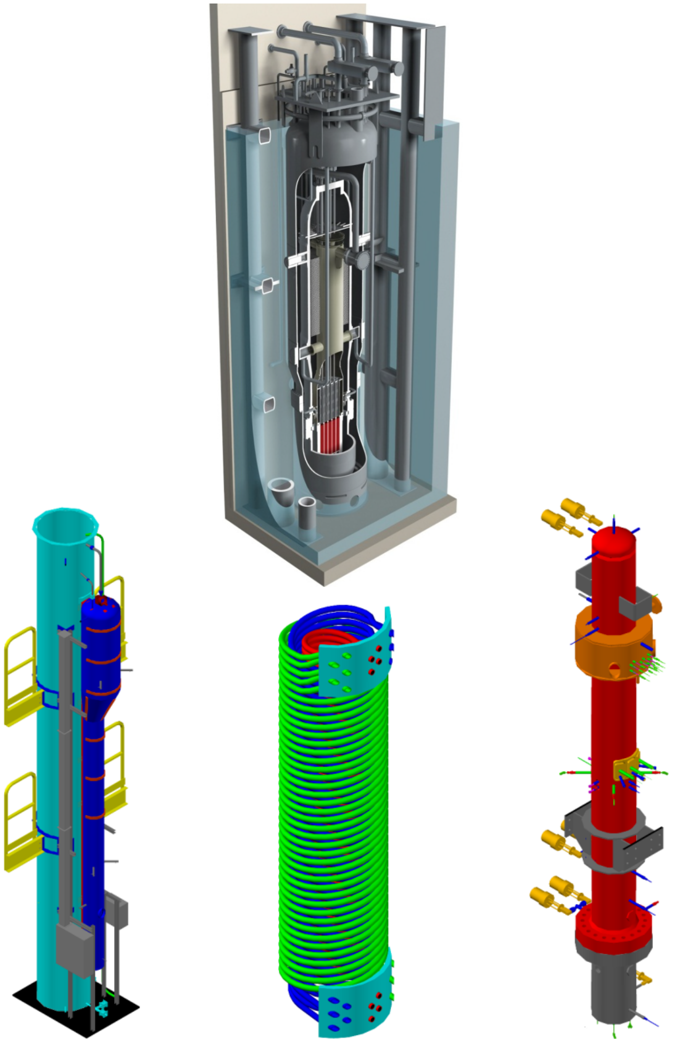
Outline

- Introduction
- EMDAP
- PIRT Process
- NuScale LOCA PIRT
- Summary
- Conclusions



Introduction

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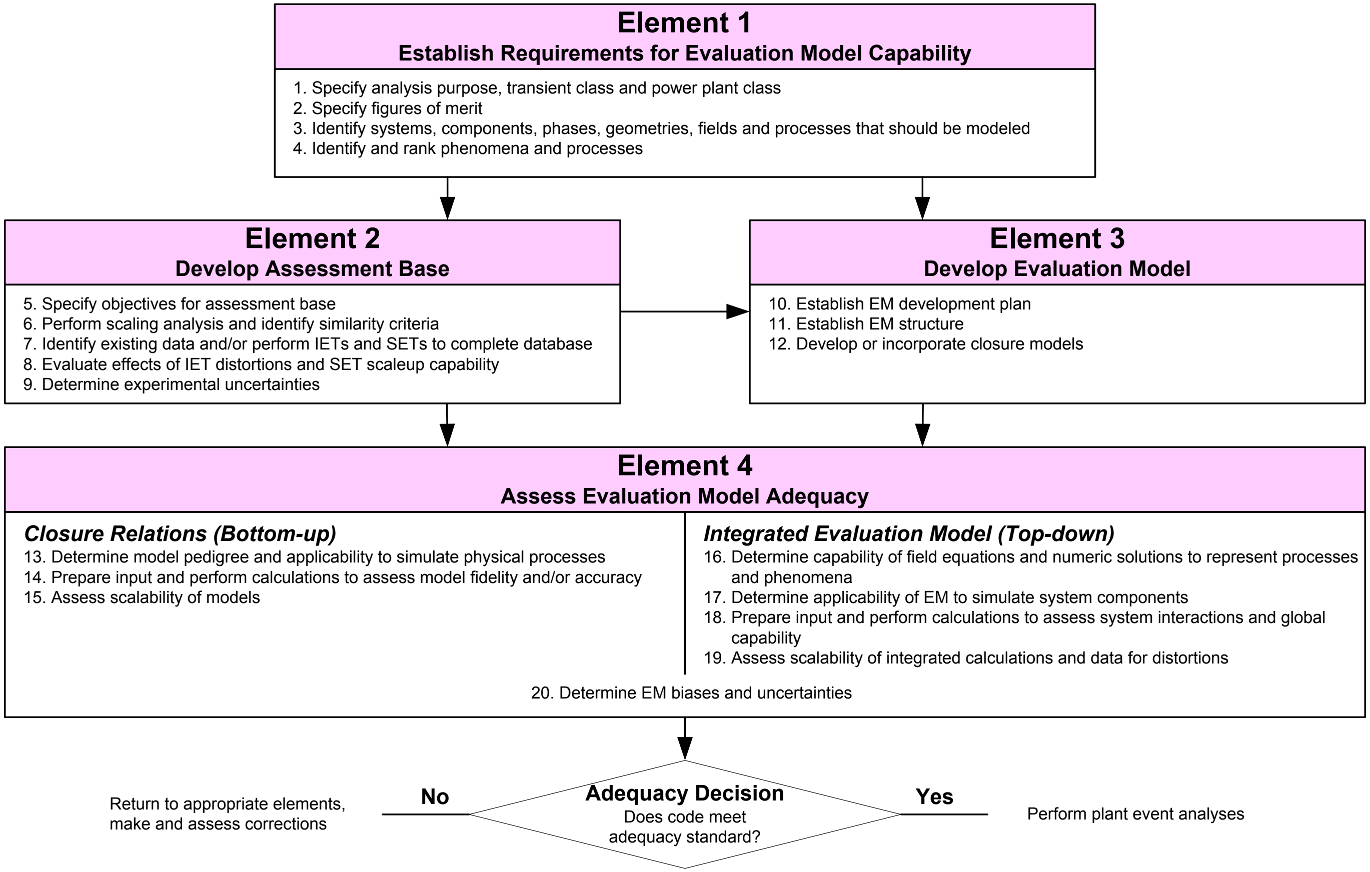
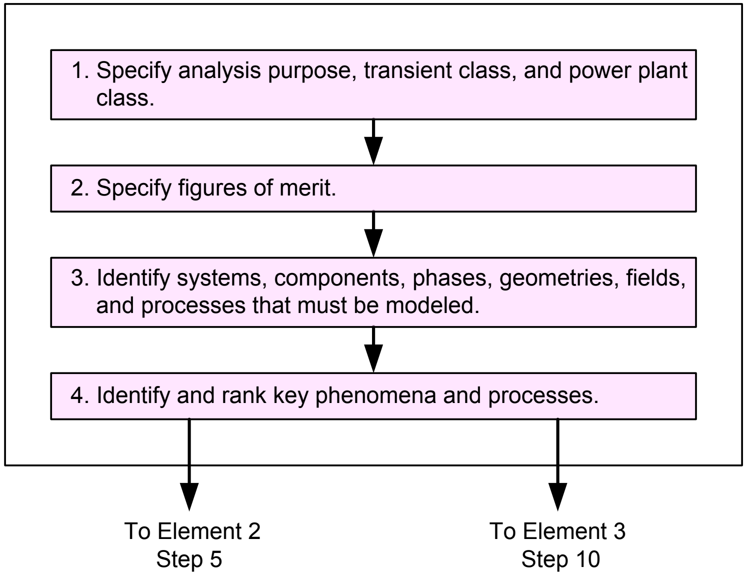
EMDAP



Evaluation Model Development and Assessment Process (EMDAP)[1]

[1] Regulatory Guide 1.203, "Transient and Accident Analysis Methods," U.S. Nuclear Regulatory Commission, December 2005.

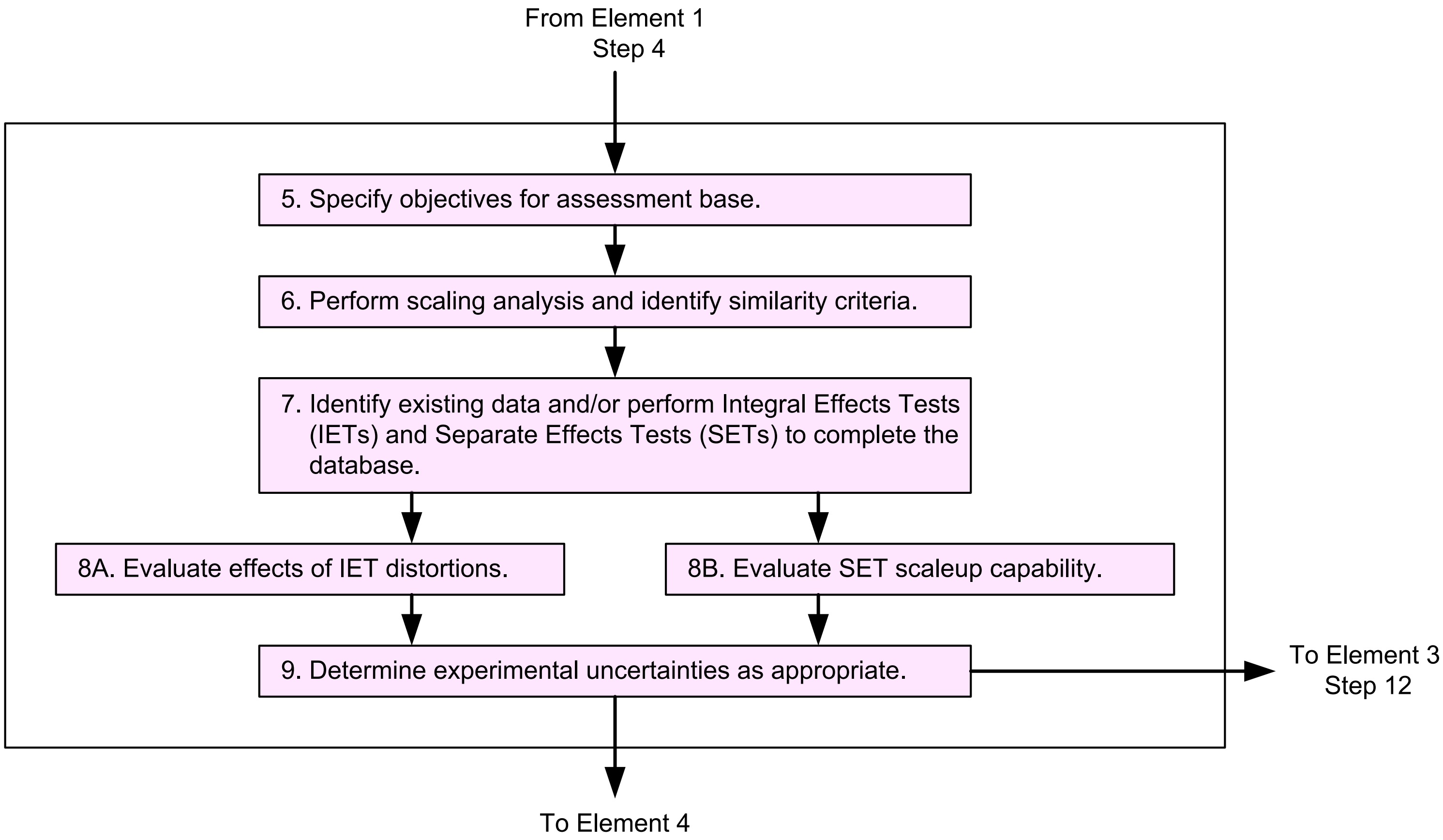
EMDAP – Element 1



EMDAP - Element 1: Establish Requirements for Evaluation Model Capability[1]

[1] Regulatory Guide 1.203, "Transient and Accident Analysis Methods," U.S. Nuclear Regulatory Commission, December 2005.

EMDAP – Element 2



EMDAP - Element 2: Develop Assessment Base[1]

[1] Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” U.S. Nuclear Regulatory Commission, December 2005.

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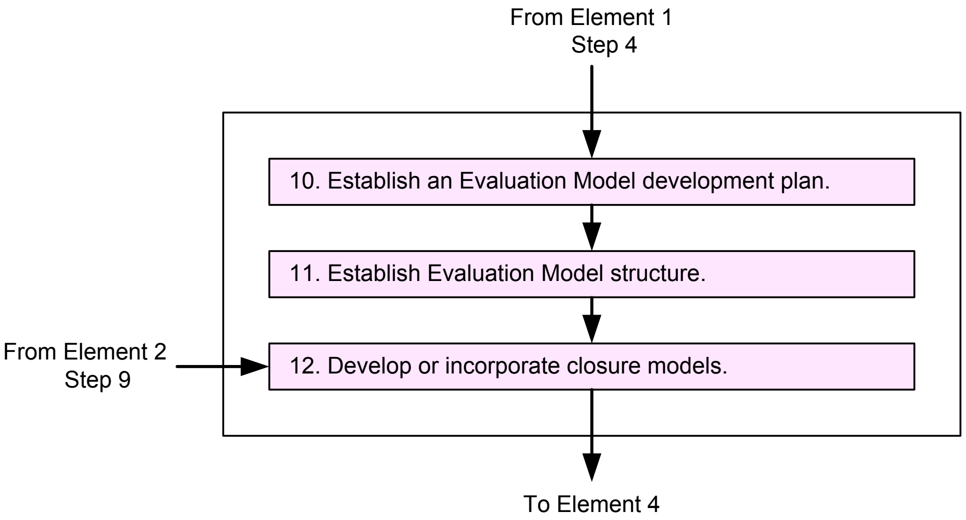
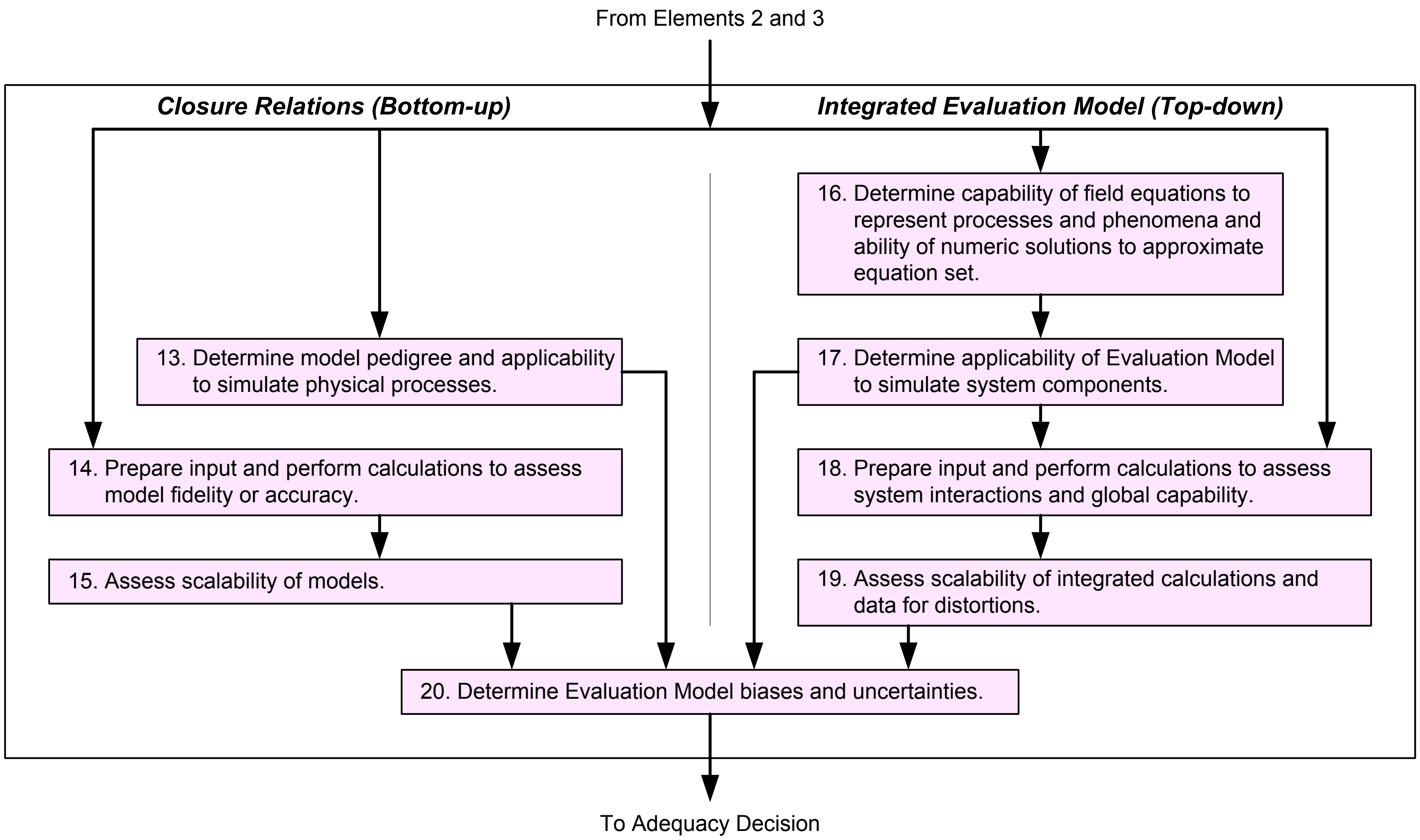
EMDAP – Element 3



EMDAP - Element 3: Develop Evaluation Model[1]

[1] Regulatory Guide 1.203, "Transient and Accident Analysis Methods," U.S. Nuclear Regulatory Commission, December 2005.

EMDAP – Element 4



EMDAP - Element 4: Assess Evaluation Model Adequacy[1]

[1] Regulatory Guide 1.203, “Transient and Accident Analysis Methods,” U.S. Nuclear Regulatory Commission, December 2005.

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PIRT

- EMDAP – Element 1 – Step 4
- **P**henomena **I**dentification and **R**anking **T**able (**PIRT**)
- LOCA PIRT[2] for NuScale

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PIRT Process

Step 1: Issue	Define issue driving the need for a PIRT
Step 2: Objectives	Define the specific objectives of the PIRT
Step 3: Database	Compile and review background information that captures relevant knowledge
Step 4: Hardware and Scenario	Specify plant and components; divide scenario into phases
Step 5: Figure of Merit	Select key figure of merit used to judge importance
Step 6: Phenomena Identification	Identify all plausible phenomena plus definitions
Step 7: Importance Ranking	Assign importance relative to figure of merit; document rationale
Step 8: Knowledge Level Ranking	Assess current level of knowledge regarding each phenomenon
Step 9: Documentation	Document the PIRT with sufficient coverage that knowledgeable reader can understand process and outcome

PIRT Process[3]

[3] Brent E. Boyack and Gary E. Wilson, "Lessons Learned in Obtaining Efficient and Sufficient Applications of the PIRT Process," Best Estimates 2004, Washington, D.C., November 14-18, 2004.

Step 1: Issue

NuScale PIRT effort is required to:

- Foster understanding of new reactor design.
- Support licensing and regulation efforts:
  - Applicability and adequacy of computational tools,
  - Applicability and sufficiency of experimental facilities.
- Promote more cost effective resource allocation.

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Step 2: Objectives

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Step 3: Database

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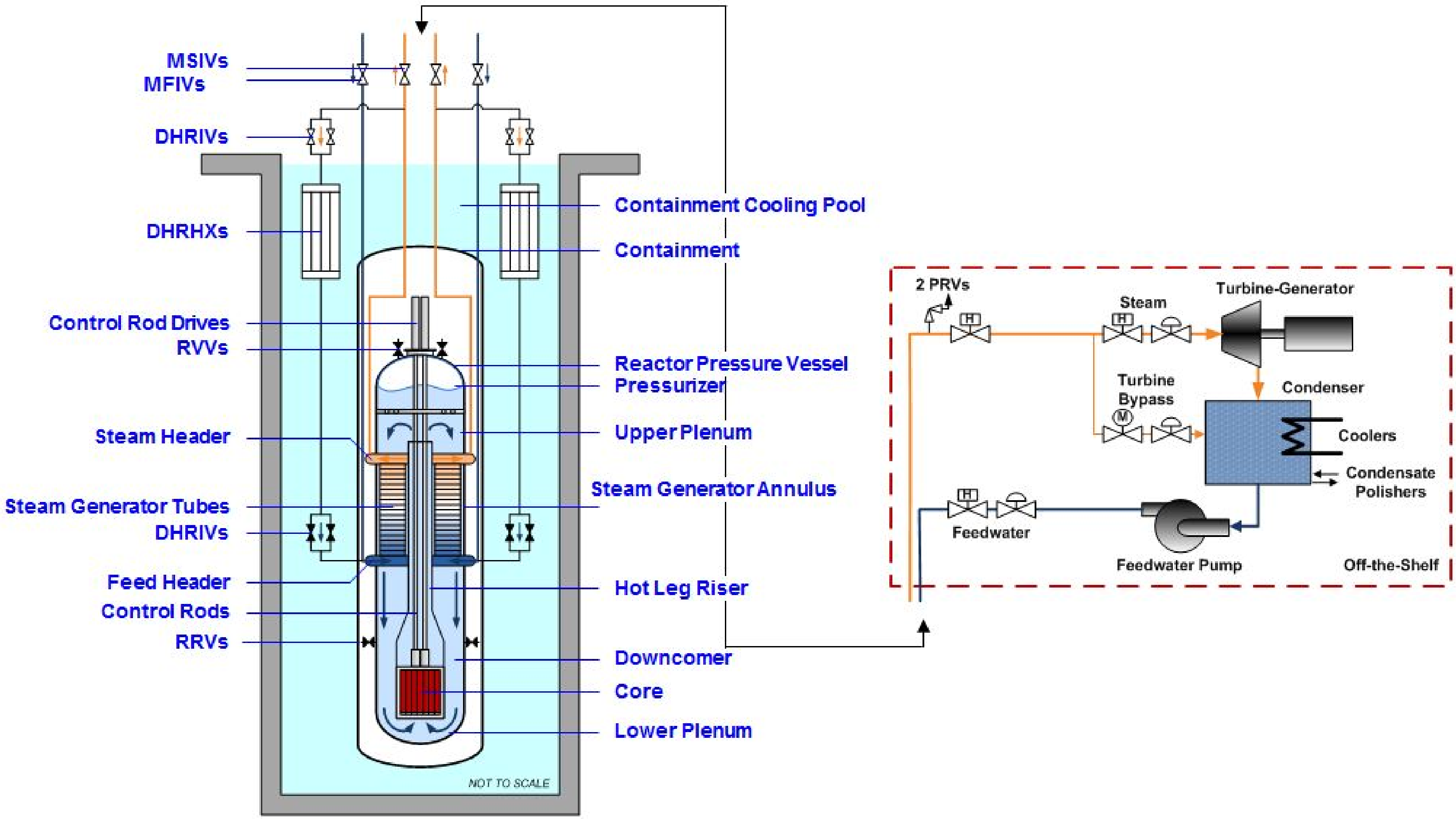
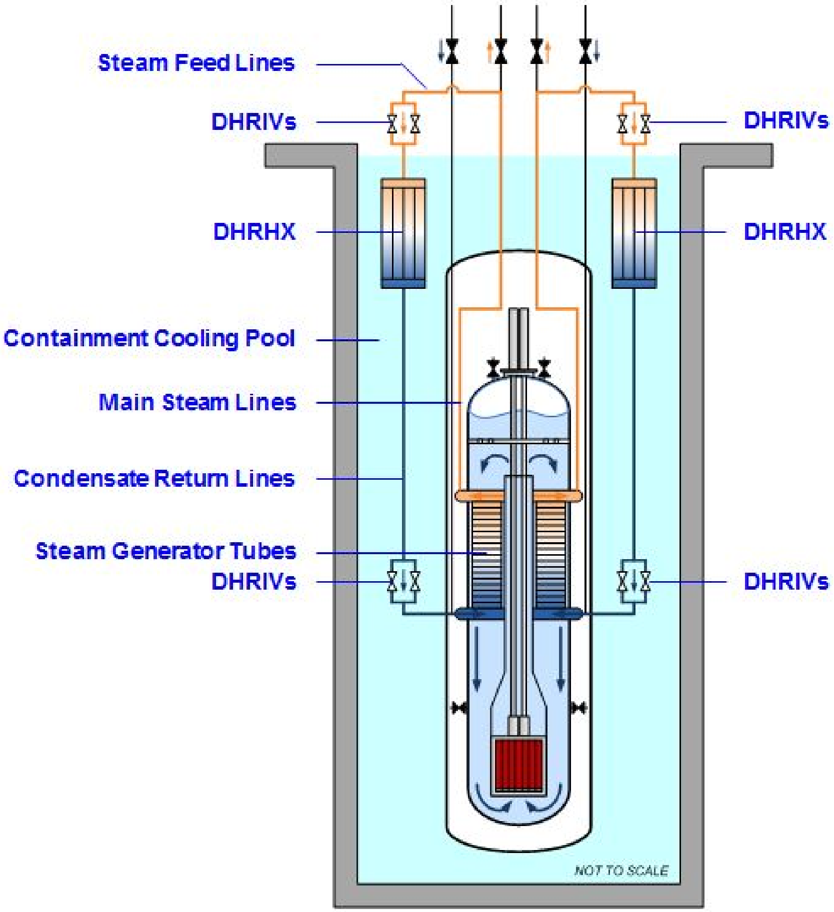
Step 4: Hardware and Scenario

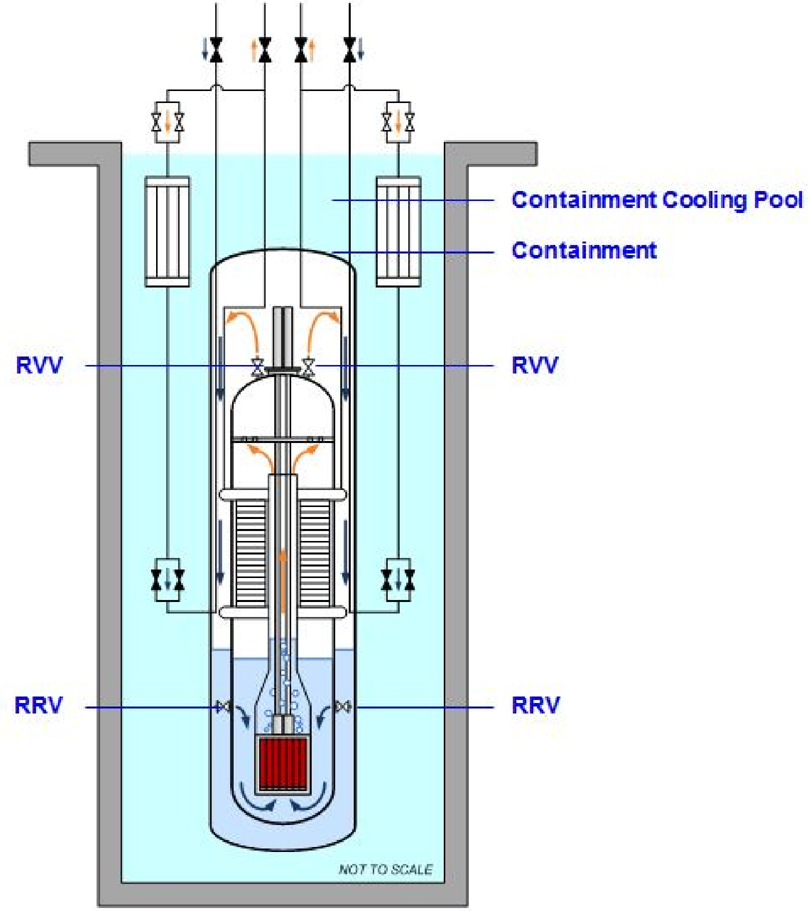


NuScale Power Module

Step 4: Hardware and Scenario



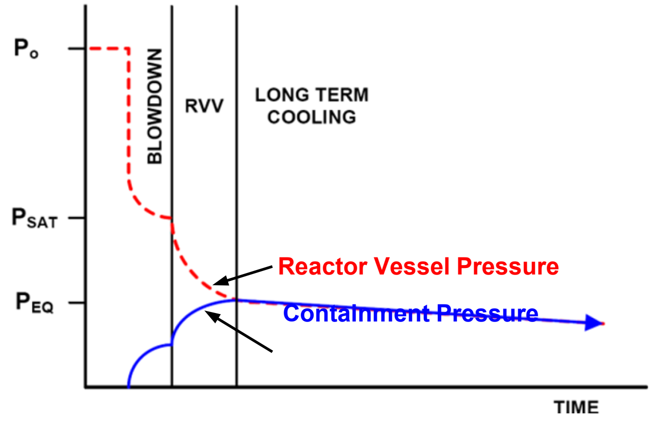
DHRS



ECCS (Including CHRS)

Step 4: Hardware and Scenario

NuScale **SBLOCA**



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Step 4: Hardware and Scenario

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Step 4: Hardware and Scenario

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Step 4: Hardware and Scenario

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Step 5: Figure of Merit

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Step 6: Phenomena Identification

The standard approach in this step:

- Is based upon collective experience of the panelists.
- Is informed by the background information of Step 3.
- Works best if the panel is highly knowledgeable in:
  - Design,
  - Processes and phenomena occurring during accidents.
- Focuses on (generally during the PIRT meetings):
  - List of potentially active systems,
  - List of relevant components of each system,
  - Identification of processes and/or phenomena in each component.
- Prohibits phenomena evaluation (ranking) during the identification step.
- Provides precise, written definition of each phenomenon to:
  - Help ensure each panelist has the same definition in mind when ranking the phenomenon (Step 7),
  - Reduce inconsistencies.

[3] Brent E. Boyack and Gary E. Wilson, "Lessons Learned in Obtaining Efficient and Sufficient Applications of the PIRT Process," Best Estimates 2004, Washington, D.C., November 14-18, 2004.

Step 7: Importance Ranking

Importance	Definition	Application Outcomes
High (H)	Phenomenon has controlling impact on figure of merit.	Experimental simulation and analytical modeling with a high degree of accuracy is critical.
Medium (M)	Phenomenon has moderate impact on figure of merit.	Experimental simulation and/or analytical modeling with a moderate degree of accuracy is required.
Low (L)	Phenomenon has low impact on figure of merit.	Modeling must be present only to preserve functional dependencies.
Insignificant (I)	Phenomenon has no, or insignificant impact on figure of merit.	Modeling must be present only if functional dependencies are required.

[3] Brent E. Boyack and Gary E. Wilson, "Lessons Learned in Obtaining Efficient and Sufficient Applications of the PIRT Process," Best Estimates 2004, Washington, D.C., November 14-18, 2004.

Step 8: Knowledge Level Ranking

Knowledge Level	Definition
4	Fully known, small uncertainty
3	Known, moderate uncertainty
2	Partially known, large uncertainty
1	Very limited knowledge, uncertainty cannot be characterized

[3] Brent E. Boyack and Gary E. Wilson, “Lessons Learned in Obtaining Efficient and Sufficient Applications of the PIRT Process,” Best Estimates 2004, Washington, D.C., November 14-18, 2004.

Step 9: Documentation

- Introduction
- PIRT Method Description
- Background Information (Database)
- Plant Description
- Scenario
- PIRT Results (essentially summary tables)
- Appendices
  - Brief Biographies of PIRT participants
  - Importance ranks and rationales
  - Knowledge levels and rationales

Panelists

Old PIRT	New PIRT
Dr. Graham Wallis, Chair	Dr. Graham Wallis, Chair
Dr. Lawrence Hochreiter	Mr. Steve Congdon
Dr. Mujid Kazimi	Dr. Tom George
Mr. Brent Boyack	Mr. Craig Peterson
Dr. Kord Smith	Mr. Gregg Swindlehurst
Dr. José N. Reyes, Jr.	Dr. José N. Reyes, Jr.
Dr. Kent Welter, Facilitator	Dr. Kent Welter, Facilitator
Dr. Eric Young, Assistant	Mr. Tristan McDonald, Assistant

Panelists

Panelist	Biography
Dr. Graham Wallis, Chair	Sherman Fairchild Professor of Engineering, Emeritus, Thayer School of Engineering at Dartmouth College; Served as a member of the Thayer School faculty from 1962 to 2001; Received the 1994 ASME Fluids Engineering Award; Author of the book "One-Dimensional Two-Phase Flow" and of over 150 publications and reports on aspects of two-phase flow; Expertise in fluid dynamics, two-phase flow, thermodynamics, heat and mass transfer, nuclear power, and energy conversion.
The Late (1941-2008) Dr. Lawrence Hochreiter	Was a Professor of Mechanical and Nuclear Engineering at Penn State University (1986-2008); Worked for Westinghouse (1971-1996); ASME Fellow (1974-2008) and a member of ASME's K-13 committee; Authored and coauthored over 200 publications in journals, transactions, and proceedings, and over 80 Westinghouse reports; Expertise in thermal-hydraulic modeling of nuclear power plants, reactor safety analysis, and experimental studies of two-phase flow and heat transfer.
Dr. Mujid Kazimi	Professor of Mechanical Engineering, TEPCO Professor of Nuclear Engineering, and Director of Center for Advanced Nuclear Energy Systems at MIT; Fellow of the International Nuclear Energy Academy, the American Association for the Advancement of Science, and the American Nuclear Society; Published over 200 journal and conference articles and is co-author of the two-volume work "Nuclear Systems"; Expertise in nuclear systems safety, nuclear fuel cycle, two-phase flow and heat transfer.
Mr. Brent Boyack	Consultant at Brent E Boyack in Los Alamos, NM; Retired from LANL; Led the pioneering PIRT effort as part of the Code Scaling, Applicability, and Uncertainty (CSAU) project; Was the Principal Investigator directing the TRAC-M code development effort at LANL for many years; Expertise on employing the PIRT process.
Dr. Kord Smith	Vice-President of Technical Development for Studsvik Scandpower; Led the development of SIMULATE-3, now the industry standard for light water reactor core analysis; Developed several key physics methods and engineering features now found in SIMULATE and other Studsvik nuclear reactor analysis products; Published over 100 technical papers for conferences and archival publications, including many ground-breaking numerical techniques.

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Panelists

Panelist	Biography
Mr. Steve Congdon	Retired from GE Nuclear Energy, Wilmington, NC in 2000 where he served as Manager of Advanced Engineering and Manager of Fuel Technology; 2001-2011 served as rehired pensioner at GE Nuclear Energy, providing assistance in BWR plant monitoring technology, qualification of GE nuclear and thermal hydraulic methods and writing NRC licensing reports supporting GE Power Uprate activities.
Dr. Tom George	Senior Consulting Engineer, Numerical Applications Division Zachry Nuclear Engineering; Dr. George specializes in numerical modeling in engineering mechanics; Primary developer for the GOTHIC code for general-purpose thermal-hydraulic analysis; Participated in wide range of safety analyses using GOTHIC to support vender and utility licensing efforts. Developed a computer program to track smoke propagation and fire growth in buildings; Developed a mechanical response model that predicts the elastic and plastic properties of restructuring sphere pac nuclear fuel and the stress-strain distribution in the fuel and cladding; Developed models for heat and mass transfer; Developed numerical techniques for thermal-hydraulic analysis in three-dimensional curvilinear or generalized coordinate systems for complex geometries."
Mr. Craig Peterson	President of Computer Simulation and Analysis, Inc. in Idaho Falls, ID since 1999; Former Nuclear Safety Analysis and Software Applications Manager at Energy Incorporated; Over 30 years experience in commercial nuclear power industry working on a wide variety of projects for numerous organizations.
Mr. Gregg B. Swindlehurst	Consultant at GS Nuclear Consulting, LLC in Charlotte, NC; Former Safety Analysis Manager Duke Power; 24 years of supervision/management in electric utility nuclear engineering organizations; 30 years of experience with pressurized water reactor transient and accident analysis methodology development and application using RETRAN, RELAP5, VIPRE, GOTHIC, SIMULATE, and FALCON codes; Served on the McGuire and Catawba corporate Nuclear Safety Review Boards and the subcommittee that reviews all license amendment requests prior to submittal for NRC review; Pioneered the use of engineering-quality transient and accident and containment analysis codes for validating training simulator software.

Panelists

Panelist	Biography
Dr. José N. Reyes, Jr.	Chief Technology Officer at NuScale Power; Co-designer of the NuScale passively-cooled small nuclear reactor; Schuette Endowed Chair Professor and Head of the Department of Nuclear Engineering and Radiation Health Physics at Oregon State University; An internationally recognized expert on passive safety system design, testing and operations for nuclear power plants; Worked nearly 10 years for the Reactor Safety Division of the U.S. Nuclear Regulatory Commission.
Dr. Eric Young, Assistant	Testing & Demonstration Supervisor at NuScale Power; Worked for over 10 years on thermal hydraulics research, LBLOCA safety analysis, and thermal-mechanical test program management at Oregon State University, AREVA, and NuScale Power; Specialized in LOCA safety analysis, passive safety system design, and testing for nuclear power plants.
Mr. Tristan McDonald, Assistant	Safety Analyst at NuScale Power in Thermal Hydraulics Codes and Methods (I'm not sure if that is the proper name, maybe check with Kent); former Nuclear Engineer I at Fort Calhoun Nuclear Station in Core Design and Safety analysis; Past Vice President North American Young Generation in Nuclear, Fort Calhoun Station chapter; Bachelor of Science in Nuclear Engineering from University of New Mexico; Past Vice President American Nuclear Society, University of New Mexico Chapter; Former member of subcommittee E28.02 on fracture toughness testing American Society of Testing and Materials (ASTM).

Design Changes

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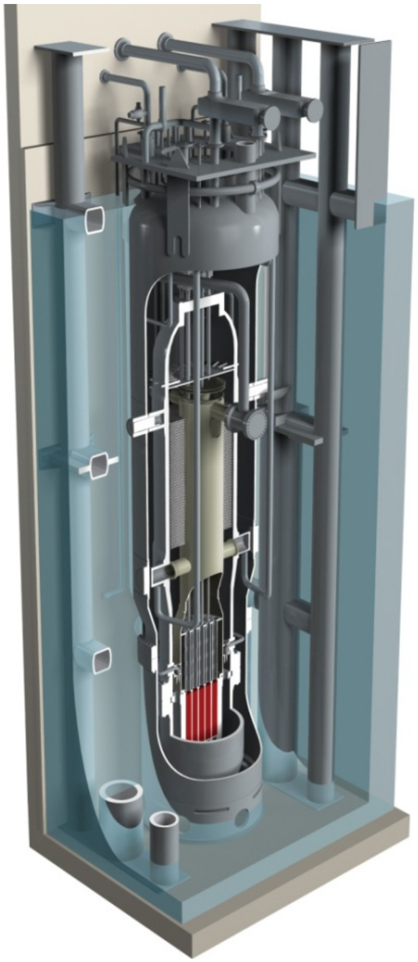
NuScale LOCA PIRT



Systems and Components

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Core

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Core – Fuel Rods

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Core – Subchannel / Coolant Flow

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Core – Others

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Primary – Hot Leg Riser

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Primary – Hot Leg Riser

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Primary – Upper Plenum

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Primary – Upper Plenum

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Primary – Pressurizer

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Primary – Steam Generator Annulus

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Primary – Downcomer

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Primary – Lower Plenum

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Emergency Core Cooling System

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Summary

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Conclusions

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