Core Thermal Hydraulics

SCANR

Code Description

- Sub-Channel Analyzer for NuScale Reactors (SCANR)
- Calculates the enthalpy and flow distributions in rod bundle nuclear fuel elements and cores under single- and two-phase flow conditions for both steady-state and transient simulations
- Minimum departure from nucleate boiling ratio (MDNBR) calculation using local thermal-hydraulic conditions
- Quasi-transient, quasi-three-dimensional, homogeneous equilibrium two-phase mixture models
- Modularized and modernized code structure using Fortran 90/95
- Input/output unit conversions, MDNBR iterations on operating parameters, and automatic generation of subchannel/lumped channel models

SCANR

Analysis Requirements

Minimum DNBR (MDNBR)

- MDNBR = Min (DNBRs throughout Reactor Core)

Departure from Nucleate Boiling Ratio (DNBR)

- DNBR = CHF/AHF

Actual Heat Flux (AHF)

- AHF = f (T_w, T_b, h from Reactor Core Thermal-Hydraulic Analysis)

Critical Heat Flux (CHF)

- CHF = f (x, G from Reactor Core Thermal-Hydraulic Analysis;
  p from System Thermal-Hydraulic Analysis;
  Geometrical Dimensions from Fuel Design)

Reactor Core Thermal-Hydraulic Analysis Using a Subchannel Analysis Code

Output

- Local Fuel Rod Surface Temperature (T_w) from Fuel Rod Conduction Equation
- Local Coolant Bulk Temperature (T_b) from Coolant Energy Equation
- Local Heat Transfer Coefficient (h) from Regime Dependent Correlations
- Local Thermodynamic Equilibrium Quality (x) from Coolant Energy Equation
- Local Mass Flux (G) from Coolant Continuity and Momentum Equations

Input

- Core Axial/Radial Power Distributions from Nuclear Analysis
- Core Power from System Thermal-Hydraulic Analysis
- Core Inlet Temperature from System Thermal-Hydraulic Analysis
- Core Inlet Mass Flux from System Thermal-Hydraulic Analysis
- System Pressure (p) from System Thermal-Hydraulic Analysis
- Geometrical Dimensions from Fuel and System Designs

MDNBR Calculation Process

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Physical Models

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- Critical heat flux models
  - Rod bundle and round tube
- Heat transfer models
  - Heat transfer regime dependent
- Pressure drop models
  - Friction factor
  - Form loss coefficient
  - Two-phase multiplier
- Void fraction models
  - Subcooled flow quality
  - Void fraction
- Inter-channel interaction models
  - Turbulent mixing
  - Void drift
- Water/steam properties
  - IAPWS-IF97
- Fuel/gap/clad properties
  - Temperature and burnup dependent

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Numerical Models

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- Coolant subchannels
  - High flow solution scheme
    - Early COBRA series of codes (COBRA-IIIC)
  - Low flow solution scheme
    - Later COBRA series of codes (COBRA-IV-I, VIPRE-01)
- Fuel rods
  - Fully implicit
    Finite difference scheme
    - Stable for all time steps
  - Solution by
    Gaussian Elimination

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Validation Plan

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Validation Matrix Example

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Design Implementation

- CHF correlation development by providing local thermal-hydraulic conditions
- DNB analyses for normal operation, anticipated operational occurrences, and postulated accidents
- Boundary conditions generation for fuel performance analyses

SCANR

Code Interfaces

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System Thermal Hydraulics

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System Thermal Hydraulic Code Selection Process

Key Requirements

- will be used by NuScale to perform transient and accident analyses (Ch. 15) and containment analyses (Sec. 6.2)
- needs to accurately model natural circulation
- provide coupled containment and reactor pressure vessel simulation
- enable evaluation of important 3-dimensional effects
- include high fidelity containment/condensation modeling
- have an existing quality pedigree and easy to maintain

Codes Evaluated

- TRACE
- RELAP5
- RETRAN
- GOTHIC

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Code Selection Results

- GOTHIC selected as licensing basis code for systems thermal hydraulics
- Primary reasons were quality pedigree, range of assessment base, and modeling fidelity (3D)
- Plan to verify and validate GOTHIC for LOCA, non-LOCA, and containment performance

GOTHIC

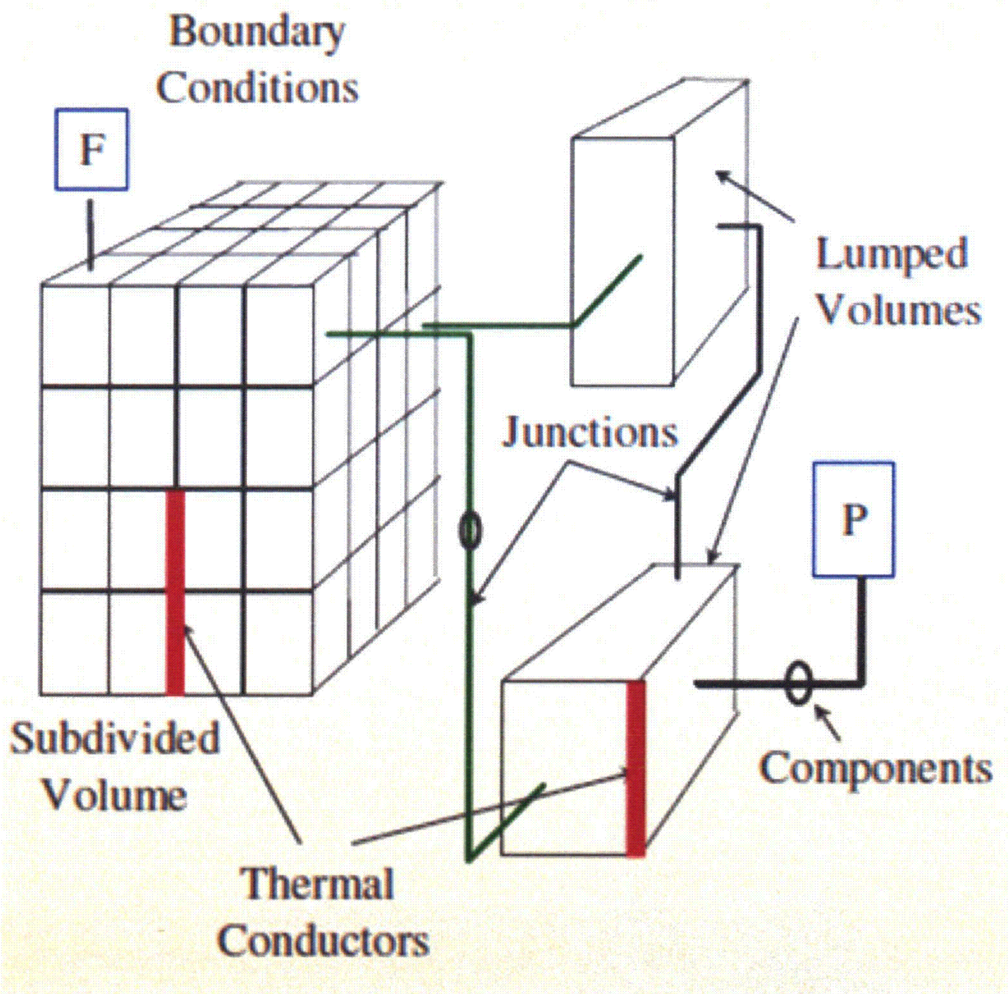
Code Description

- A general purpose thermal-hydraulic analysis code
- Porous media approach for transient, multi-dimensional, multi-component, multi-phase flow systems in complex geometries
- Key Features
  - condensation and film heat transfer
  - 3-D fluid flow behavior
  - subdivided modeling
  - comprehensive drop behavior models
  - mixing and stratification models
  - variable porosity
  - equipment models: pumps, valves, heat exchangers, fan coolers, etc.

GOTHIC

Multi-Zone Modeling

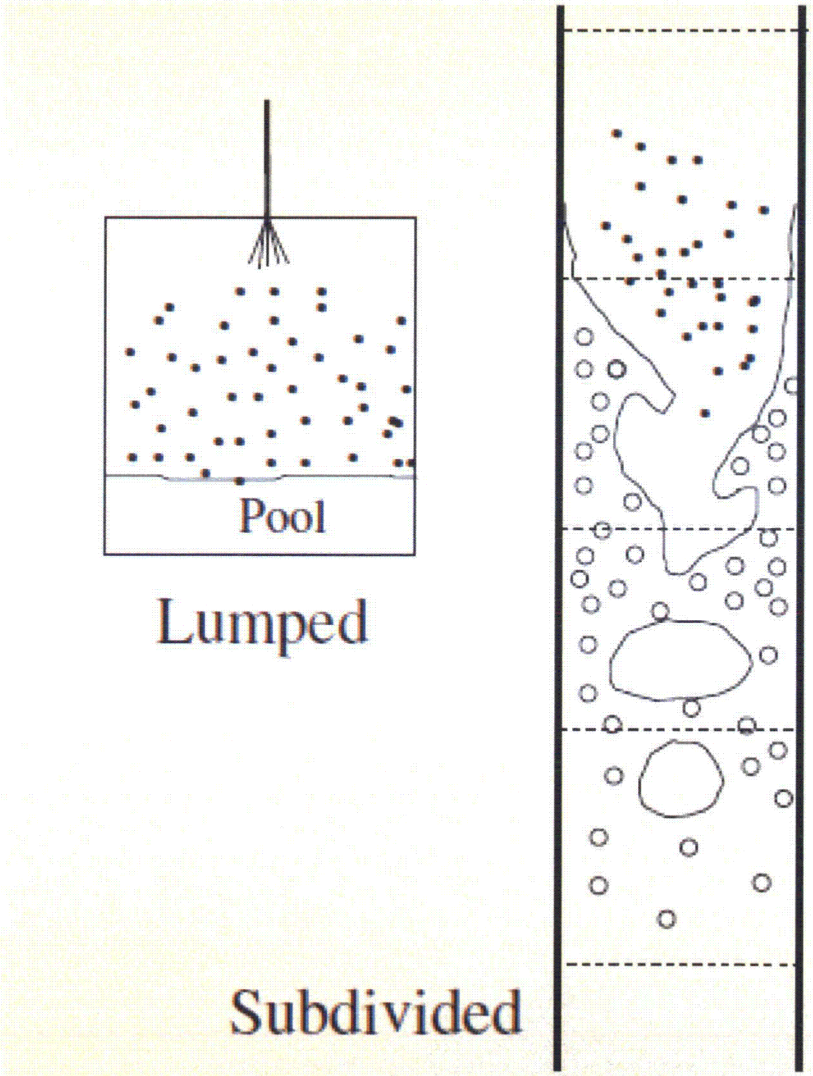
- Combine lumped and subdivided volumes
- Superimposed conductors
- Finite volume solution
  - Semi-explicit



GOTHIC

Multiphase/Multicomponent

- Vapor
  - Steam
  - N gas components
- Drops
  - N fields
- Liquid
  - Films
  - Pools
  - Slugs
  - Stratified flow
- Mist
- Ice



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GOTHIC

Development Plan

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GOTHIC

Validation Plan

- Existing test data used in the assessment of GOTHIC for containment analyses
- Existing test data to be used in the assessment of GOTHIC for transient and accident analyses
- Planned NuScale integral effects test program
- Planned SIET helical coil steam generator test program

GOTHIC

Code Interfaces

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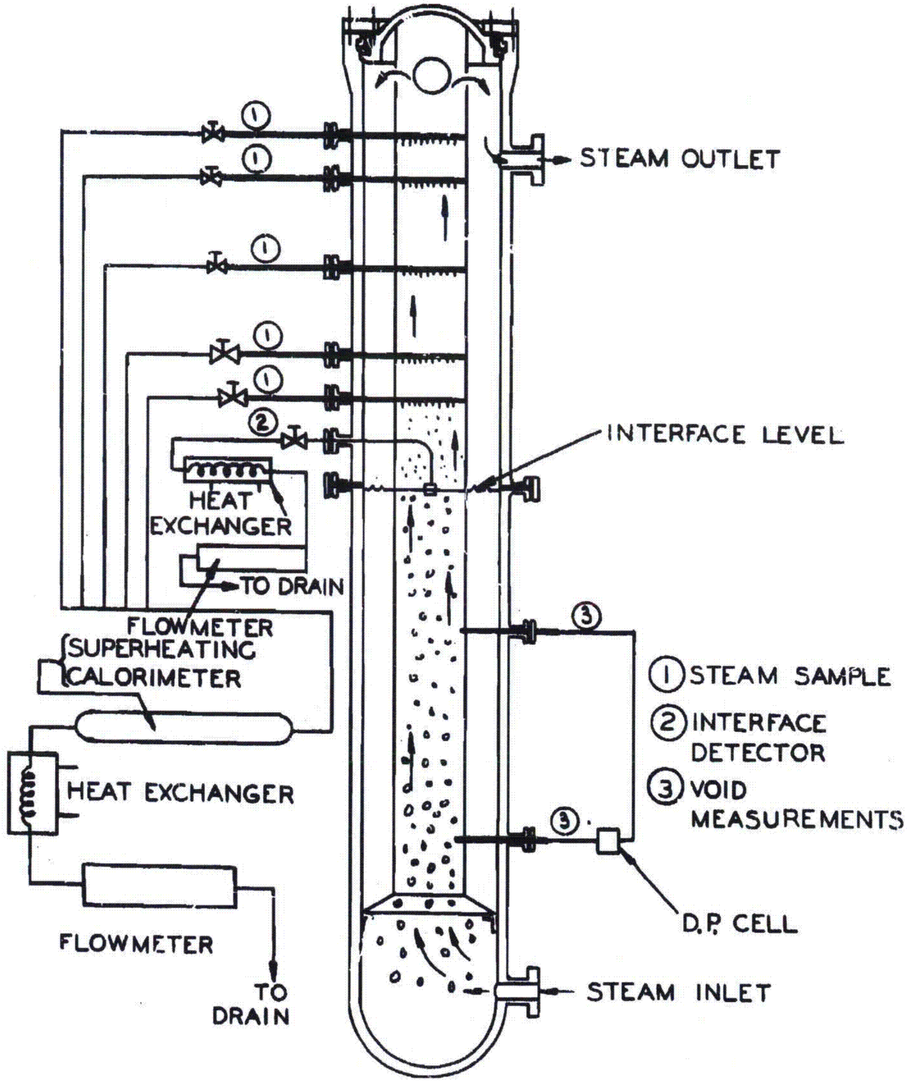
Example Assessment

Void Fraction in Large Pipes

- Large diameter, quiescent voiding may become important during SBLOCA if primary natural circulation is broken.
- Most steam/water experimental data are at much smaller diameters than the NuScale riser.
- Search for large diameter data yielded an interesting experiment

Void Fraction Experimental Facility

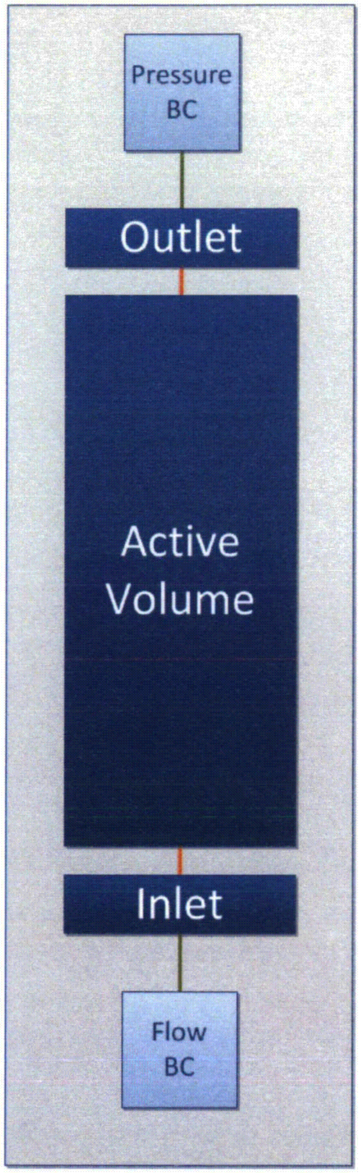
- 36” steady state void fraction data* is available, which will be part of our code assessment
  - variety of pressures
  - variety of steam mass flow rates
  - void fractions measured based upon dPs at three radial locations (measured radial profile)



*ACNP-63035, *Joint US/Euratom R&D Program: AT(11-1)-1186, Quarterly Progress Report*, October-December 1963.

GOTHIC Model

- Match experimental flow areas and approximate L/D measurement location
- Modeling options are set to defaults – no tuning
- Results shown are time and axially averaged



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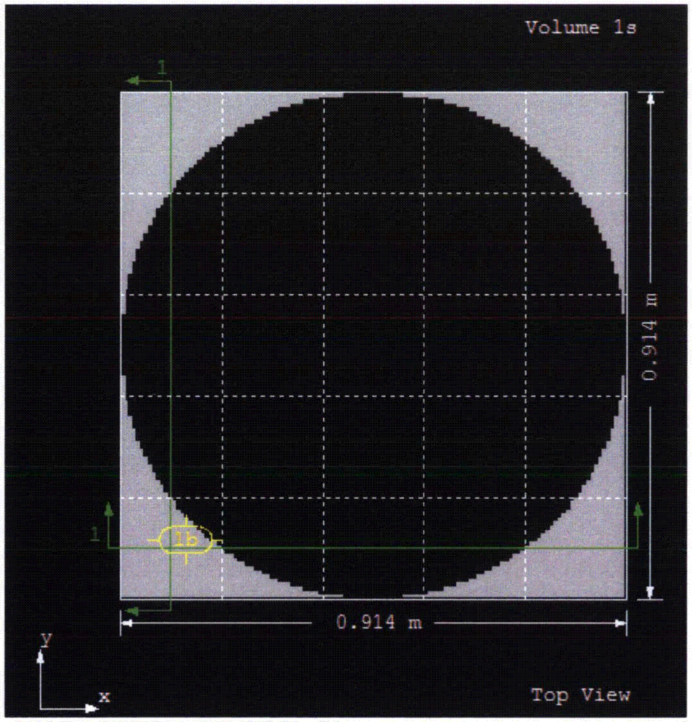
GOTHIC Model

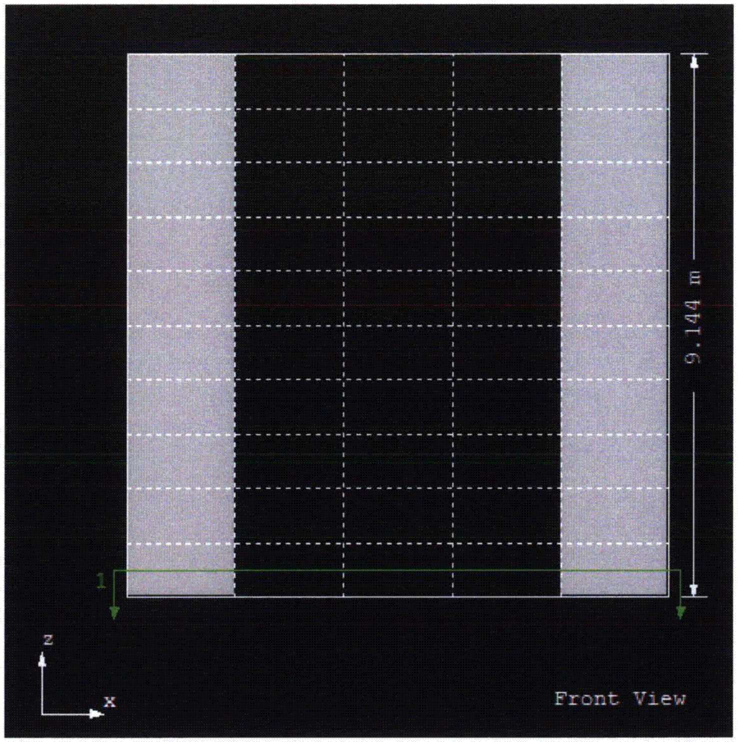
Top View:

- 36” round tube
- 5x5 Cartesian grid



Side View:

- 10 L/D tube
- 10 axial levels – 4 to 6 averaged



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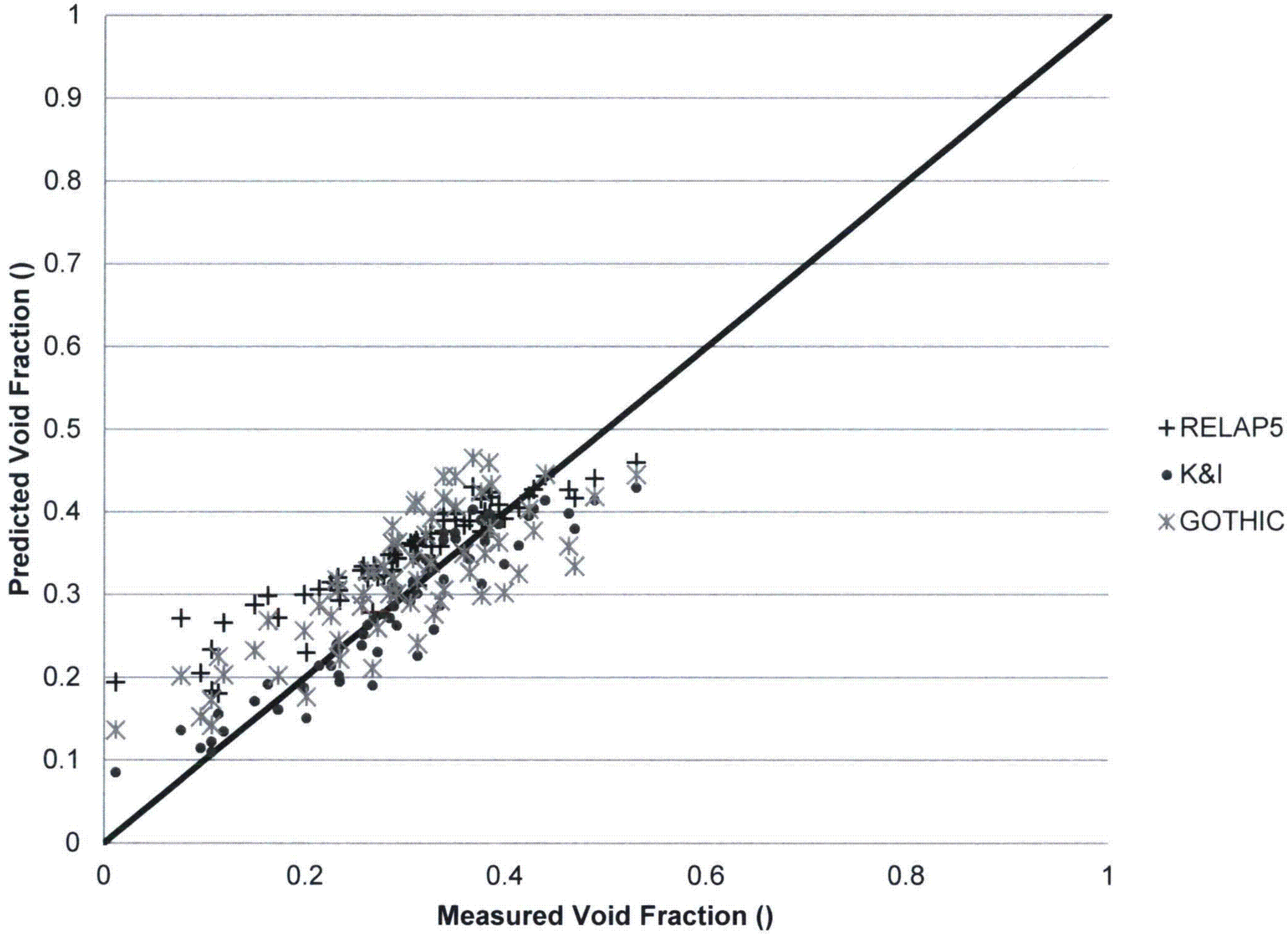
Predicted versus Measured

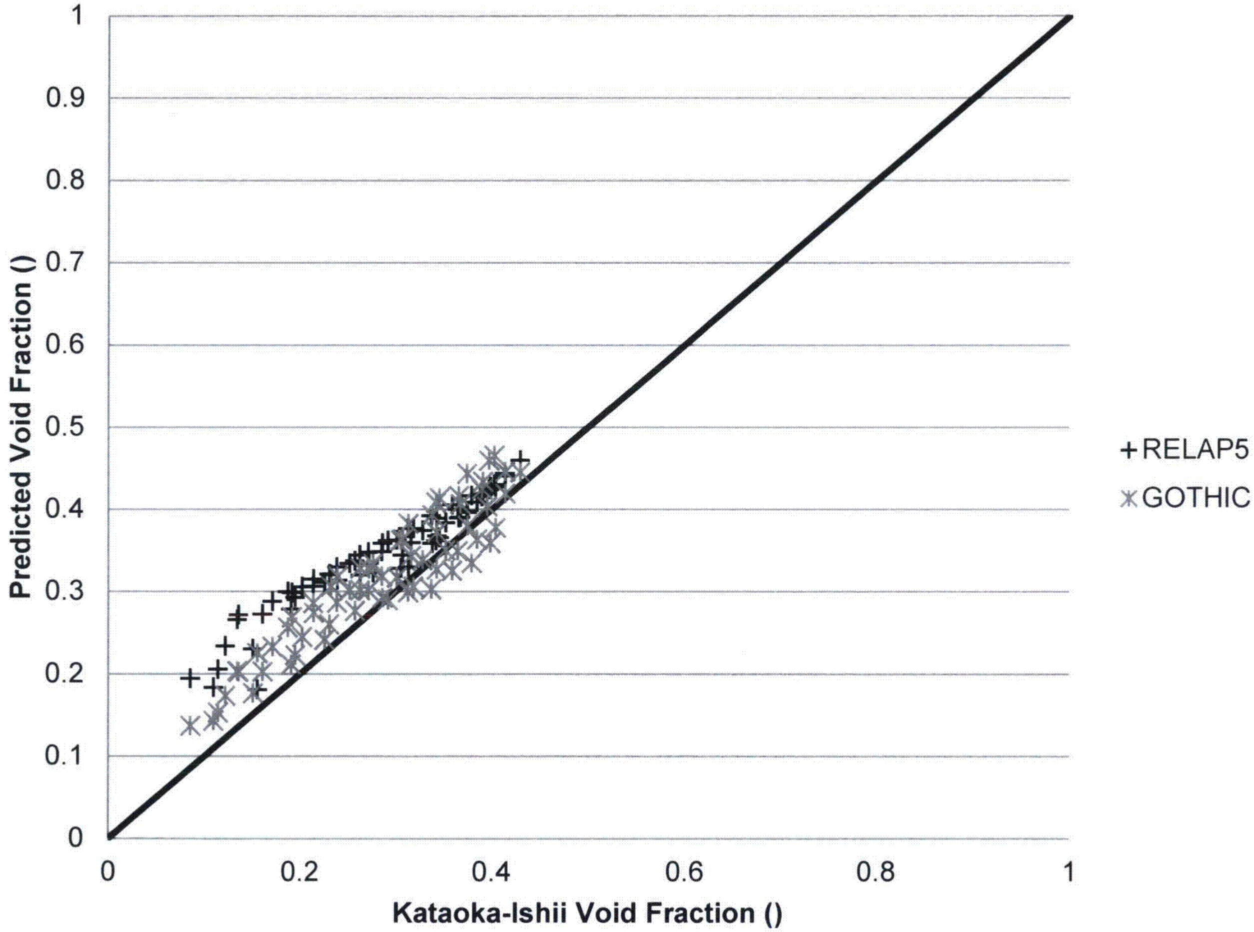
- Kataoka-Ishii* correlation similar to code results
- Similar trend for all three predictions
- Data is not qualified yet



*I. Kataoka, M. Ishii, "Drift Flux Model for Large Diameter Pipe and New Correlation for Pool Void Fraction," *Int. J. Heat Mass Transfer*, Vol. 30, 1987, No 9, pp. 1927-1939.

Predicted versus Kataoka-Ishii Model

- Good agreement for RELAP5 and especially GOTHIC
- Compared to raw data:
  - Less scatter
  - Better trending



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Event Selection and Categorization

Event Selection Process

- Failure modes and effects analysis of plant systems, structures, and components
- Comprehensive list of initiating events
- Categorization of events by frequency of occurrence
  - Anticipated operational occurrences (AOOs)
  - Postulated accidents (PAs)

Event Selection Process

Categorization of Events by Type

SRP CH 15

1. Increase in heat removal by the secondary system
2. Decrease in heat removal by the secondary system
3. Decrease in reactor coolant system (RCS) flow rate
4. Reactivity and power distribution anomalies
5. Increase in reactor coolant inventory
6. Decrease in reactor coolant inventory
7. Radioactive release from a subsystem or component

NuScale DCD CH 15

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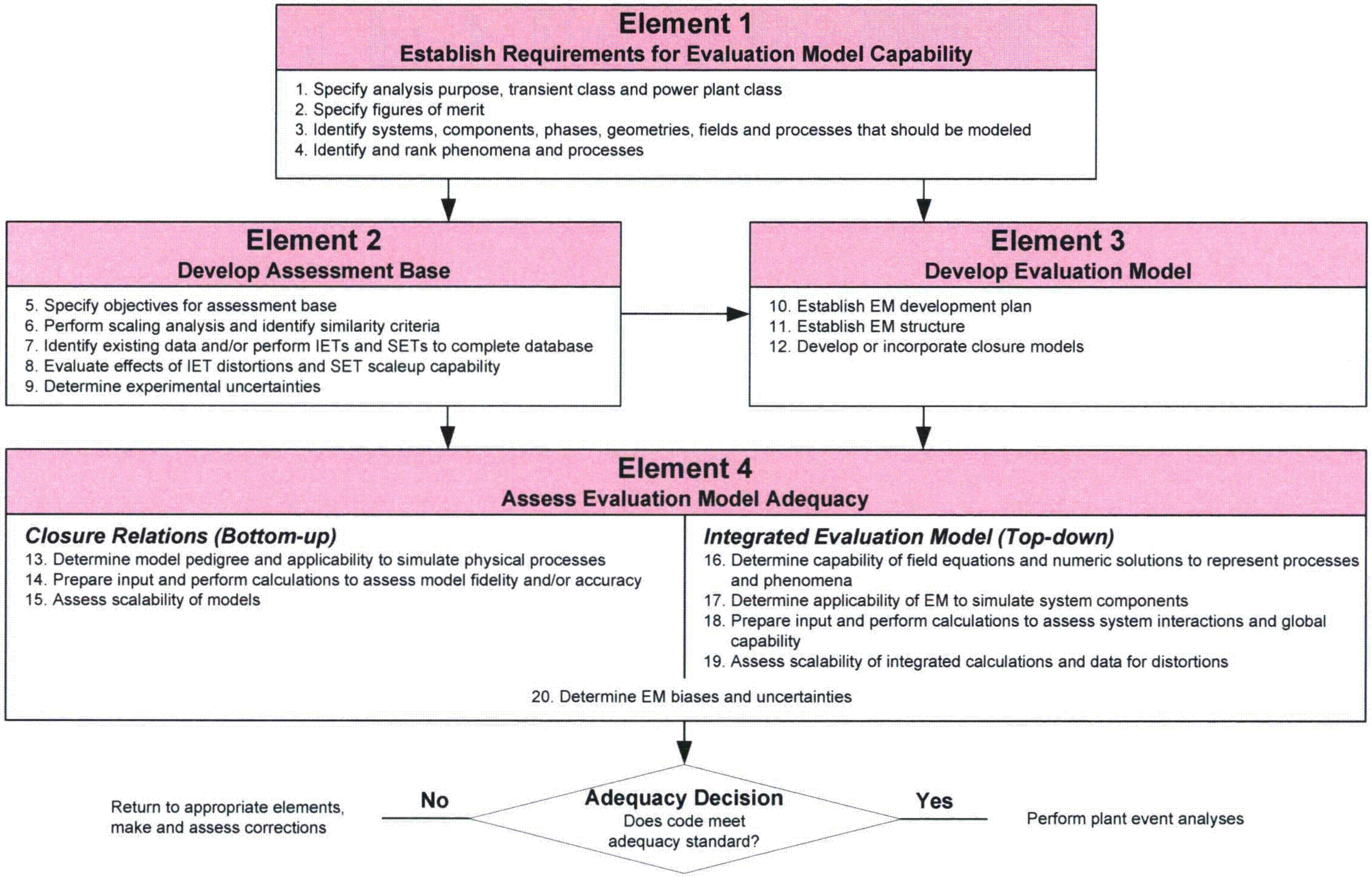
Methods Development

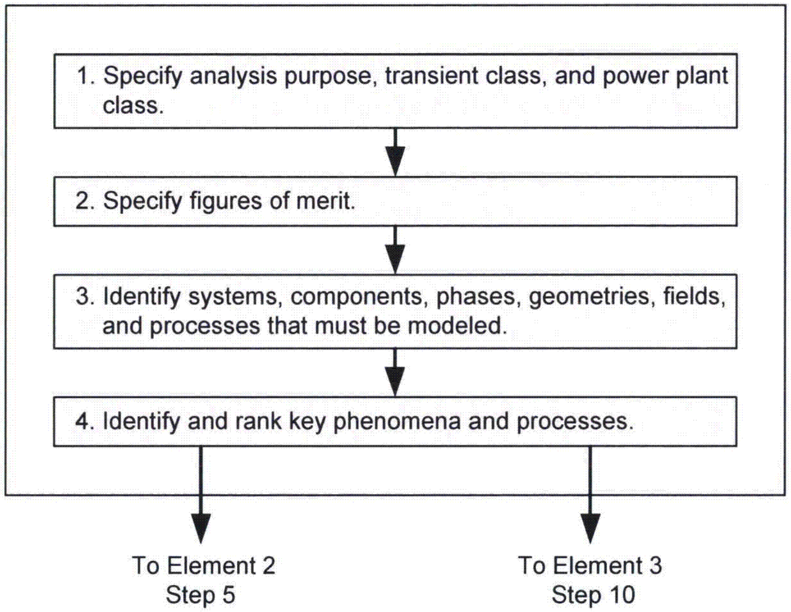
EMDAP for NuScale LOCA Analysis



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

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

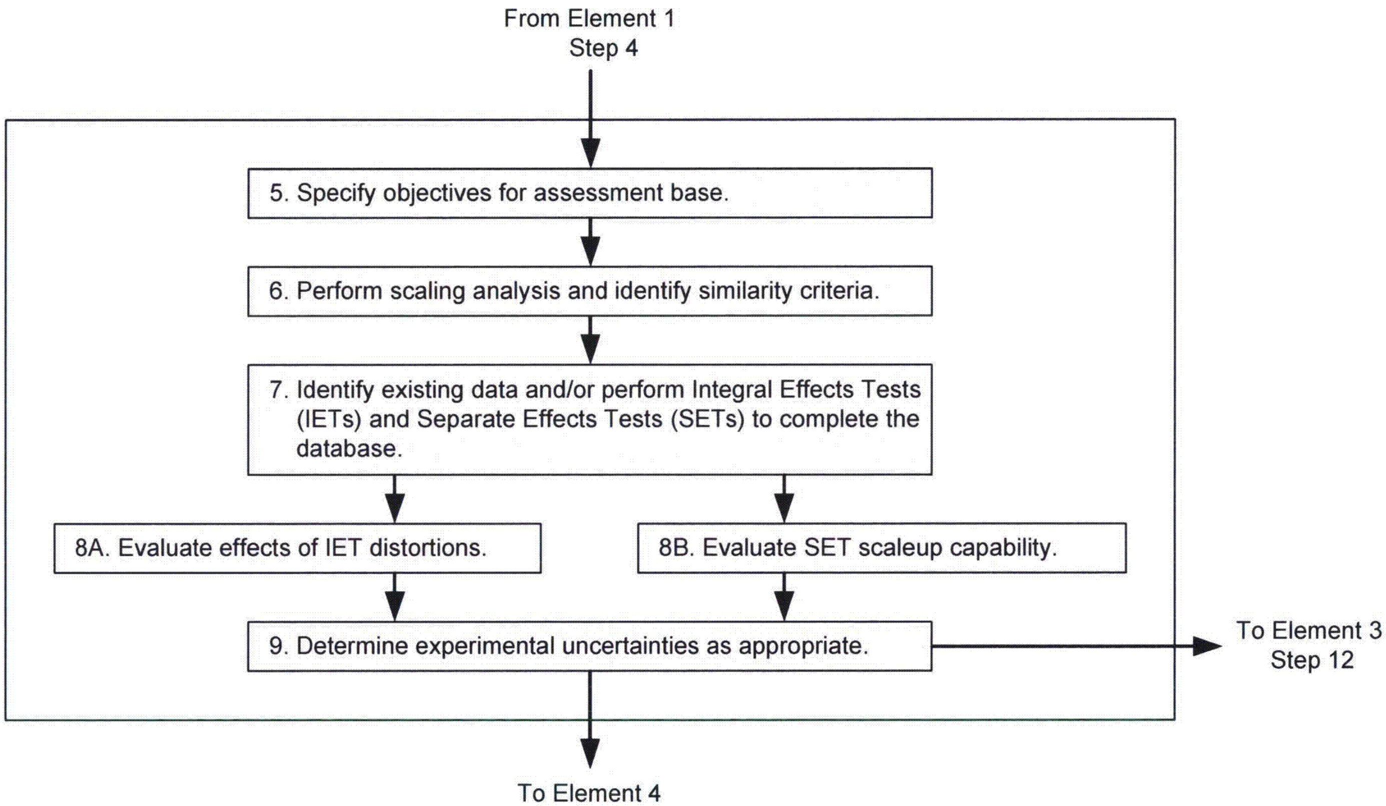
EMDAP – Element 1



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

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

EMDAP – Element 2



EMDAP - Element 2: Develop Assessment Base[1]

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

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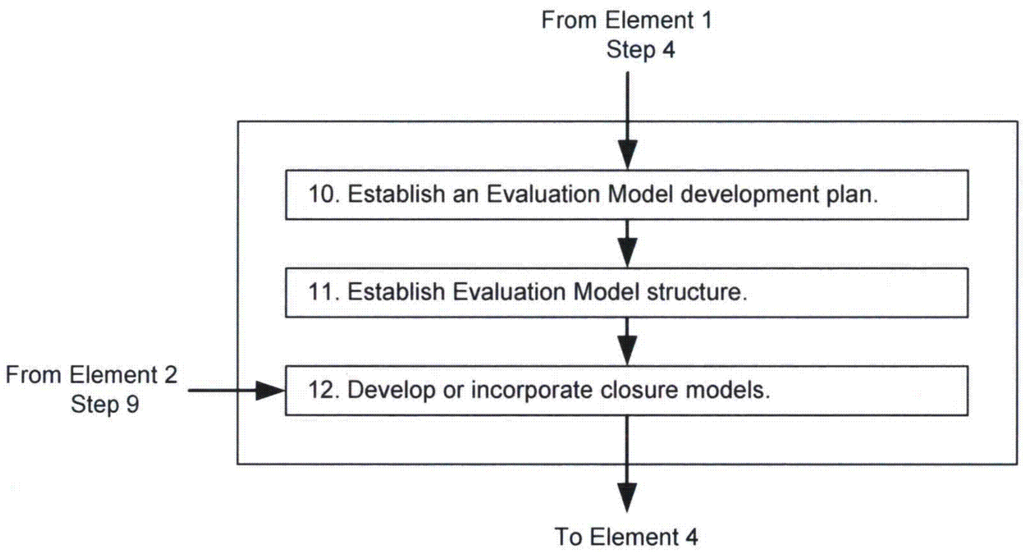
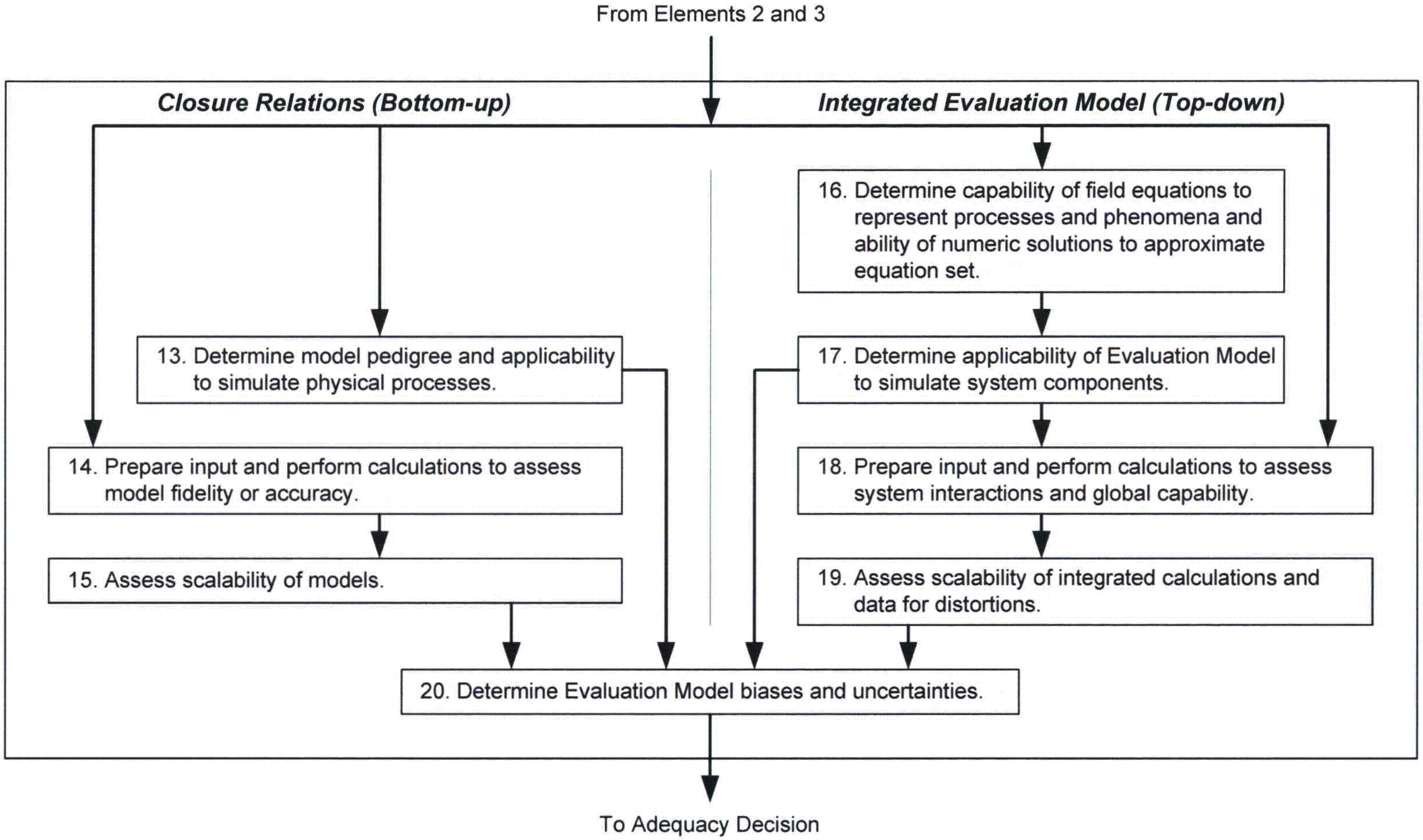
EMDAP – Element 3



EMDAP - Element 3: Develop Evaluation Model[1]

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

EMDAP – Element 4



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

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

Methods Development Summary

Method Class	Use/Application
Core neutronics methods	• Core power distributions • Nuclear uncertainties • Safety analysis physics parameters • Safety limits • Operating limits
Core TH methods	• CHF correlation development • DNBR correlation safety and design limits • DNBR operating limits
SBLOCA methods	• Decrease in RCS inventory/SBLOCA
Non-LOCA methods	• Increase in heat removal by the secondary system • Decrease in heat removal by the secondary system – Decrease in RCS flow rate • Reactivity and power distribution anomalies • Increase in reactor coolant inventory

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Core Neutronics Methods

- Benchmarking of CASMO/SIMULATE to MCNP
- Nuclear uncertainties
- Core power distributions
- Safety analysis physics parameters
- Safety limits
- Operating limits
- S3K RIA applications

Core Thermal Hydraulic Methods

- Benchmarking to STERN CHF and mixing tests
- CHF correlation development
- DNBR correlation safety and design limits
- DNBR operating limits

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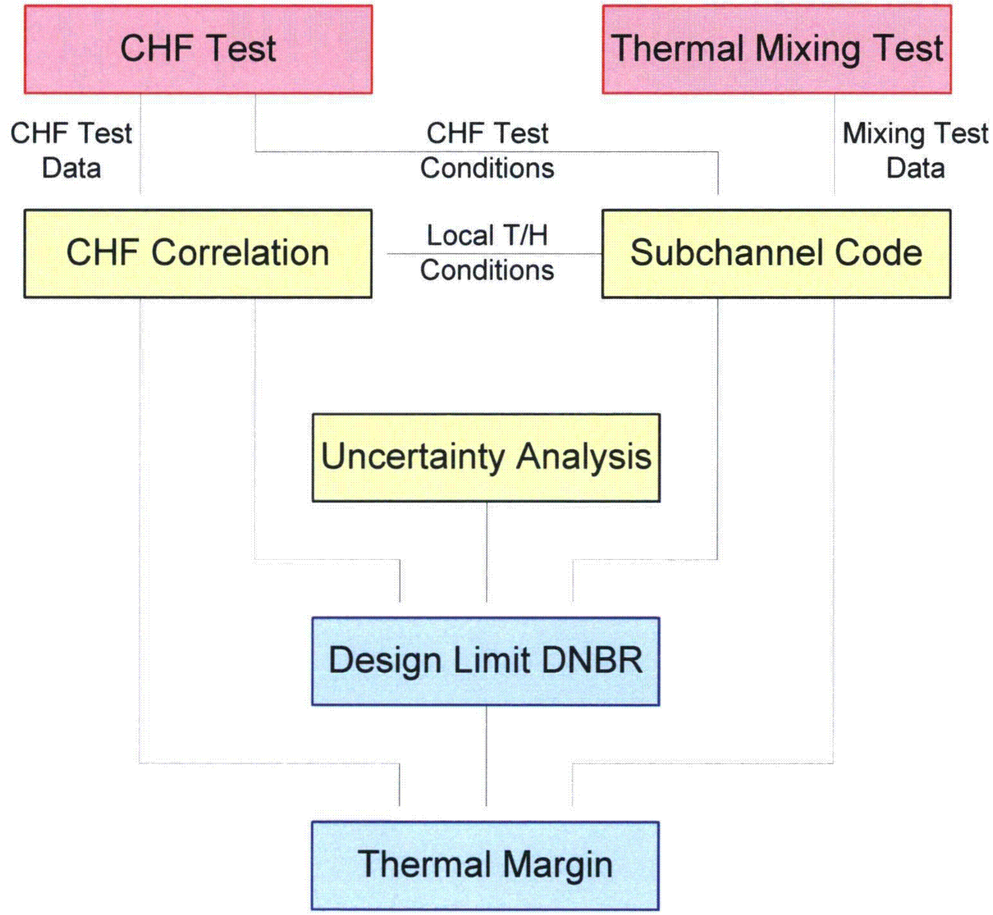
Core Thermal Design



Core Thermal Design Process

SBLOCA Methods

- No core uncovery/less complex EM sufficient
- R. G. 1.203 process
  - PIRT
  - Assessment base includes NuScale integral data and large-scale helical coil steam generator data
  - GOTHIC validation
- Appendix K approach
  - Identification of elements that are not applicable
  - GOTHIC EM
  - Break spectrum
  - Compliance with 50.46(c)

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SBLOCA PIRT Results

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SBLOCA PIRT Results

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PIRT Conclusions

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Non-LOCA Methods

- PIRT
- Benchmarking to integral test data
- GOTHIC reactor coolant system and containment models
- DCD Chapter 15 AOO and accident identification

Control Rod Ejection Methodology

Regulatory Acceptance Criteria

- R.G. 1.183 radiological limits
  - Gap release from cladding failures (PCMI or DNBR)
  - Source term transport
- Cladding failure due to temperature (cal/gm), PCMI (Δcal/gm), or DNBR
- Core coolability limits on cal/gm, incipient melting, mechanical energy, and loss of geometry (fuel rod ballooning post-DNB)
- RCS peak pressure < ASME Service Limit C

Control Rod Ejection Methodology

Methodology for Fuel-Related Acceptance Criteria

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Control Rod Ejection Methodology

SCANR Model

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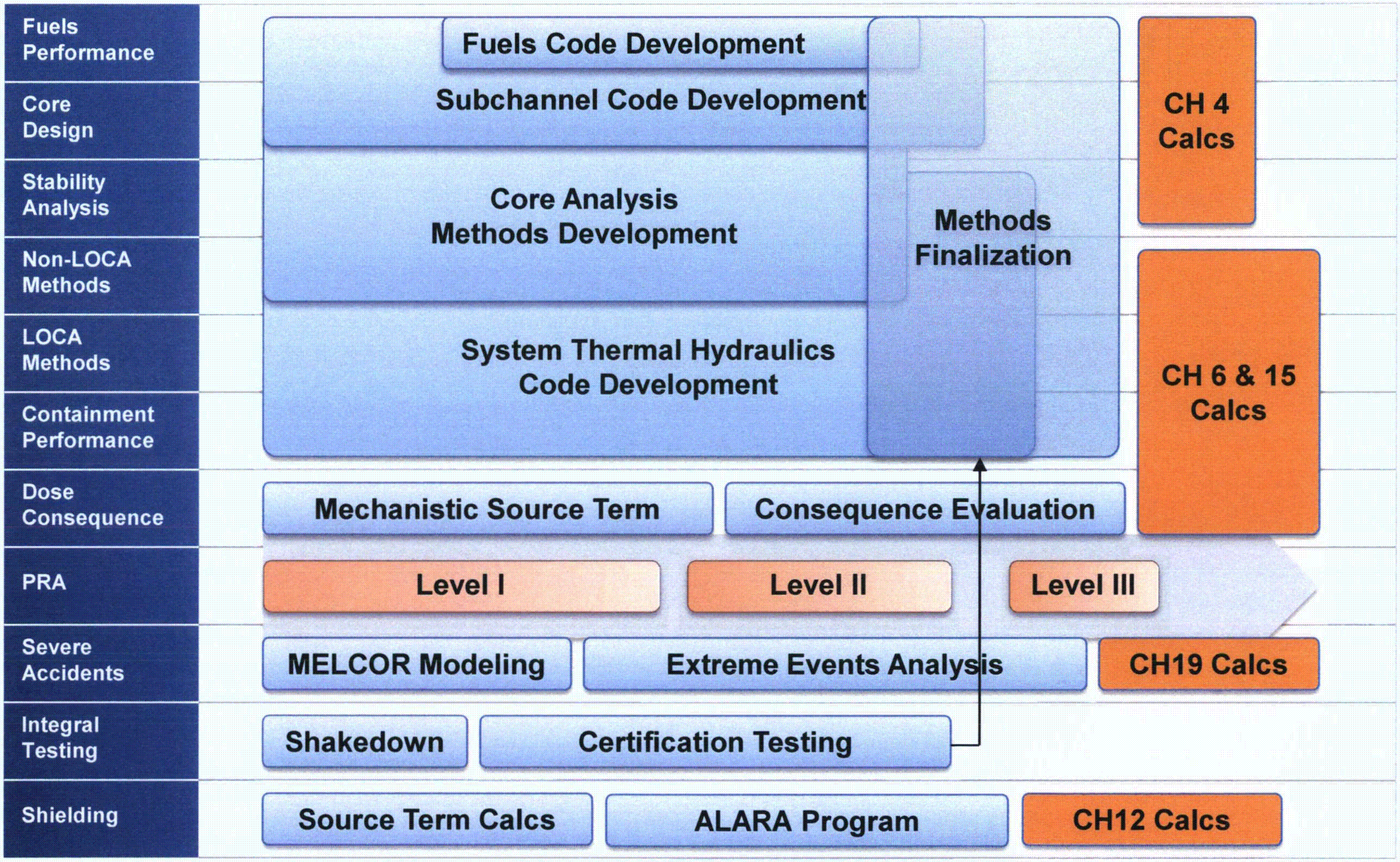
Control Rod Ejection Methodology

NuScale versus Large PWR Applications

- S3K has been reviewed and approved by NRC for PWR rod ejection incident (REA).
- SCANR will be benchmarked to VIPRE-01.
- Full core S3K and SCANR models
- Only one symmetric ejected rod core location
- Low initial fuel power density compared to PWRs
- Low initial mass flux compared to PWRs
- Low fuel exposure compared to PWRs

Schedule and Work Plan

Safety Analysis and Nuclear Methods Work Plan



DBA: Design Basis Accident
BDBA: Beyond Design Basis Accident

Pre-Application Schedule

NuScale Power Responses to NRC's RIS 2011-02 Rev. 1

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SA Codes and Methods Report Outline

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Conclusions

- Relying on well-established industry codes and data for safety analysis
- A large majority of models, correlations, and data are directly applicable
- Some additional data and model/code development are needed to address unique design characteristics (e.g., helical coil steam generators)
- Passive and inherent safety features provide significant margins to safety in a number of key areas (e.g., no core uncovery during LOCAs).
- GOTHIC used in a different application envelope than the NRC uses
- Other code packages such as SIMULATE or SCANR may require additional attention during pre-application.
- Preparation, review, and approval of safety analysis codes and methods are resource intensive for both the applicant and the NRC.
- An agreed upon pre-application engagement activities and schedule early is important for success.