NON-PROPRIETARY

NuScale Power

Codes and Methods: Core Design

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November 20, 2008

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

Outline

- Introduction
- Computer Codes
- Code Applicability
- Sample Core Neutronics Calculations
- Summary

Introduction

- The NuScale Core Design will use the Studsvik CMS code suite
- Demonstrate the applicability of the CMS suite to the NuScale operating conditions
- A Review of an Initial Core Design

Computer Codes

- The Studsvik CMS suite includes:
  - INTERPIN-4 – Fuel Temperature Calculations
  - CASMO-4E – 2D Lattice Physics Code/Cross Section Generation
  - SIMULATE-3 – 3D Core Simulation with Depletion

Code Applicability

- The code and models for the CMS suite are Verified by the code vendor Studsvik Scandpower
- NuScale Power will Validate the codes for use in the NuScale Power Module under the NuScale QA program
- The validation process will consider:
  - NuScale fuel will be shorter then typical PWR fuel 1.35m vs. 3.65m
  - Appropriate flow modeling of a Natural Circulation Core (i.e. flow/power coupling, possible non uniform inlet flow, low flow rates)
  - Operating plant data for current PWR's
  - Critical Benchmark Experiment Comparisons (i.e. B&W Criticals, etc.)
  - Isotopic Measurement Comparisons (i.e. Yankee Rowe, etc.)

Code Applicability *(continued)*

- The validation of the CMS codes will:
  - Lead to the generation of appropriate Biases and/or Uncertainties
  - Demonstrate in-house expertise in the correct use of the codes

Sample Core Neutronics Calculations

NuScale Initial Core Design Basic Assumptions

- Core Thermal Power Output = 150 MWt
- Reactor Flow Rate = 600 kg/s (13,413 gpm)
- Core Bypass Flow = 4% reactor design flow
- Core Inlet Temperature = 218.4°C (425.2°F)
- 24 Month Cycle Length at 95% capacity factor (695 EFPD)

NuScale Initial Core Design Objectives

- Lower Power Density
- Minimize Power Peaking
  - Maximum Node Relative Power Factor ≤{ }[4a]
  - Assembly Average Radial Peaking Factor ≤{ }[4a]
- Negative Reactivity Coefficients
- Use Standard PWR Fuel Lattice

INTERPIN Calculations

- Single fuel pin and coolant channel calculation
- Generic 17x17 fuel assembly
  - See Table to Right
- Results
  - Average fuel temperature =
    { }4a

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Input Parameter	Value
Core Inlet Temperature	
Reactor Operating Pressure	
Cladding Inner Radius	
Cladding Outer Radius	
Active Fuel Height	
Back Fill Gas Pressure	
Upper Plenum Gas Volume	
Fuel Pellet Density	
Fuel Pellet Radius	
Percent Pellet Dish Volume	
Percent Pellet Chamfer Volume	
Core Average Linear Heat Rate	
Upper Plenum Length	
Per Rod Mass Flow	
Fuel Pin Pitch	

CASMO-4E Calculations

- 2-D Fuel Lattice Calculations
- An Integral Burnable Absorber Gadolinia (Gd_2O_3) is used
- Results input to CMSLINK to create a binary library for SIMULATE-3

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Parameter	Value
Core Average Power Density	
System Operating Pressure	
Average Fuel Temperature	
Core Inlet Temperature	
Core Average Coolant Temperature	
Cycle Average Boron Concentration	
Pins per Side of Assembly	
Pin Pitch	
Fuel Assembly Pitch	
Spacer Grid Mass Per Unit Height of Fueled Region	
Fuel Pellet Density	
Fuel Enrichment (w/o U235)	
Instrument/Guide Tube Inside Radius	
Instrument/Guide Tube Outside Radius	
Control Rod Absorber Radius	
Control Rod Inside Radius	
Control Rod Outside Radius	
Fuel Pellet Radius	
Fuel Rod Inside Radius	
Fuel Rod Outside Radius	

Initial Core Type A05 Assembly Design

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4a

CASMO-4E Eigenvalue Trajectories (Gadolinia Bearing Segments)

4a

CASMO-4E Maximum Relative Pin Peaking (No Gadolinia Segments)

4a

CASMO-4E Maximum Relative Pin Peaking (Gadolinia Bearing Segments)

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SIMULATE-3 Calculations

- Simulation of the 3-D core and all representative geometry
- The core simulation is done in three phases
  1. Base Core Model
  2. Base Depletion
  3. Branch Cases

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Parameter	Value
Number of Fuel Assembly Rows	
Number of Axial Nodes	
Fraction of Core Modeled	
# of Nodes per Assembly	
# of Radial Reflector Nodes	
Fuel Assembly Pitch	
Active Fuel Height	
Core Inlet Flow Density	
Core Inlet Temperature	
Core Rated Power	
System Operating Pressure	
Spacer Grid Height	
Spacer Grid Locations (measured from bottom of core)	
Control Rod # of Steps	
Control Rod Step Size	

Core Layout

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- [ ] 4a 17x17 Assemblies
- Core Encircling Diameter
  [ ] 4a
- Control Rod Clusters
  [ ] 4a
  - B4C Absorber Material
- Assembly Quantities
  - 05 - Type A00
  - 04 - Type A01
  - 12 - Type A02
  - 08 - Type A03
  - 04 - Type A04
  - 04 - Type A05

[ ] 4a

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Hot Full Power, All Rods Out Critical Boron Concentration

4a

Maximum Node Power Peaking (3PIN)

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Core Average Axial Power Shape (Fz)

4a

Axially Integrated Radial Power Distributions

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Beginning of Cycle (BOC)

}4a

Middle of Cycle (MOC)

}4a

End of Cycle (EOC)

Peak Assembly

Reactivity Coefficients

	BOC			MOC			EOC		
	HZP	70% Power and Flow	HFP	HZP	70% Power and Flow	HFP	HZP	70% Power and Flow	HFP
ITC pcm/°F									
MTC pcm/°F									
Doppler pcm/°F									

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Worst Rod Stuck Out

Worst Rod Stuck Out Initial Conditions

Case	Power Level	Flow	Inlet Temp.	Pressure	Fission Products
Hot Zero Power					
Hot Full Power					
Cold Shutdown					

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Beginning of Cycle (BOC)

Rod Configuration	Hot Zero Power		Hot Full Power		Cold Shut Down	
	Keff	Rod Worth %Δk/k	Keff	Rod Worth %Δk/k	Keff	Rod Worth %Δk/k
ARI-#1						
ARI						
ARI-#5						
ARI						
ARI-#6						
ARI						

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End of Cycle (EOC)

Rod Configuration	Hot Zero Power		Hot Full Power		Cold Shut Down	
	Keff	Rod Worth %Δk/k	Keff	Rod Worth %Δk/k	Keff	Rod Worth %Δk/k
ARI-#1						
ARI						
ARI-#5						
ARI						
ARI-#6						
ARI						

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Summary

- The NuScale Core Design will be performed with Studsvik CMS Suite of Codes
  - Validation of the codes will be done by NuScale
  - A validation plan is being created with assistance from Studsvik
- An Initial Core Design has been performed
  - Standard fuel lattice was used
  - The results show the core meets the design objectives

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NuScale Power

Codes and Methods: Fuels Analysis

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November 20, 2008

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Rockville, MD

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Outline

- Introduction
- Acceptance Criteria
- Computer Codes
- Methods
- Example COBRA-IV Subchannel Calculations
- Summary

Introduction

- Steady-state core/fuels performance analysis covers:
  - Core Neutronics ✔
  - **Core Thermal-Hydraulics**
  - **Thermal-Mechanical Fuels Analysis**
- Code and method adequacy considerations for the NuScale design
  - Subchannel cross-flow
  - Local core power and flow conditions due to natural circulation flow
  - Rod backfill pressure

Acceptance Criteria

- Departure from nucleate boiling ratio (DNBR) below 95/95 limit
- Fuel centerline temperature below fuel melt temperature
- Internal rod pressure below primary system pressure
- Cladding hoop strain below 1%

N-COBRA-TF

- Widely used in the nuclear industry for both steady-state and transient thermal-hydraulic analysis of both BWRs and PWRs
- Validation
  - Incorporates thermal-hydraulic and heat transfer models that have been extensively validated against experimental data
  - NuScale will reassess N-COBRA-TF against this experimental database
  - A determination will be made whether a physical model, numerical method or model as coded in N-COBRA-TF properly describes the phenomena to be expected in the NuScale core
- Only development work expected is for selection of appropriate CHF correlation

N-FRAPCON-3

- FRAPCON-3 represents over 30 years of knowledge regarding LWR nuclear fuels thermal-mechanical performance
- Generates initial conditions for transient fuel rod analysis by FRAPTRAN, the successor to the FRAP-T6 code, the companion transient fuel rod analysis code
- Validation
  - Incorporates thermal and mechanical models that have been extensively validated against irradiated light water reactor nuclear fuel data
  - The current NuScale module nuclear fuel design is a modified version of standard PWR 17x17 fuel assembly geometry. The only difference is a reduced assembly length.
  - The peak fuel rod burn-up is less than 65 GWd/MTU
  - N-FRAPCON-3 was developed to calculate fuel performance for LWR fuel with burnups of up to 76 GWd/MTU in its integral assessment with experimental data from 45 nuclear fuel rods
  - A determination will be made whether a physical model, numerical method or model as coded in N-FRAPCON-3 properly describes the phenomena to be expected in the NuScale fuel
- No development work is expected

Methods

- N-COBRA-TF
  - DNBR Limit
  - MDNBR/Steady-State Operation Map
- N-FRAPCON-3
  - Fuel Centerline Temperature
  - Cladding Hoop Strain
  - Internal Rod Pressure

Core Thermal-Hydraulics Calculations

- Peak fuel rod axial and radial power distribution and power level from SIMULATE-3
- N-RELAP5/K will provide the hottest core subchannel coolant inlet temperature, pressure, and flow rate for the fuel rod channel
- DNBR Limit
  - Fuel acceptance criterion used to ensure acceptable thermal margin
  - Derived from a statistical analysis
  - N-COBRA-TF will be run using a Monte Carlo approach to determine the DNBR limit in a statistical manner
  - The total number of simulations will be determined using Wilkes formula
- MDNBR/Steady-State Operation Map
  - Series of steady-state calculations to cover normal operation regime
  - MDNBR calculated for these cases shall be above the DNBR limit by at least 15%

Fuels Thermal-Mechanical Calculations

- Fuel Centerline Temperature, Cladding Hoop Strain, and Internal Rod Pressure
  - Peak fuel rod axial and radial power distribution and power level from SIMULATE-3
  - N-RELAP5/K will provide the hottest core subchannel coolant inlet temperature, pressure, and flow rate for the fuel rod channel
  - Standard 17x17 PWR fuel design parameters, adjusted to account for uncertainties in dimensions and material parameters
  - Sensitivities will be conducted to determine impact of:
    - Shorter fuel rod length
    - Lower reactor coolant system pressure
    - Natural convection driven core coolant flow

Example COBRA-IV
Core Subchannel Calculations

Detailed Core Model

- 1/8 Core Model
  - [ ] [4a] Fuel Rods
  - [ ] [4a] Guide Tubes
  - [ ] [4a] Instrumentation Tubes
  - [ ] [4a] Channels

[ ] [4a]

[ ] [4a]

Full and partial elements

Detailed Assembly Model

- COBRA-IV Model
  - { }4a Fuel Rods
  - { }4a Guide Tubes
  - { }4a Instrumentation Tube
  - { }4a Channels

{ }4a

{ }4a

Full and partial elements

Base Case: 1/8 Core Model

- 600 kg/s coolant flow
  (4% bypass flow)
- [ ]4a inlet temperature
- 8.8 MPa system pressure
- Peak local power factor of [ ]4a
- UO2 pellets with Zircaloy cladding
- [ ]4a spacer grids, no mixing vanes

[ ]4a

Sensitivity Matrix

System Pressure	8.8 MPa (1276 psia)	9.0 MPa (1305 psia)	9.2 Mpa 1305 psia	9.4 Mpa 1335 psia	9.6 Mpa (1365 psia)
Inlet Temperature	486 K (415°F)	489 K (420°F)	491 K (425°F)	494 K (430°F)	497 K (435°F)
Mass Flow Rate	600 kg/s (1320 lbm/s)	650 kg/s (1430 lbm/s)	700 kg/s (1540 lbm/s)	750 kg/s (1650 lbm/s)	800 kg/s (1760 lbm/s)

Sensitivity Results: MDNBR

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Sensitivity Results: MDNBR

(continued)

	System Pressure (MPa)	Inlet Flow Temperature (K)	Core Flow Rate (kg/s)	Minimum DNBR
Base Case				
System Pressure Sensitivity				
Inlet Flow Temperature Sensitivity				
Core Flow Rate Sensitivity				

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Effect of Mixing Vanes

- Assembly Model
  - Mixing Vanes Located every other interstices position (3 total mixing vane grids)
  - Vane Angles (2.9 – 8.6^o)
  - Forcing function (0 – 15%)
  - Results

%	Peak (^{o}K)	δ Peak-Avg (^{o}K)	% reduction

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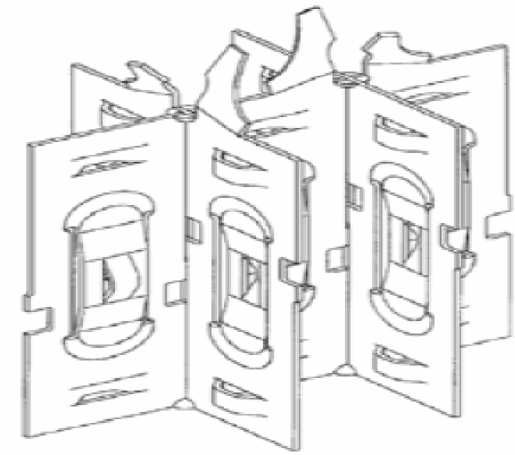

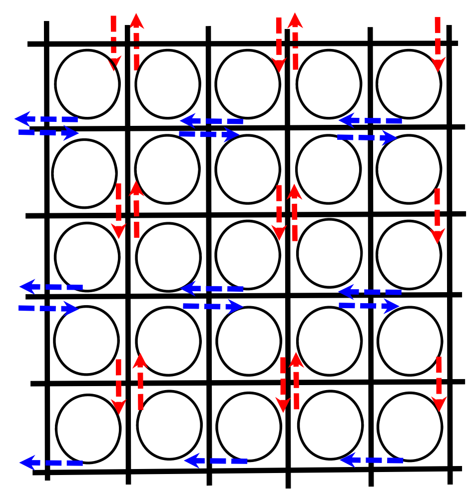

Effect of Mixing Vanes

(continued)

- *Little or no effect seen for range examined*

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Summary

- N-COBRA-TF and N-FRAPCON-3 will be used to ensure compliance with fuel acceptance criteria
- These codes have been used in the analysis of typical PWRs
- The codes have been previously assessed against an extensive database of LWR experimental benchmarks
- Developmental assessment plans are being prepared by NuScale in accordance with RG 1.203
- Preliminary calculations will inform developmental assessment plans
- At this time, the only code development work expected is to implement an appropriate CHF correlation in N-COBRA-TF

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NUSCALE POWER

Codes and Methods: Non-LOCA Analysis

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Outline

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- Acceptance Criteria
- Computer Codes
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Introduction

- Non-LOCA analysis covers:
  - Anticipated Operational Occurrences (AOOs)
  - Postulated Accidents (PAs) that don't result in a loss of primary coolant
- AOOs typically result from one of the following:
  - A single component failure
  - A single malfunction, including passive failures such as leaks or minor pipe breaks, which could occur during the life of the plant while the plant is operating
  - A single operator error
- Considering NuScale Unique Features
  - Loss-of-primary-flow events not applicable (no pumps)
  - Containment flooding is a new event to be analyzed (most likely bounded by main steam line break)
  - Planning to eliminate control rod ejection accidents by design

Acceptance Criteria: AOOs

- MDNBR greater than or equal to 95/95 DNBR limit
- Primary and secondary pressures not to exceed 110% design pressure
- Maximum fuel temperature less than melting point
- "An AOO shall not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the reactor coolant system or reactor containment barriers"

Acceptance Criteria: Non-LOCA PAs

- Primary and secondary pressures kept below brittle-fracture and ductile-failure limits
- Fuel cladding integrity maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR falls below the limit, the fuel is assumed to fail
- Maximum radiological consequences less than 25 rem total effective dose equivalent (TEDE)
- “A postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system”

Computer Codes

- **N-RELAP5/K (Primary and Secondary System Response)**
- N-COBRA-TF (Steady-State Overpower Map & Transient Calc) ✔
- **SIMULATE-3K (3-D Core Neutronics)**
- N-FRAPCON-3 (Steady-State Linear Heat Generation Map) ✔
- **N-FRAPTRAN (Time-Dependent Fuel Centerline Temperatures)**

N-RELAP5/K

- RELAP family of codes used extensively by industry
- Appendix K and improved steam generator models will be added to base code and new code will be called N-RELAP5/K
- Validation
  - Phenomena Identification and Ranking Table (PIRT) will be completed for non-LOCA events
  - NuScale will develop a non-LOCA validation suite that will overlap considerably with the LOCA validation suite and leverage additional data from OSU test facility
  - Developmental assessment plans are being developed in accordance with RG 1.203

SIMULATE-3K

- Approved by the NRC for LWR reactivity insertion transients and dynamic rod worth measurements for a variety of operating PWRs
- Verification and Validation
  - Studsvik has conducted extensive verification of SIMULATE-3K under their NQA-1 quality assurance program
  - Code has been evaluated against various international reactor transient calculations
  - NuScale will conduct additional in-house assessments of SIMULATE-3K to quantify uncertainties

N-FRAPTRAN

- FRAPTRAN was developed under NRC funding for calculating transient fuel rod thermal-mechanical response
- FRAPTRAN models and methodology benefit from more than 30 years of development effort involve the FRAPT series of codes
- Validation
  - NuScale module fuel design is within the parameters that FRAPTRAN was developed to analyze
  - Non-LOCA AOOs and PAs for NuScale module are a subset of those previously analyzed by FRAPTRAN for a typical PWR
  - Developmental assessment plans are being developed in accordance with RG 1.203
  - No development work expected

Methods

- Slow-Acting Transient Calculations
- Fast-Acting Transient Calculations

The "speed" of a transient will be based on a evaluation of the rate of change of the local core power during the event.

Slow-Acting Transients

- Primary, secondary, and containment response calculated by N-RELAP5/K using point-kinetics for neutronic feedback
- Local core thermal-hydraulics will be calculated by N-COBRA-TF run in transient mode
- A linear heat generation map will be generated using N-FRAPCON-3 that will be used to determine conformance to fuel acceptance criteria

Fast-Acting Transients

- Primary, secondary, and containment response calculated by N-RELAP5/K using SIMULATE-3K for neutronic feedback
  - Codes will not be “coupled”
  - Codes will be executed together using scripts
  - These scripts will also manage the flow of data between the two codes
- N-COBRA-TF will be run in transient mode using time-dependent output from N-RELAP5/K and SIMULATE-3K
- N-FRAPTRAN will be run in transient mode using time-dependent output from N-RELAP5/K (or N-COBRA-TF) and SIMULATE-3K

Summary

- NuScale codes and methods for non-LOCA analysis have a well-established bases
- Unique aspect of NuScale design has led to some differences in initiating event selection (e.g., loss of flow, containment flooding, control rod ejection)
- A combination of five codes will be used to calculate conformance to acceptance criteria. These codes are well-known and have been used extensively for analyzing typical PWRs.
- Developmental assessment plans are being prepared in accordance with RG 1.203 to address code adequacy to the NuScale design
- Development work is only expected for one code: N-RELAP5/K
- Two main methods have been identified for analyzing non-LOCA events in the NuScale design
- Additional testing will be run at OSU to confirm integral system performance

NuScale Power

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NuScale Power

Codes and Methods: LOCA Analysis

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Outline

- Introduction
- Acceptance Criteria
- Evaluation Model Development and Assessment Process (EMDAP)
- Computer Codes
- Appendix K Required Modifications
- Integral Assessment of Generic Code RELAP5 3.3
- Sample NuScale Power Module LOCA Calculations

Introduction

- NuScale ECCS Evaluation Model (EM)
  - Developed following guidance described in RG 1.203
  - Conform to 10 CFR 50, Appendix K
- LOCAs are defined as hypothetical breaks in the reactor coolant pressure boundary pipes up to and including a double-ended rupture of the largest pipe in the reactor coolant system
  - Double-ended guillotine breaks are precluded in the NuScale module by design
  - A single ECCS EM will be used for analyzing the entire potential break spectrum
- The break spectrum for the NuScale module will be developed according to requirements described in SRP Chapter 15

Acceptance Criteria: LOCAs

(10 CFR 50.46)

- Calculated maximum fuel clad temperature shall not exceed 2200 °F
- Calculated total oxidation of the cladding thickness shall nowhere exceed 17%
- Calculated total amount of hydrogen generated from cladding shall not exceed 1% of potential amount generated from cladding surrounding the fuel
- Calculated changes in core geometry shall be such that the core remains amenable to cooling
- Long term cooling must be demonstrated

ECCS EM Development

- Evaluation Model Development and Assessment Process (EMDAP)
  - Element 1 - Establish Requirements for Evaluation Model Capability
  - Element 2 - Develop Assessment Base
  - Element 3 - Develop Evaluation Model
  - Element 4 - Assess Evaluation Model Adequacy
- The EMDAP process has been reviewed for potential issues for development of a NuScale EM, and none were found

Computer Codes

- **N-RELAP5/K**
  - Primary and Secondary System Response
- SIMULATE-3 ✔
  - 3-D Core Neutronics, Axial and Radial Power Distributions
- N-FRAPCON-3 ✔
  - Steady-State Fuel Temperatures for Initial Stored Energy

N-RELAP5/K

- RELAP family of codes used extensively by industry
- Appendix K and improved steam generator models will be added to base code and new code will be called N-RELAP5/K
- Validation
  - A preliminary PIRT has been completed for LOCA events
  - LOCA phenomena expected in the NuScale module are essentially a subset of the LOCA phenomena expected in a typical PWR
  - NuScale will develop a LOCA validation suite that will overlap considerably with the non-LOCA validation suite and leverage additional data from the OSU test facility
  - Developmental assessment plans are being developed in accordance with RG 1.203

N-RELAP5/K: Appendix K Modifications

- Seven modifications are identified for RELAP5 3.3 to be brought into compliance with 10 CFR 50 Appendix K requirements:
  1. Add ANS proposed 1971 decay heat model
  2. Add Baker-Just model for rate of energy release, hydrogen generation, and cladding oxidation
  3. Add Moody model for 2ϕ discharge rate calculation
  4. Add one of the six different steady-state CHF correlations for critical heat flux
  5. Add post-CHF regime transition and film boiling heat flux correlations. Potentially, the Groeneveld correlation for high flow film boiling conditions; the modified Bromley for low-flow film boiling conditions; and McDonough, Milich and King correlation for transition boiling
  6. Add logic to lock out nucleate boiling correlations, at any location, subsequent to CHF, during blowdown period
  7. Appendix K modeling of the hot region (a single fuel assembly, cross flow, cladding swelling/rupture flow blockage, restricting flow changes to having a period of less than 0.1 seconds)

Sample Integral System Benchmark

- 1/3-scale Integral Test Facility of the MASLWR reactor concept was constructed at Oregon State University
- The RELAP5 input model used form loss coefficients based on experimental steady-state flow tests in the facility and were not “tuned” to achieve better comparisons
- For the tests carried out in the MASLWR test facility, accompanying calculations using the two versions of the RELAP5 code were completed: RELAP5-3D and RELAP5/MOD3.3

Test #	Simulation	Low ADS 1 (%)	Low ADS 2 (%)	High ADS 1 (%)	High ADS 2 (%)	Sump Recirc. 1 (%)	Sump Recirc. 2 (%)
OSU-MASLWR-001	Inadvertent Acutation of 1 Submerged ADS Valve	Failed Shut	100	100	100	100	100
OSU-MASLWR-002	Natural Circulation at Core Power up to 210 kW	-	-	-	-	-	-
OSU-MASLWR-003A	Natural circulation at Core Power of 210kW (Continuation of test 002)	-	-	-	-	-	-
OSU-MASLWR-003B	Inadvertent Actuation of 1 High Containment ADS Valve	Failed Shut	100	Failed Shut	100	100	100

Integral Benchmark RELAP5 MASLWR Nodalization

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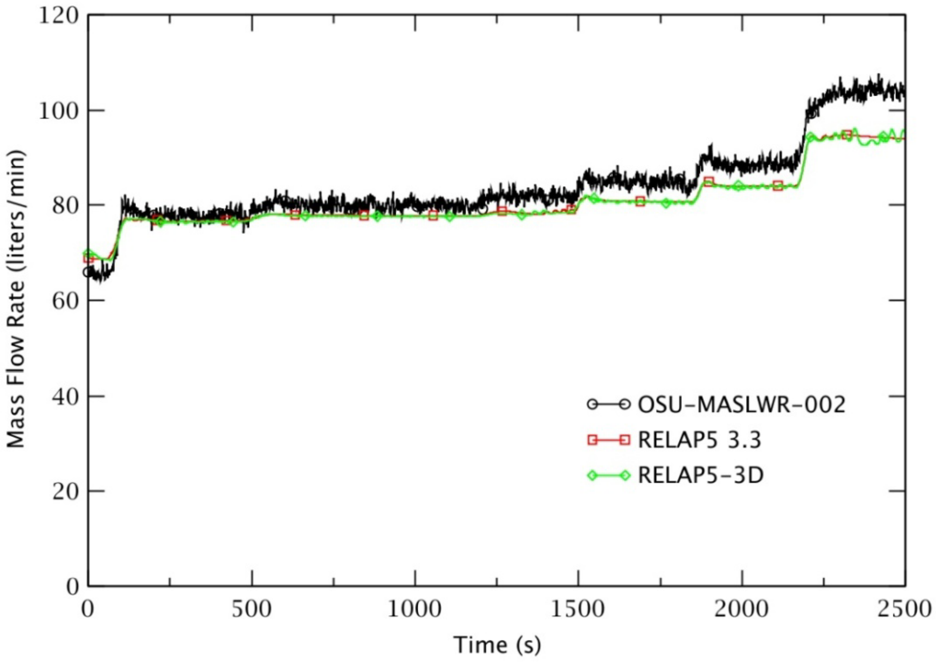
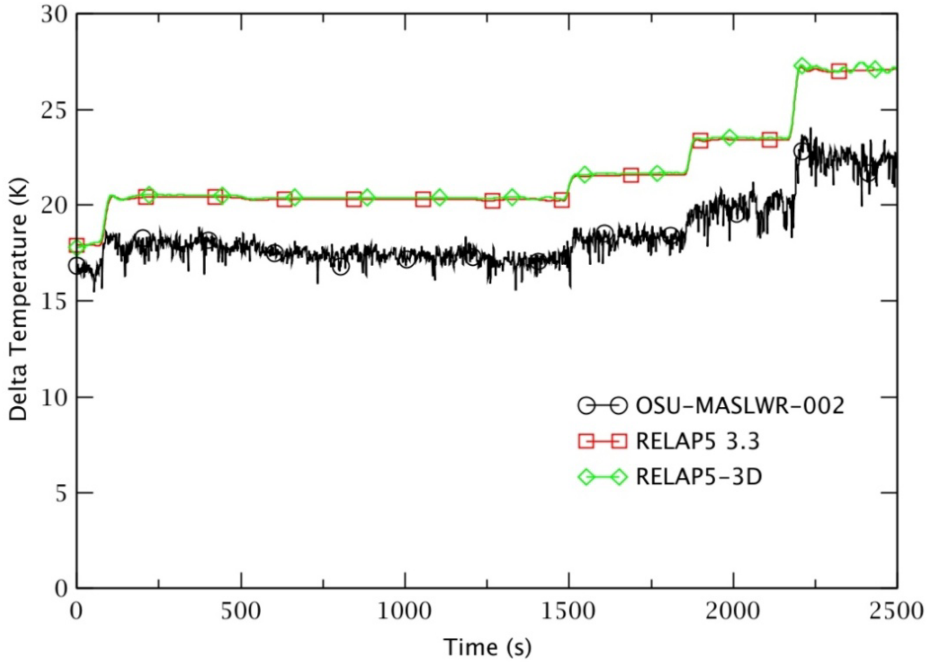
Integral Benchmark: OSU-MASLWR-002

PRIMARY MASS FLOW RATES

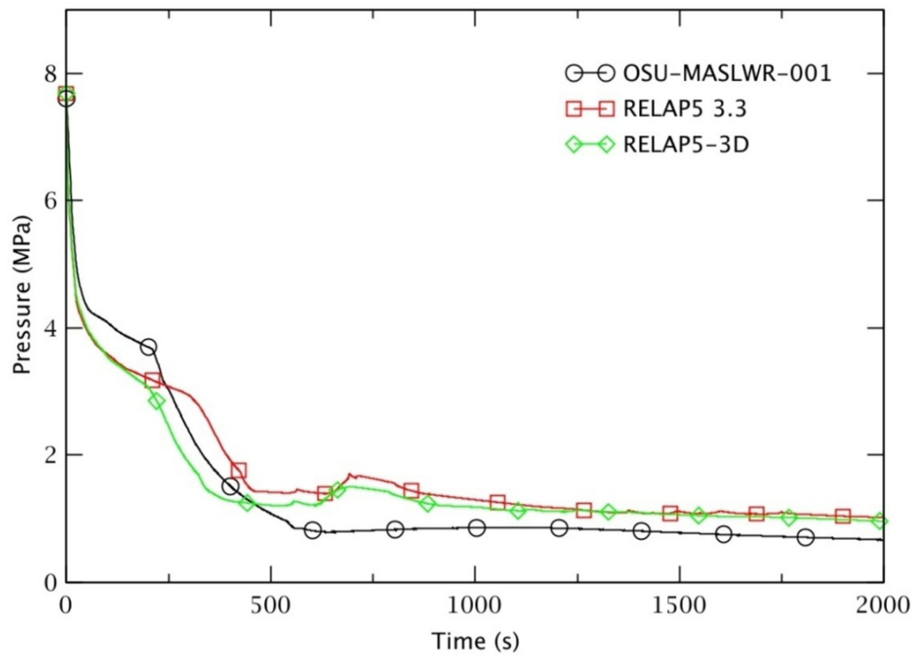


CORE TEMPERATURE RISE

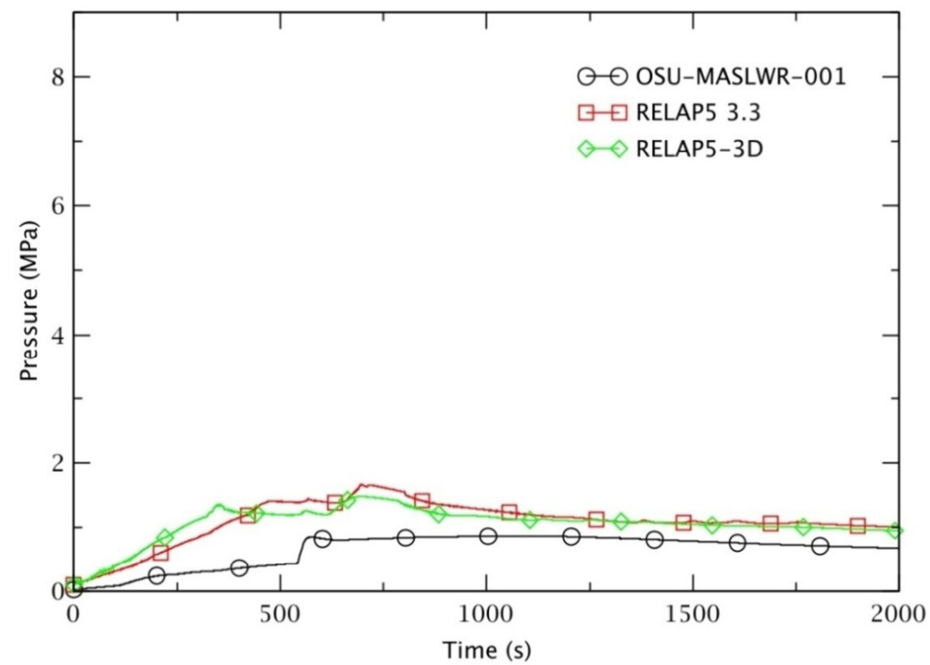


Integral Benchmark: OSU-MASLWR-001

PRIMARY SYSTEM PRESSURE

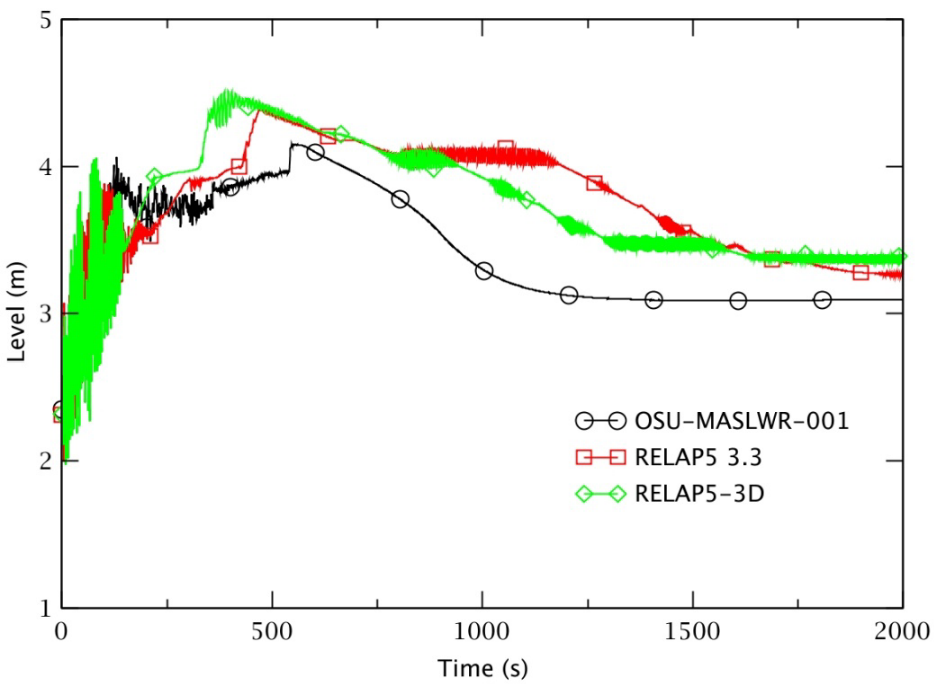


CONTAINMENT PRESSURE

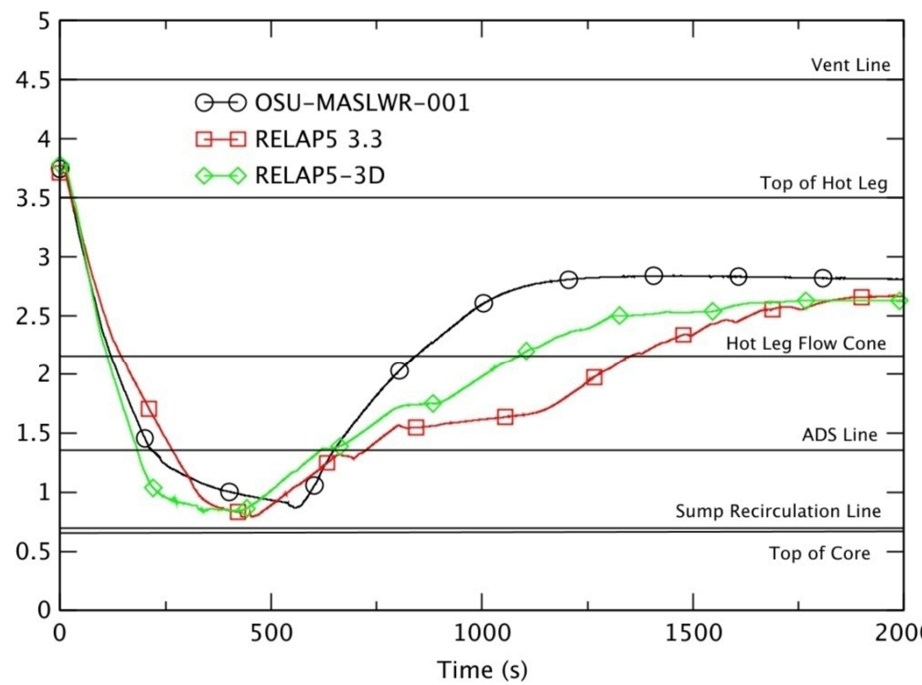


Integral Benchmark: OSU-MASLWR-001 *(continued)*

COLLAPSED CONTAINMENT LIQUID LEVEL



COLLAPSED DOWNCOMER LIQUID LEVEL



Sample LOCA Calculations

- Two sample cases
  - Dual RVV break
  - Dual RRV break
- Full scale RELAP5 3.3 input model of the NuScale Power Module
  - Total core peaking of { }4a, with axial and radial chopped cosine distributions
  - 3-channel core (w/o cross-flow) was modeled
  - Fuel was modeled w/o gap resistance
  - All major heat structures were included except hot leg riser
  - Core scram delayed { }4a seconds from LOCA initiation
  - No feedwater accumulators credited, and the SGs immediately isolated
- Results
  - No core uncovery for either case
  - No core heat up observed
  - Maximum cladding temperature was the initial cladding temperature

RELAP5 Nodalization

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Dual RVV Calculations

PRIMARY/SECONDARY PRESSURES

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PEAK CLADDING TEMPERATURE

{ }4a

Dual RVV Calculations *(continued)*

COLLAPSED RISER LIQUID LEVEL

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COLLAPSED CONTAINMENT LIQUID LEVEL

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Dual RRV Calculations

PRIMARY/SECONDARY PRESSURES

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PEAK CLADDING TEMPERATURE

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Dual RRV Calculations *(continued)*

COLLAPSED RISER LIQUID LEVEL

4a

COLLAPSED CONTAINMENT LIQUID LEVEL

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Primary Natural Circulation Flow Calculations

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Dual RVV Case

Dual RRV Case

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Pressure Comparison: RVV and RRV Cases

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Summary

- NuScale codes and methods for LOCA analysis have well-established bases
- N-RELAP5/K will be the main code used for LOCA analysis and be modified to include Appendix K and NuScale steam generator models
- Developmental assessment plans are being prepared in accordance with RG 1.203 to address code adequacy to the NuScale design and are being informed by:
  - PIRT results
  - Integral System Benchmark Assessments
  - Preliminary LOCA Calculations
- Additional testing will be run at OSU to confirm integral system performance

NuScale
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NON-PROPRIETARY

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Codes and Methods: Containment Analysis

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Outline

- Introduction
- Acceptance Criteria
- Containment Phenomena
- Sample RELAP5 Calculations
- Methods
- Summary

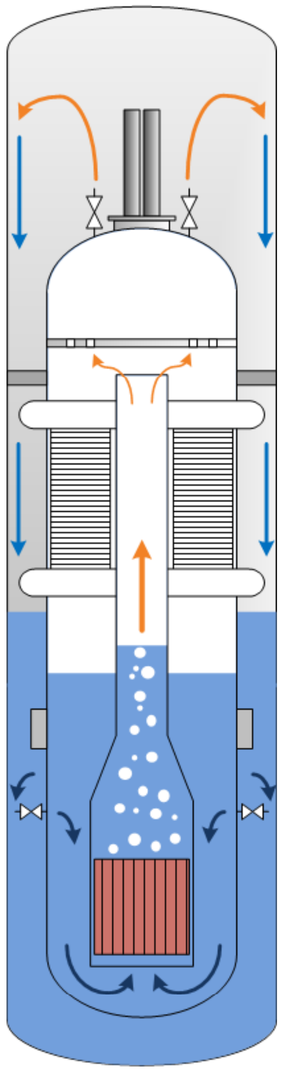
Introduction

- N-RELAP5/K will be used to analyze NuScale containment performance
- N-RELAP5/K verification and validation efforts for ECCS EM will be leveraged for containment performance analysis
- Preliminary LOCA PIRT covered containment phenomena

Acceptance Criteria

- General Design Criteria
  - GDC 16: Reactor containment and related systems provide: (1) leaktight barrier against uncontrolled release of radioactivity to the environment and (2) ensure design conditions important to safety are not exceeded…for as long as required
  - GDC 50: Reactor containment structure be designed so that all comprising systems/structure accommodate calculated (P,T) caused by a LOCA
- SRP Expectations
  - All potential force loadings on containment by a primary or secondary break into containment are evaluated (SRP Section 6.2)
  - Force loading is provided to the structural/mechanical team for evaluation with respect all possible loading combinations (SRP Section 3.8.2.I.5)
  - Limiting scenario may arise from localized effects of a smaller break in combination with max (P,T)

Containment Phenomena



STABLE LONG-TERM CONDITION

- Reactor Vent Valves Open
- Containment Vessel Partially Filled
- Reactor Recirculation Valves Open
- Containment and Reactor Vessel Cooling

IMPORTANT PHENOMENA

- Critical Break flow (single-phase & two-phase)
- Condensation
- Boiling

Methods

- The NuScale containment design simplicity and size does not require using a specialized containment code (e.g., GOTHIC, CONTEMP)
- Worst case (max & min) pressure and temperature transients will be identified
- External/environmental effects are not analyzed due to separate external shield structures
- Conservatisms will be included:
  - Condensation modeling
  - Calculated boiling on the exterior of the reactor
  - Direct convection cooling of the reactor vessel
- Direct heating and cooling of localized portions of the containment vessel are possible
- Coupled containment /reactor vessel heat transfer
- Leak before break may be considered (SRP Section 3.6.3)

RELAP5 Sample Containment Calculations

4a

Summary

- N-RELAP5/K will be used to calculate conformance to acceptance criteria
- Developmental assessment plans are being prepared in accordance with RG 1.203 to address code adequacy to the NuScale design
- NuScale codes and methods for containment analysis have well-established bases
- Additional testing will be run at OSU to confirm containment performance

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