Enclosure:

"ACRS Full Committee Presentation: NuScale Topical Report – Rod Ejection Accident Methodology," PM-0320-69146, Revision 0
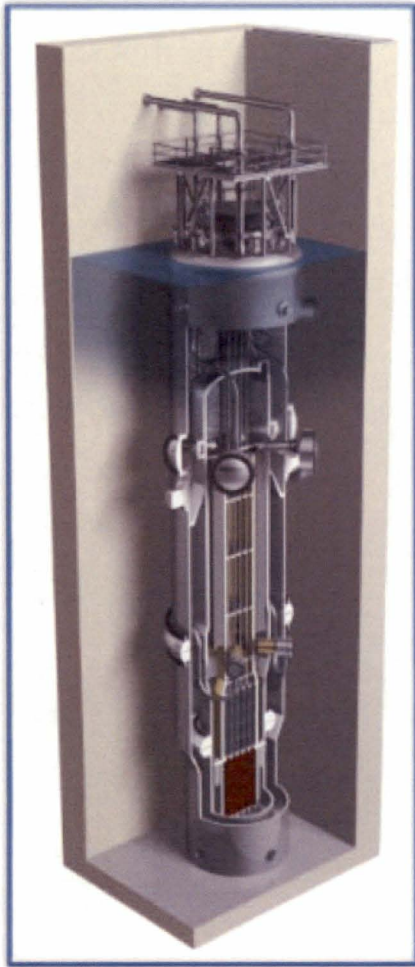
ACRS Full Committee Presentation

NuScale Topical Report

Rod Ejection Accident Methodology

March 5, 2020

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Template #: 0000-21727-F01 R5

Presenters

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Opening Remarks – NuScale T/H Methods

System T/H Analysis Basis

- NRELAP5 code developed from RELAP5-3D
  - Modified to address NuScale-specific phenomena/systems

- LOCA Evaluation Model (EM) developed following RG 1.203 EMDAP
  - LOCA EM extended to derive EMs for other events as shown in this figure.
  - LOCA EM assessment basis leveraged for non-LOCA.

- Additional supporting EMs include
  - Nuclear Analysis Codes – TR-0716-50350-P-A
  - Critical Heat Flux – TR-0116-21012-P-A
  - Subchannel Analysis – TR-0915-17564-P-A

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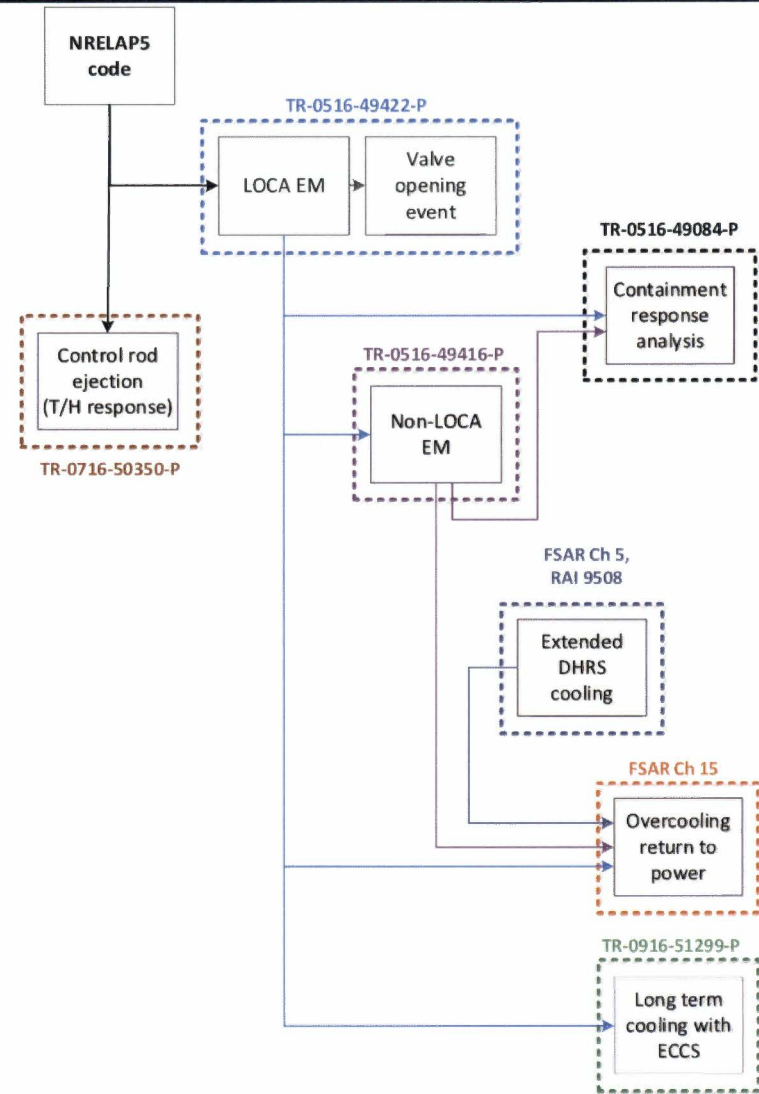
Agenda

- Event Overview

- Acceptance Criteria

- PCMI Criteria – DG-1327

- Method Flowchart

- Steady State Initialization

- Event Evaluations

- Summary

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Overview

- NuScale seeks approval of methodology for modeling rod ejection accident (REA) events

- Bounding reactivity initiated accident (RIA) from General Design Criteria (GDC) 28

- REA is unique in comparison to other Ch. 15 events

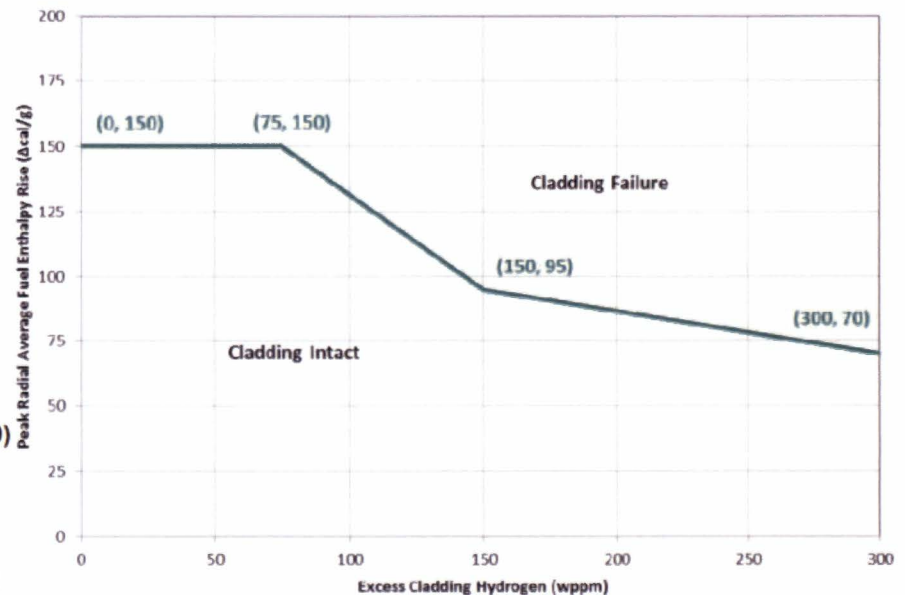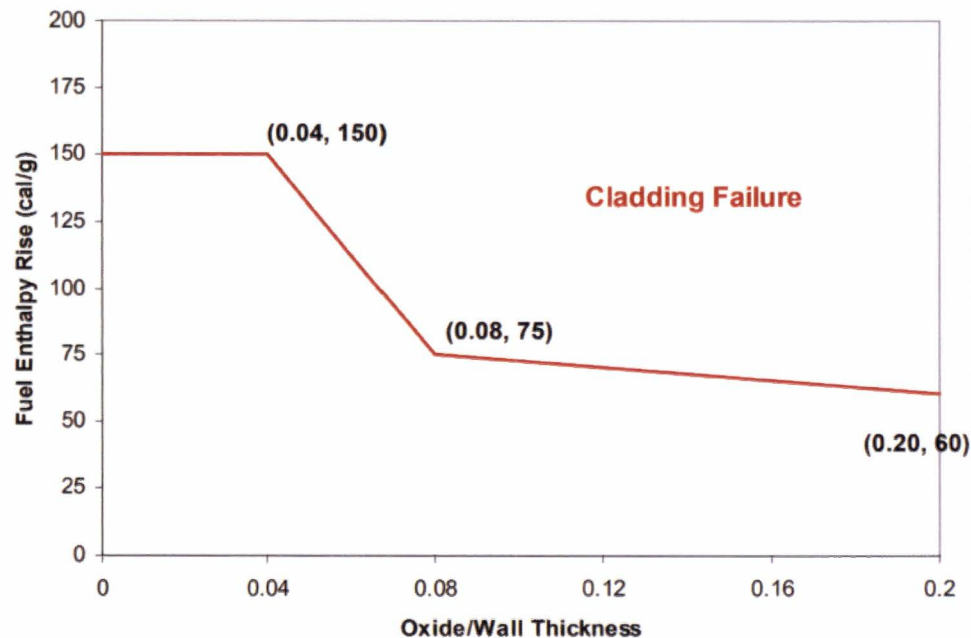| Description	Rod Ejection	Other Events
Dominant Physics	Nuclear	Thermal-Hydraulics
Timing	milli-sec	sec to hr
Spatially	Local	Global
Peak power	~5x Full Power	~1.2x Full Power
Integrated Energy	Low	Low to High
Postulated Cause	Failure of ASME Class 1 Pressure Boundary	Single Equipment Failure
Acceptance Criteria	Specialized	Generic

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Unique Event Acceptance Criteria

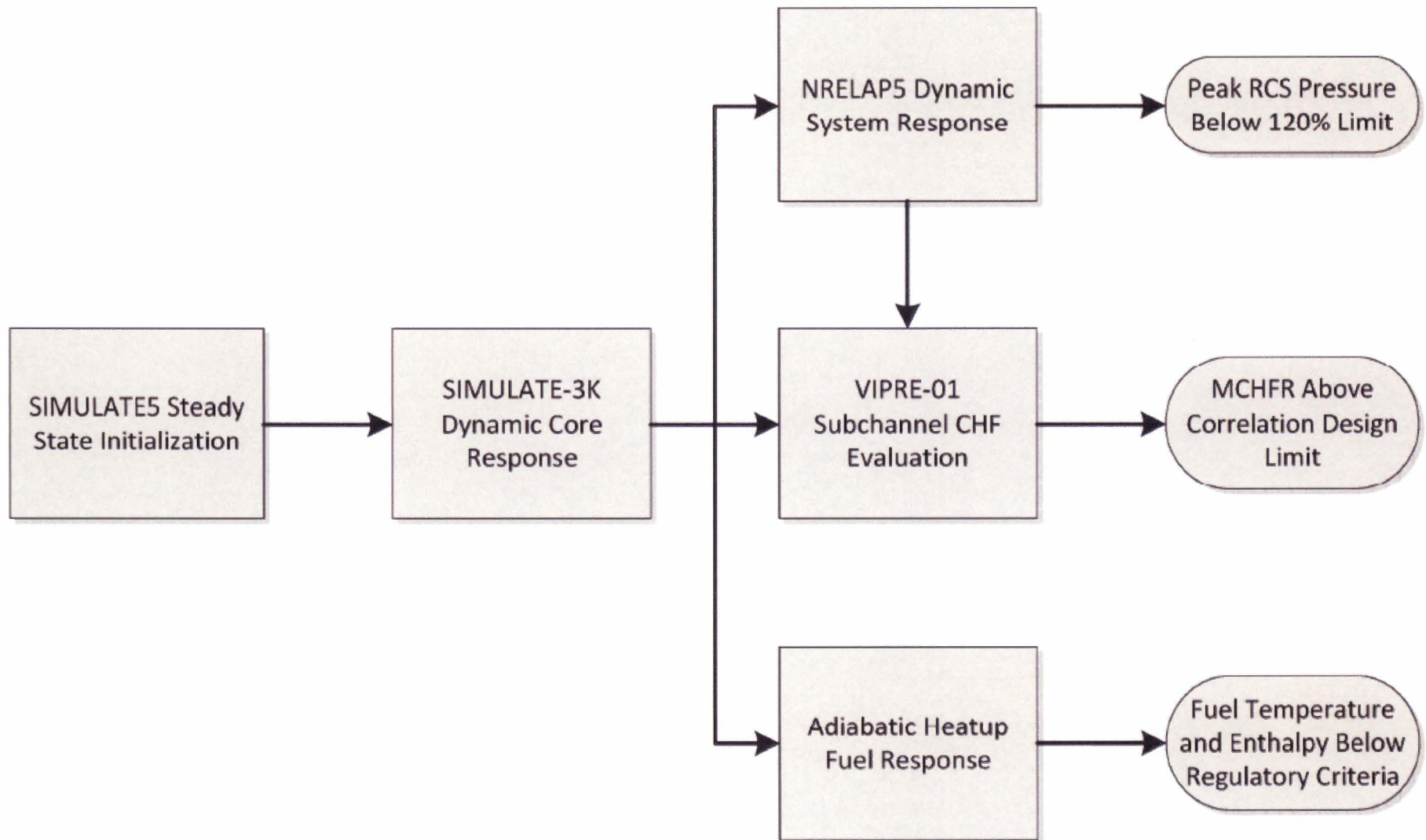| Criteria Description	Topical Section	Unique?
Maximum reactor coolant system pressure	5.3	No
Hot zero power (HZP) fuel cladding failure	5.5.2	Yes
FGR effect on cladding differential pressure	N/A	Yes
Critical heat flux (CHF) fuel cladding failure	5.4.1	No
Cladding oxidation-based PCMI failure	5.5.3	Yes
Cladding excess hydrogen-based PCMI failure	N/A	Yes
Incipient fuel melting cladding failure	5.5.1	No
Peak radial average fuel enthalpy for core cooling	5.5.2	Yes
Fuel melting for core cooling	5.5.1	No
Fission product inventory (failed fuel census)	5.6	Yes

- Submitted NuScale design and method inherently precludes fuel failure, thus no accident radiological consequences are evaluated.

- PCMI: Pellet-Clad Mechanical Interaction

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Revised PCMI Criteria

- In general, the NuScale REA methodology has adopted the limiting criteria of the 'Clifford Letter' (ML14188C423), now included in draft guide DG-1327 (ML16124A200). In spirit, NuScale is prepared for this regulatory change:
  - Closed session presents example results, showing large margins for enthalpy rise
  - A technical 'formality' inhibits complete adoption at this time. NuScale does not currently have a validated cladding H_2 model to convert local exposure to excess cladding hydrogen
  - Oxidation criteria from NUREG-0800 Section 4.2, Appendix B (ML07074000) is used
  - To simplify method, no exposure is credited (Limit: 75 Δcal/gm)
  - NuScale M5 cladding less susceptible than other zirc alloy-type clad used in the industry

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Unique Event Method (Flowchart)

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Steady-State Initialization

- SIMULATE5: Setup the core response analysis

- Code shown to be appropriate in TR-0616-48793-A (Nuclear Analysis Codes and Methods Qualification)

- Determination of the worst rod stuck out (WRSO)
  - Assumption bounds potential for ejected assembly to damage adjacent control rod assembly
  - Due to rapid nature of the event, location does not significantly affect the results in NuScale application

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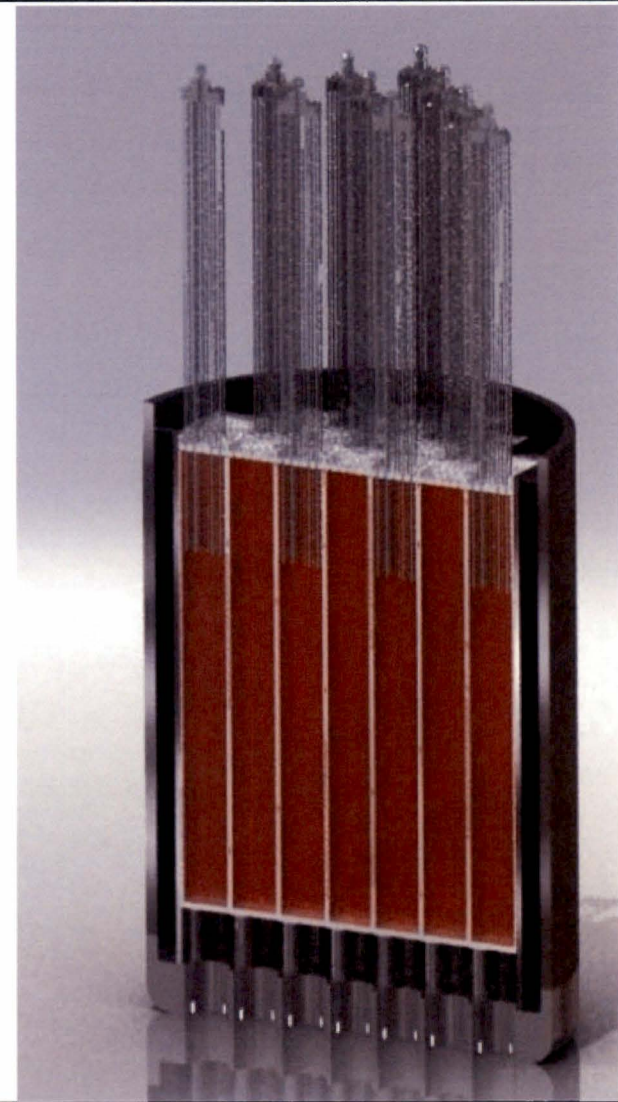
Dynamic Core Response

- SIMULATE-3K: Model transient core response

- Benchmarked to SPERT-III experiment and NEACRP computational benchmark

  - Benchmarks demonstrate the combined transient neutronic, thermal-hydraulic, and fuel pin modeling capabilities

  - SIMULATE-3K results generally in excellent agreement with the results from the two benchmark problems

- Uncertainties applied for each simulation:

  - Delayed Neutron Fraction

  - Ejected Rod Worth

  - Doppler Temperature Coefficient

  - Moderator Temperature Coefficient

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CHF Evaluation

- VIPRE-01: Model detailed thermal-hydraulics

- Evaluate critical heat flux (CHF) acceptance c

- Code shown to be appropriate in TR-0915-17
  Analysis Methodology)

- Unique event differences in method:

  – Smaller axial nodalization (smaller time steps)

  – Radial power distribution (case-specific)

  – Axial power distribution (peak assembly)

  – Convergence parameters

- Additional parametric sensitivity cases pe
  application to holistically justify difference

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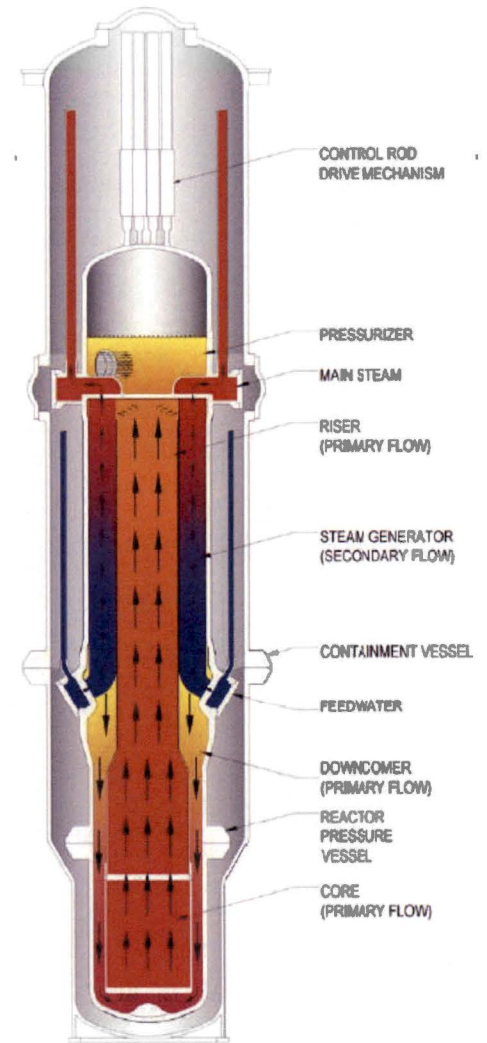
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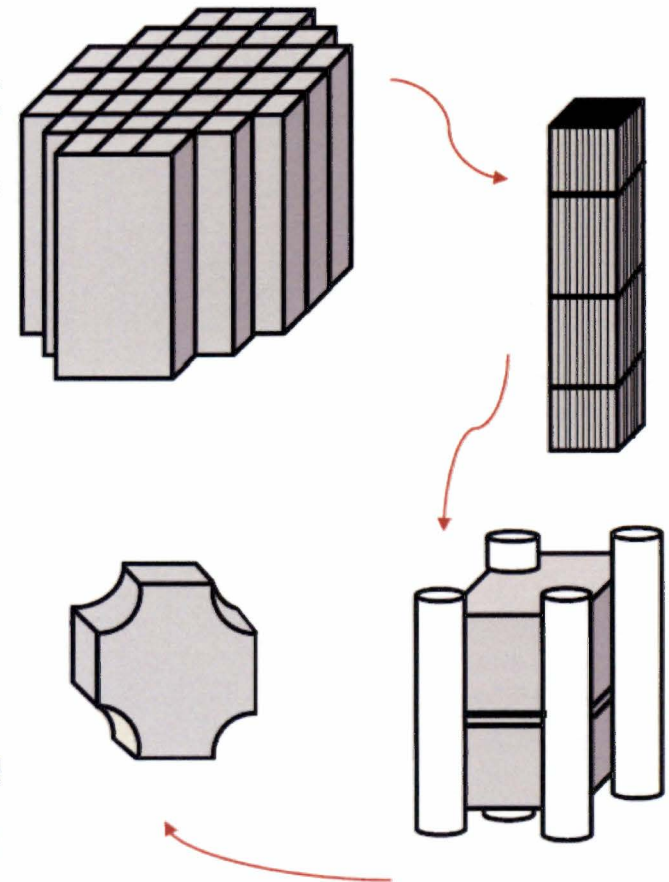
Adiabatic Fuel Heatup

- Hand-Calculation: Model fuel response

- Total energy (from SIMULATE-3K) during the transient is integrated

- Conservative as no energy is allowed to leave the fuel rod

- Energy is then converted into either a temperature or enthalpy increase

- Fuel rod geometry, heat capacity, and power peaking factors taken into account

- Calculated values compared to NRC developed acceptance criteria

  – Example values provided in closed session

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Dynamic System Response I

- NRELAP5: Evaluate system response for input to <u>CHF Evaluation</u>

- Code shown to be appropriate in TR-0516-49416 (Non-LOCA Methodologies)

- Transient power from SIMULATE-3K utilized as input

  – No reactivity calculation performed in NRELAP5

- Provides system thermal-hydraulic conditions to subchannel (CHF) evaluation

  – System flow, pressure, and inlet temperature

  – 'Screens' cases for potential to be limiting

  – Family of limiting cases evaluated with VIPRE-01

CONTROL ROD
DRIVE MECHANISM

PRESSURIZER

MAIN STEAM

RISER
(PRIMARY FLOW)

STEAM GENERATOR
(SECONDARY FLOW)

CONTAINMENT VESSEL

FEEDWATER

DOWNCOMER
(PRIMARY FLOW)

REACTOR
PRESSURE
VESSEL

CORE
(PRIMARY FLOW)

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Dynamic System Response II

- NRELAP5: Evaluate system response for <u>pressurization</u>

- Limiting scenario: Low ejected worth that raises the power quickly to just below both the high power and high power rate trip 'setpoints'

- Point-kinetics model used based on bounding static worth

- Peak system pressure calculated compared to acceptance criteria

- Example results to be presented in closed session

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Summary

- A conservative analysis method for the unique rod ejection accident

- Topical report provides details and justification for:

  – Software tools and acceptance criteria used

  – Applicability of the method and tools

  – Appropriate treatment of uncertainties

- Results from application of the method provide input to FSAR Chapter 15

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Acronyms

- **CHF – Critical Heat Flux**

- **GDC – General Design Criteria**

- **HZP – Hot Zero Power**

- **MCHFR – Minimum Critical Heat Flux Ratio**

- **NEACRP – Nuclear Energy Agency Committee on Reactor Physics**

- **PCMI – Pellet Clad Mechanical Interaction**

- **REA – Rod Ejection Accident**

- **RIA – Reactivity Initiated Accident**

- **WRSO – Worst Rod Stuck Out**

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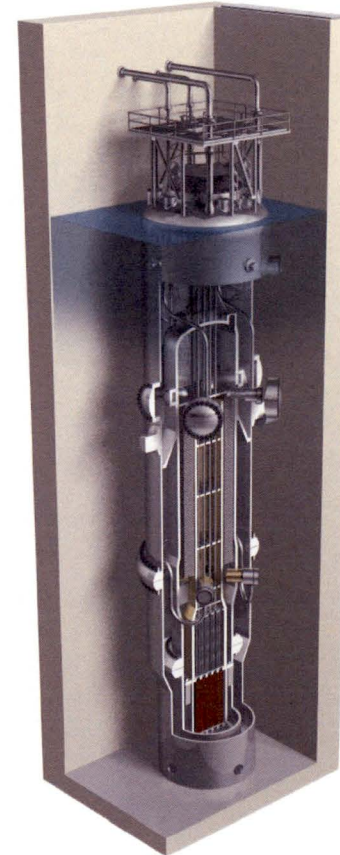Corvallis, OR 97330
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