NUSCALE™
Power for all humankind

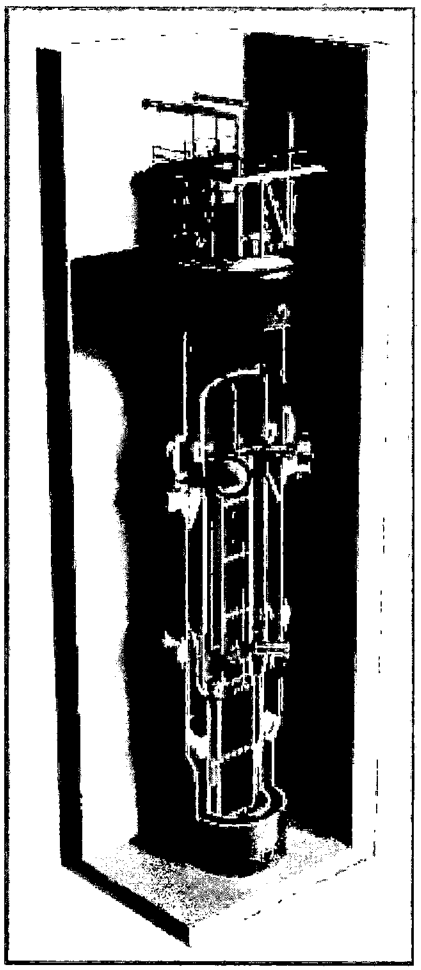
LO-0220-68846

Enclosure:

"ACRS Subcommittee Presentation: NuScale Topical Report - Rod Ejection Accident Methodology," PM-1019-67365, Revision 0

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

ACRS Subcommittee Presentation



NuScale Topical Report

Rod Ejection Accident Methodology

February 19, 2020

PM-1019-67365
Revision: 0

Template #: 0000-21727-F01 R5

Presenters

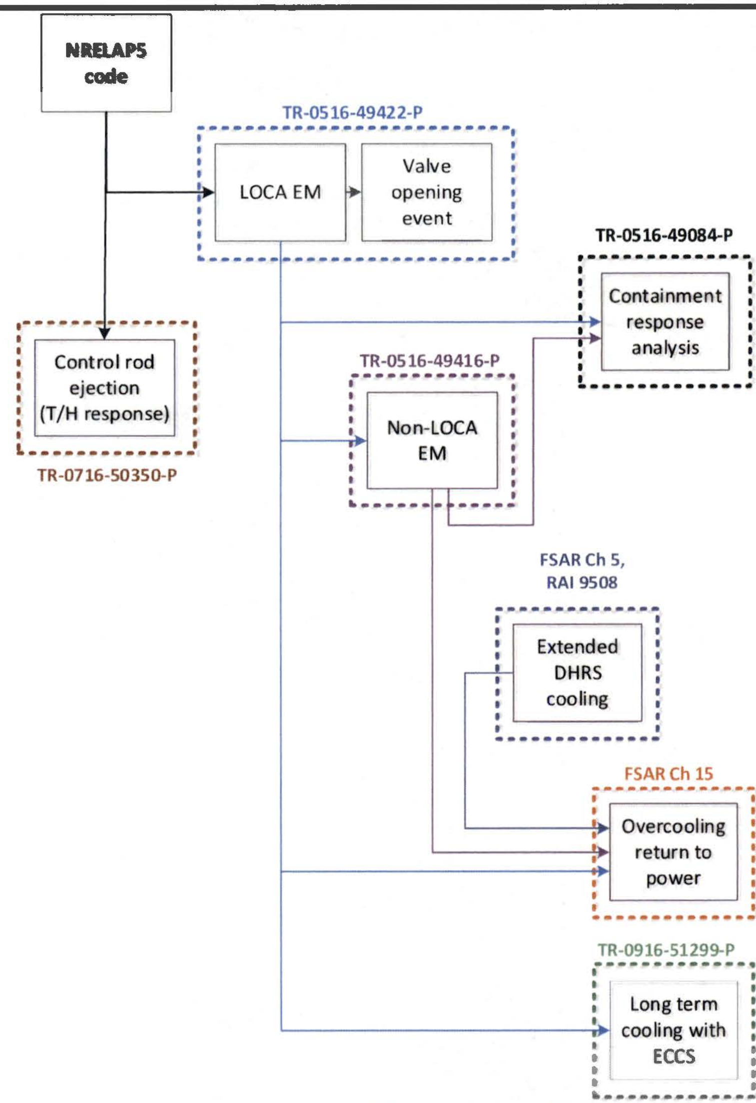
Kenny Anderson
Nuclear Fuels Analyst

Matthew Presson
Licensing Project Manager

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



Agenda

- Event Overview
- Acceptance Criteria
- PCMI Criteria – DG-1327
- Method Flowchart
- Steady State Initialization
- Event Evaluations
- Summary

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

Unique Event Acceptance Criteria

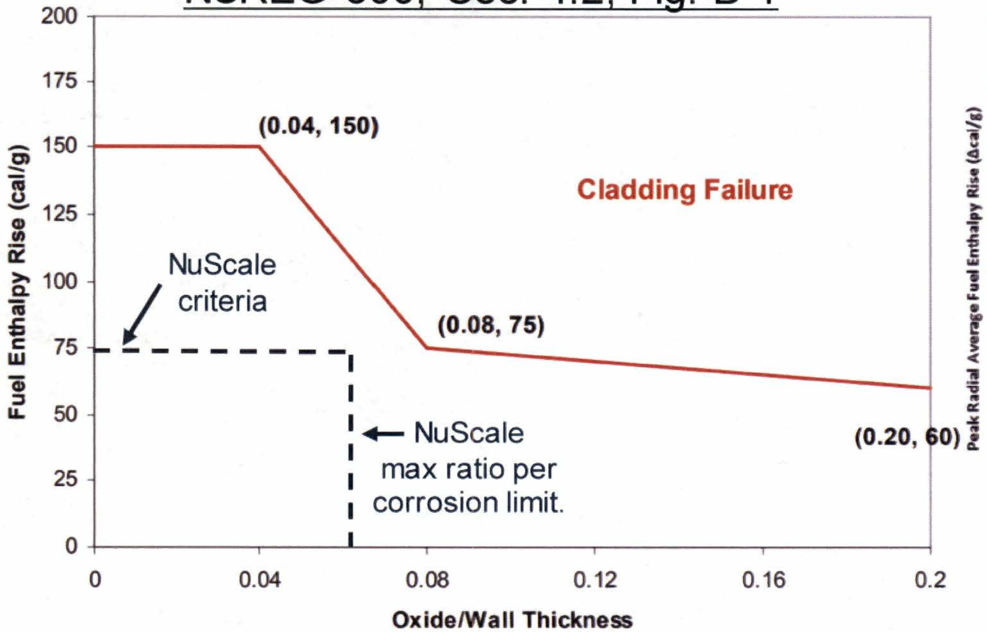
Criteria Description	Topical Section	Unique?
Maximum reactor coolant system pressure	5.3	No
Hot zero power (HZP) fuel cladding failure	5.5.2	Yes
FGR effect on cladding differential pressure	N/A	Yes
Critical heat flux (CHF) fuel cladding failure	5.4.1	No
Cladding oxidation-based PCMI failure	5.5.3	Yes
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

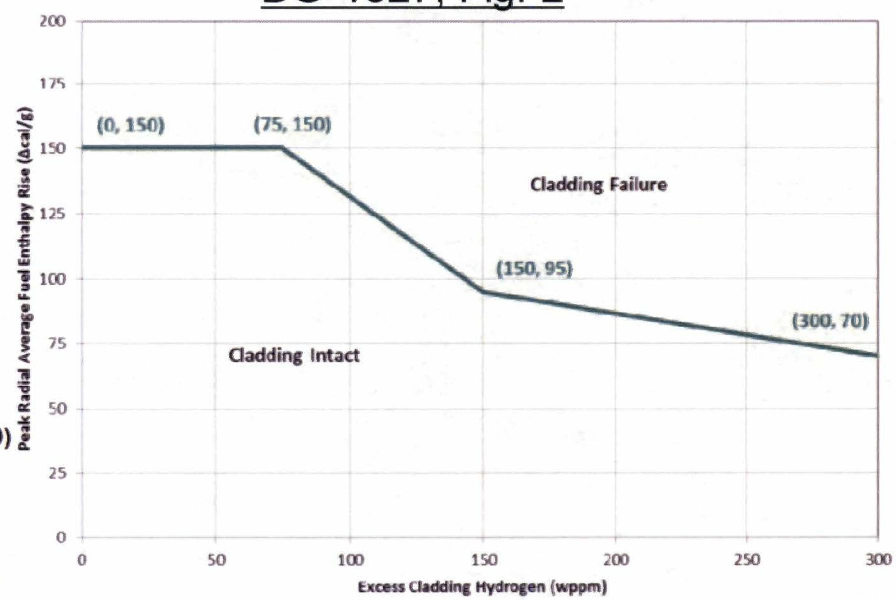
Revised PCMI Criteria

- 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

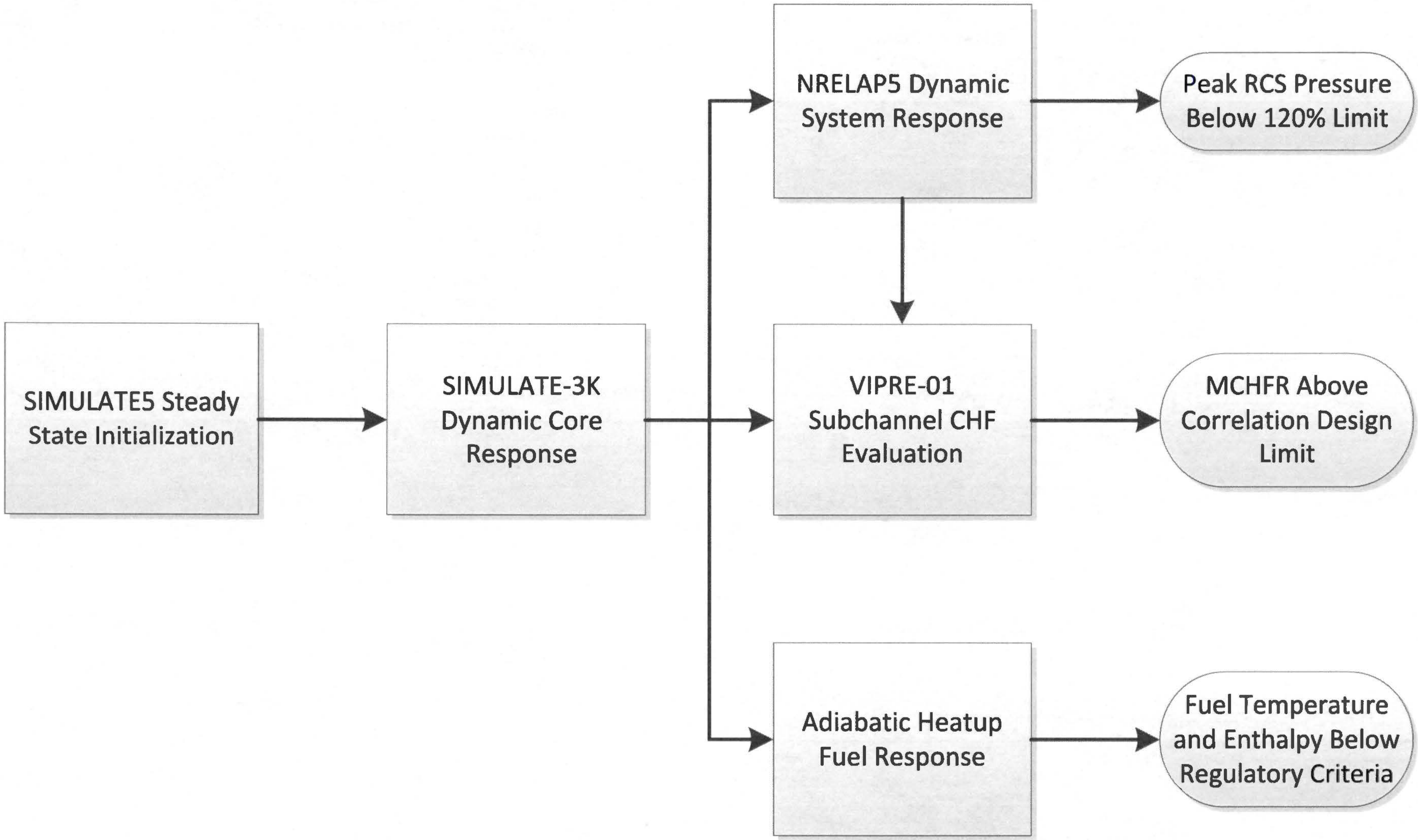
NUREG-800, Sec. 4.2, Fig. B-1



DG-1327, Fig. 2

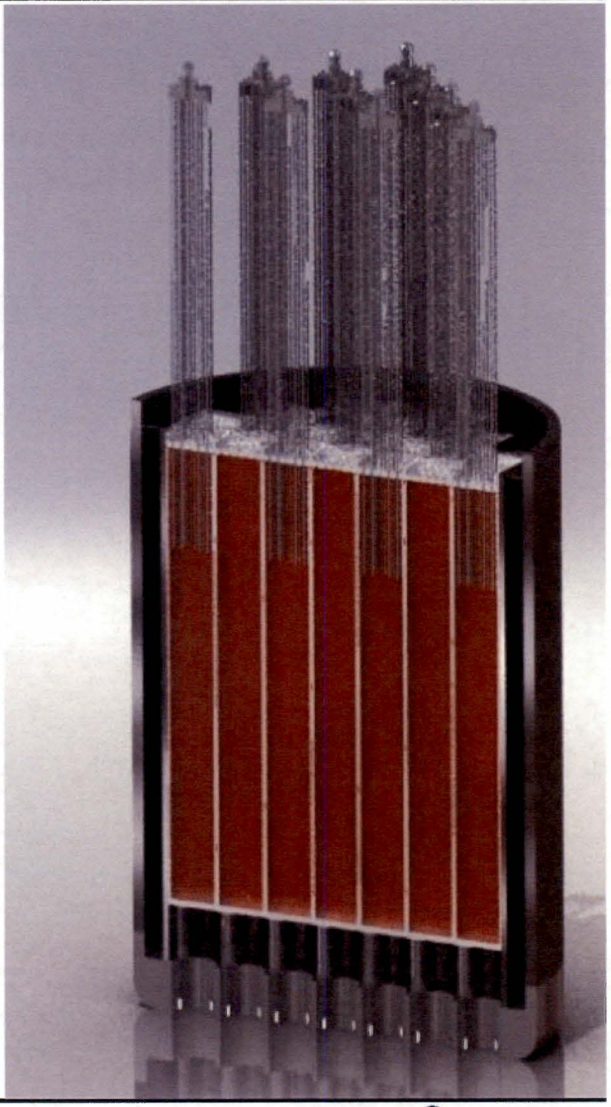


Unique Event Method (Flowchart)



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

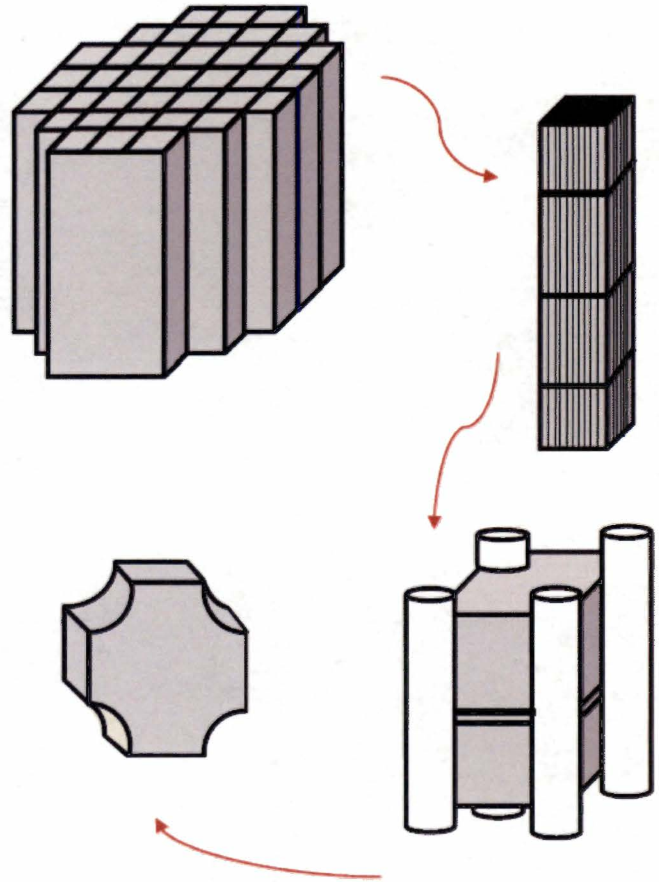


Dynamic Core Response

- 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

CHF Evaluation

- VIPRE-01: Model detailed thermal-hydraulics
- Evaluate critical heat flux (CHF) acceptance criteria
- Code shown to be appropriate in TR-0915-17564-A (Subchannel 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 performed with each application to holistically justify differences

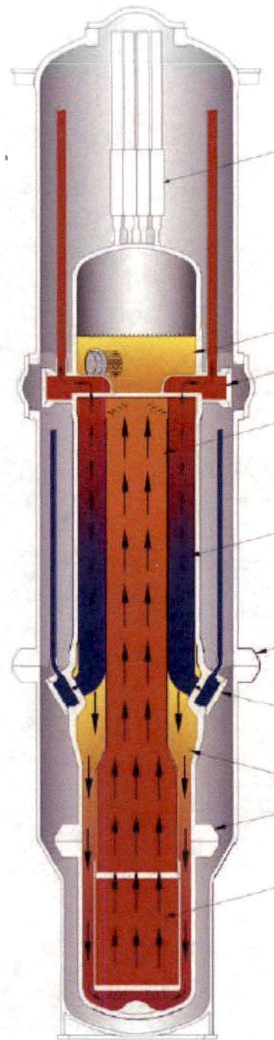


Adiabatic Fuel Heatup

- 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

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



NUSCALE Power for all humankind

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

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

Acronyms

- CHF – Critical Heat Flux
- GDC – General Design Criteria
- HZP – Hot Aero 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

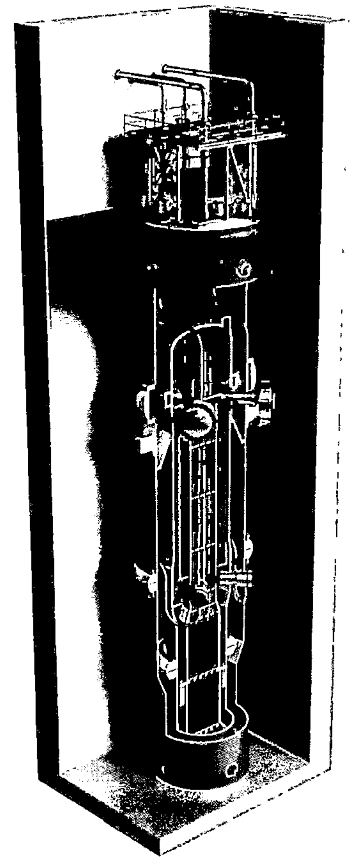
Portland Office
6650 SW Redwood Lane,
Suite 210
Portland, OR 97224
971.371.1592

Corvallis Office
1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330
541.360.0500

Rockville Office
11333 Woodglen Ave., Suite 205
Rockville, MD 20852
301.770.0472

Richland Office
1933 Jadwin Ave., Suite 130
Richland, WA 99354
541.360.0500

Charlotte Office
2815 Coliseum Centre Drive,
Suite 230
Charlotte, NC 28217
980.349.4804



http://www.nuscalepower.com
Twitter: @NuScale_Power

NUSCALE™
Power for all humankind