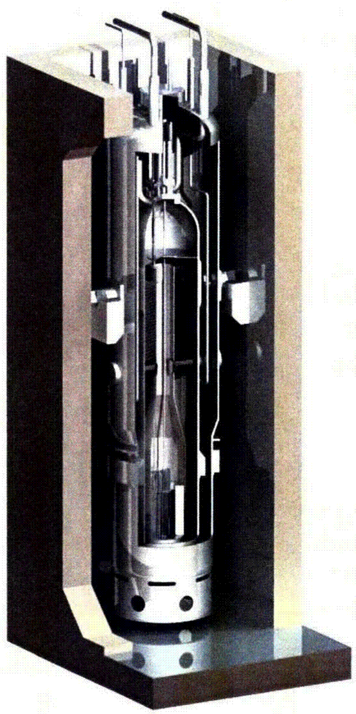
NuScale Power

Introduction



Dr. Paul Lorenzini

Chief Executive Officer

November 20, 2008

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

Proposed Pre-Application Schedule

	FY2008	FY2009		
	4Q	1Q	2Q	3Q
1st Meeting • NuScale and Design Introduction	▼			
Submit Design Description Report		▼		
2nd Meeting • Codes and Methods Topical Report		▼		
3rd Meeting • Online Refueling Topical Report • Multi-Module I&C and Operations Staffing Topical Report			▼	
4th Meeting • Multi-Module PRA Topical Report • Severe Accidents Topical Report • Dose Calculations and Emergency Planning Topical Report				▼

Purpose

- Present NuScale's plan for codes and methods that will be used in core design and safety analyses
- Share our integral system testing plan
- Assure consistency with the current light-water reactor (LWR) licensing framework

Codes and Methods

- Unique NuScale Design Features
  - Small, reduced height core
  - Natural circulation under normal operation
  - Passive safety systems
- Key Elements to Address
  - Core and Safety analysis calculational framework
  - Selection of computer codes
  - Verification and validation plans
  - Experimental programs and applicability of existing LWR benchmark data
  - Planned integral system testing as it relates to verification and validation of these methods

Agenda

NuScale Plant Design Overview	José Reyes
Summary of Codes and Methods	José Reyes
***Core Design**	Brandon Haugh
***Fuel Performance Analysis Methods**	Kent Welter
***Non-LOCA Methods**	Kent Welter
***LOCA Methods**	Eric Young
***Containment Performance Analysis Methods**	Eric Young

**Contains Proprietary Information*

NuScale Power

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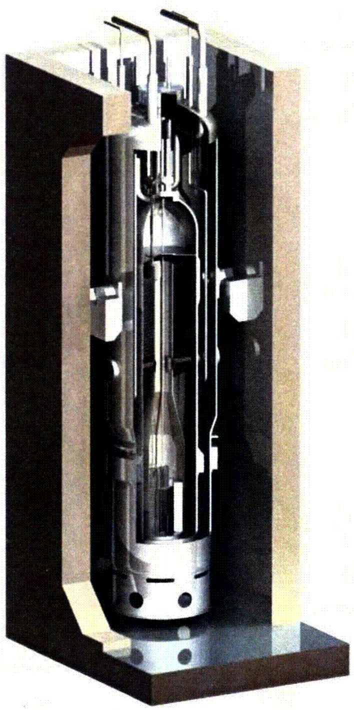
Overview of NuScale Design



Dr. José N. Reyes, Jr.

Chief Technical Officer

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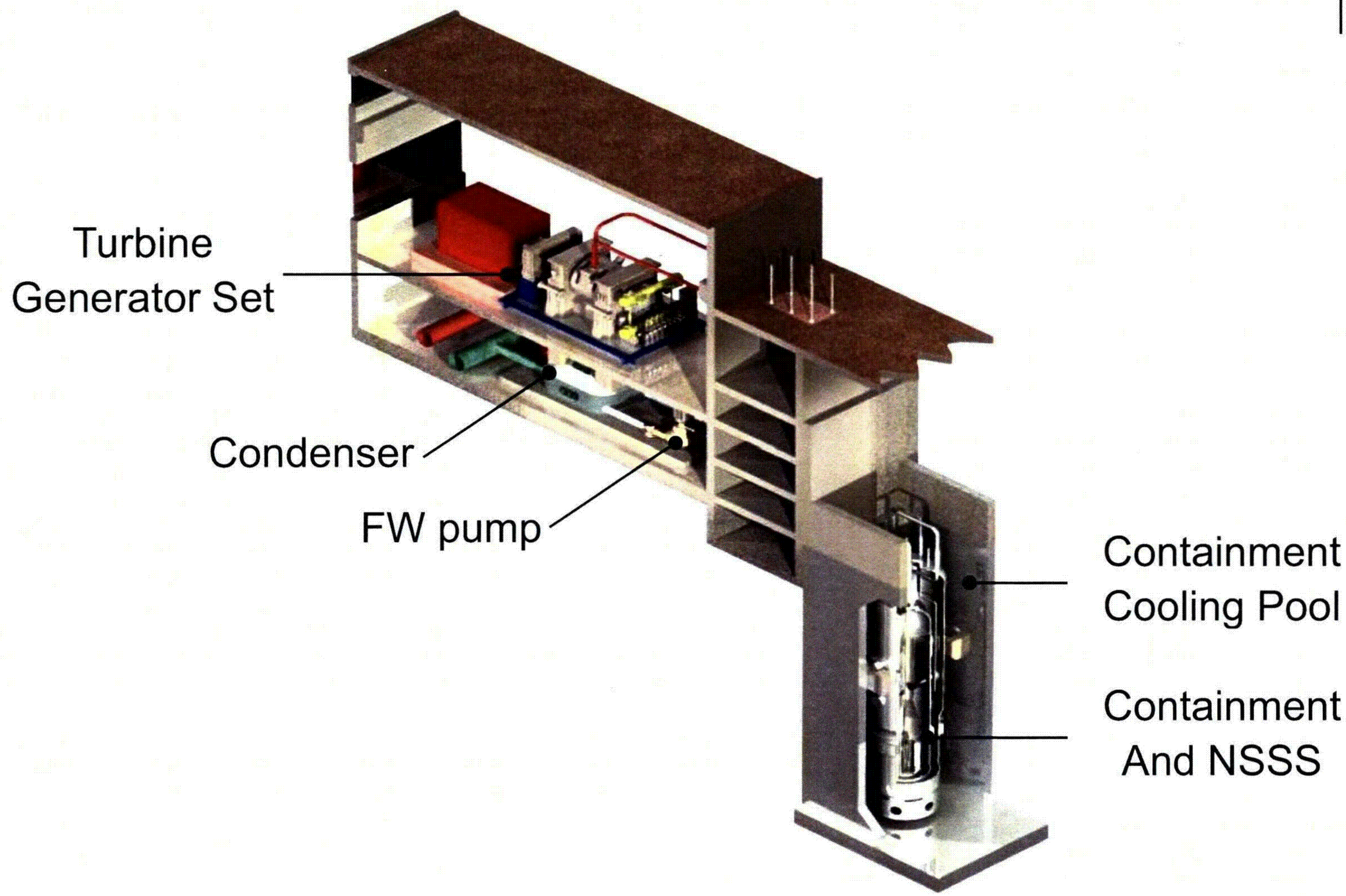
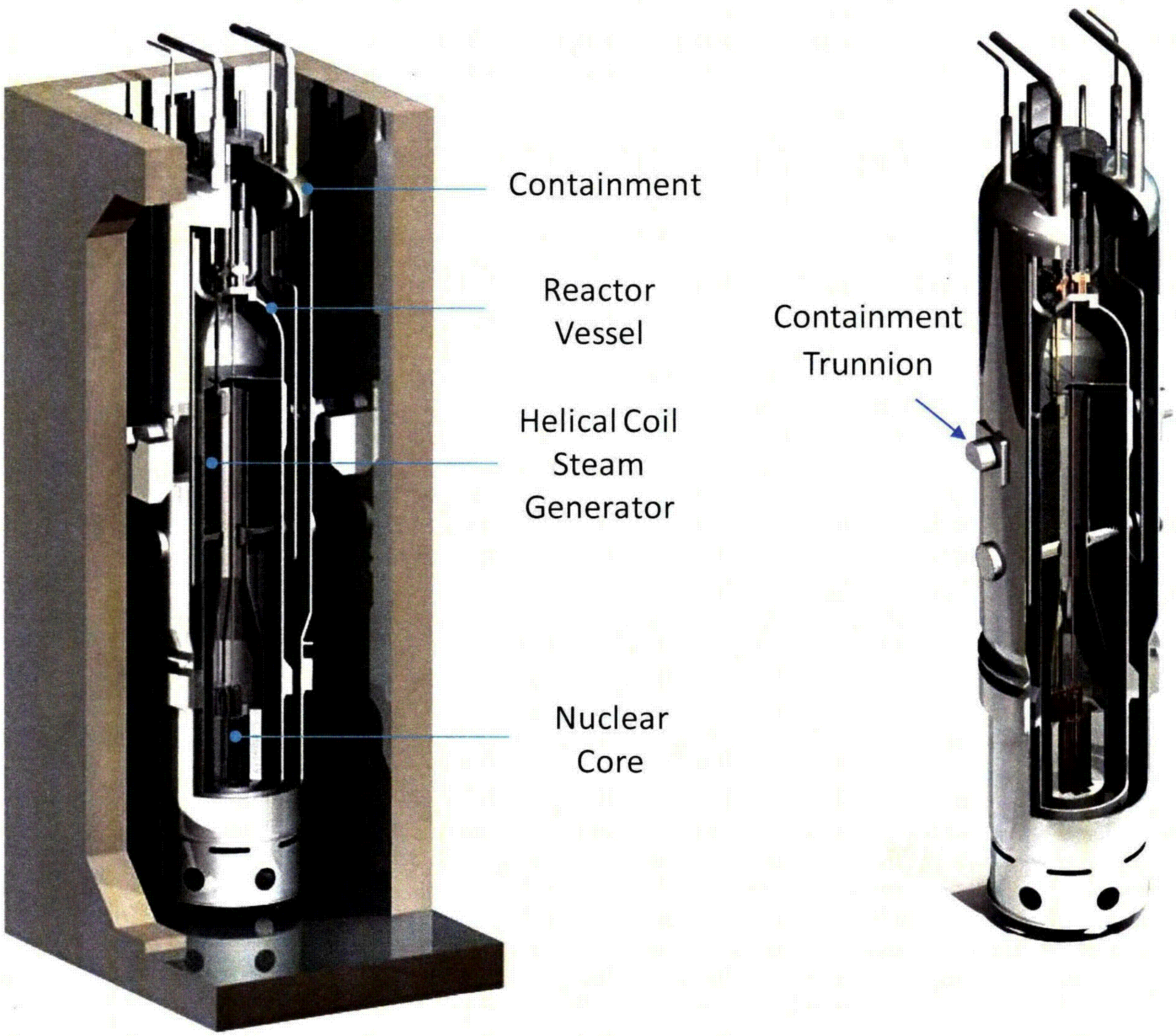
Outline

- Single Power Module
- Multi-Module Plant
- Engineered Safety Features

Power Generation Unit

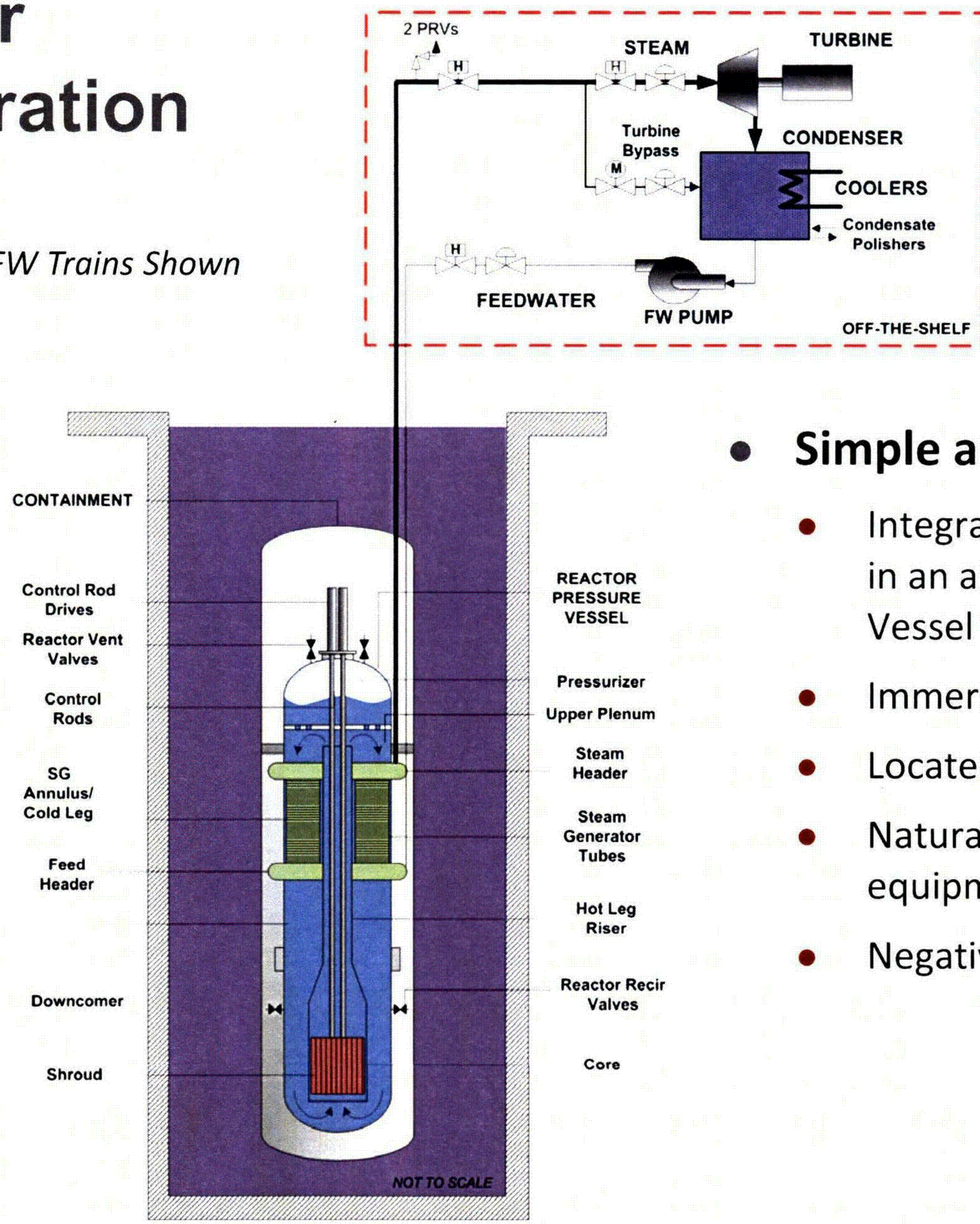


NSSS and Containment



Power Generation Unit

One of Two FW Trains Shown



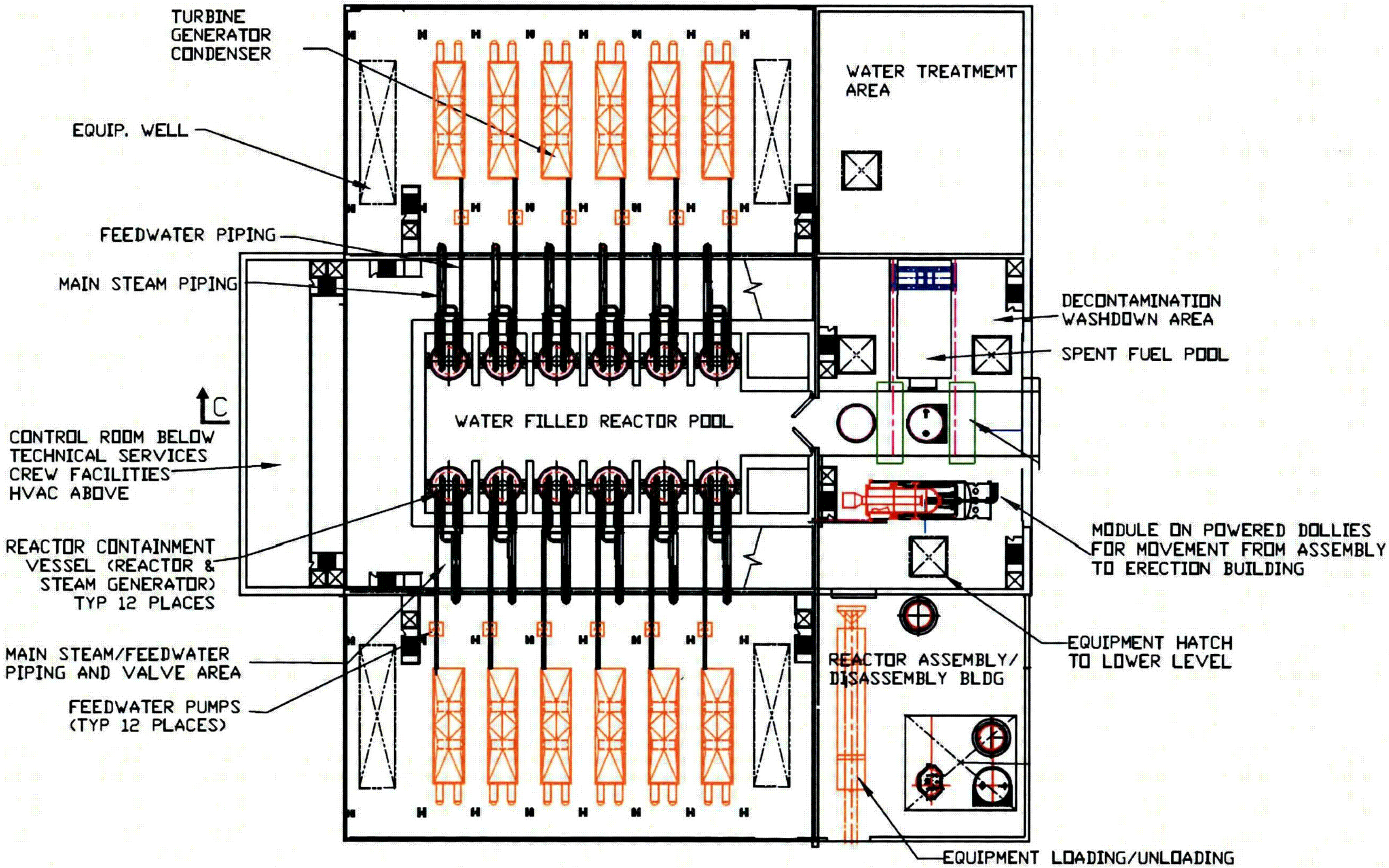
- **Simple and Robust Design**
  - Integrated Reactor Vessel enclosed in an air evacuated Containment Vessel
  - Immersed in a large pool of water
  - Located below grade
  - Natural Circulation eliminates equipment and components
  - Negatively buoyant

Preliminary Plant Parameters

Overall Plant	
• Nominal Electrical Output	540 MW(e)
• Nominal Station Efficiency	30%
• Number of Power Generation Units	12
• Nominal Plant Capacity Factor	> 90%
Power Generation Unit	
• Number of Reactors	One
• Nominal Electrical Output	45 MW(e)
• Steam Generator Number	Two independent tube bundles
• Steam Generator Type	Vertical helical tube
• Steam Cycle	Slightly superheated
• Turbine Type	3600 rpm, single pressure
• Turbine Throttle Conditions	3.1 MPa/264 °C (450 psia/507 °F)
Module	
• Thermal Power Rating	150 MWt
• Cold Leg/Hot Leg Temperature	247.9 °C/288.85 °C (478.1 °F /551.9 °F)
• Coolant Mass Flow Rate	~700 kg/s
• Operating Pressure	~9.0 MPa
Reactor Core	
▪ Fuel	UO_2 (< 4.95% enrichment)
▪ Refueling Intervals	24 months

Multiple-Module Complex – Example Layout

12 modules - 540 MW(e) Nominal Power

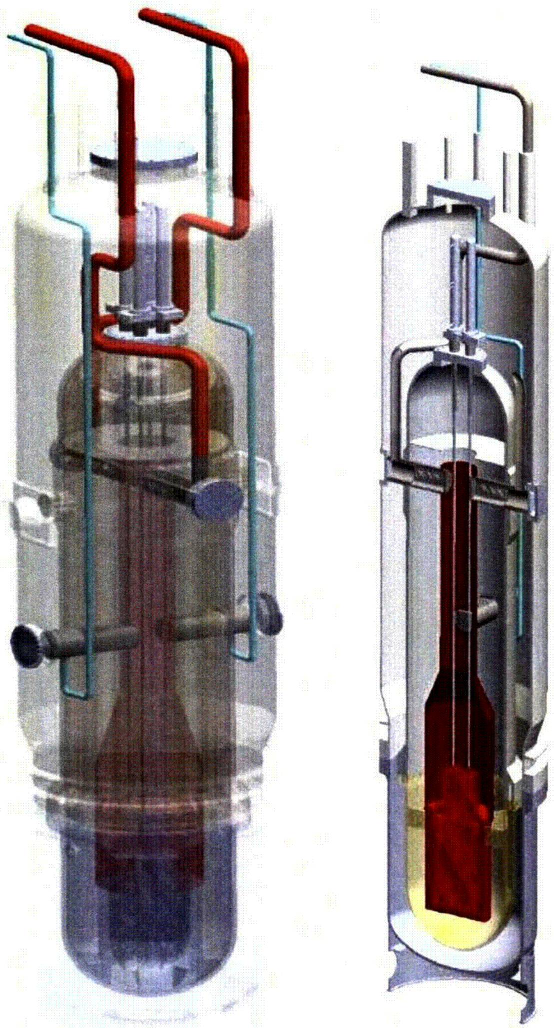


Engineered Safety Features

- High Pressure Containment Vessel
- Decay Heat Removal System (DHRS)
- Containment Heat Removal System (CHRS)
- Emergency Core Cooling System (ECCS)
  - Reactor Vent Valves
  - Reactor Recirculation Valves
  - CHRS

High Pressure Containment

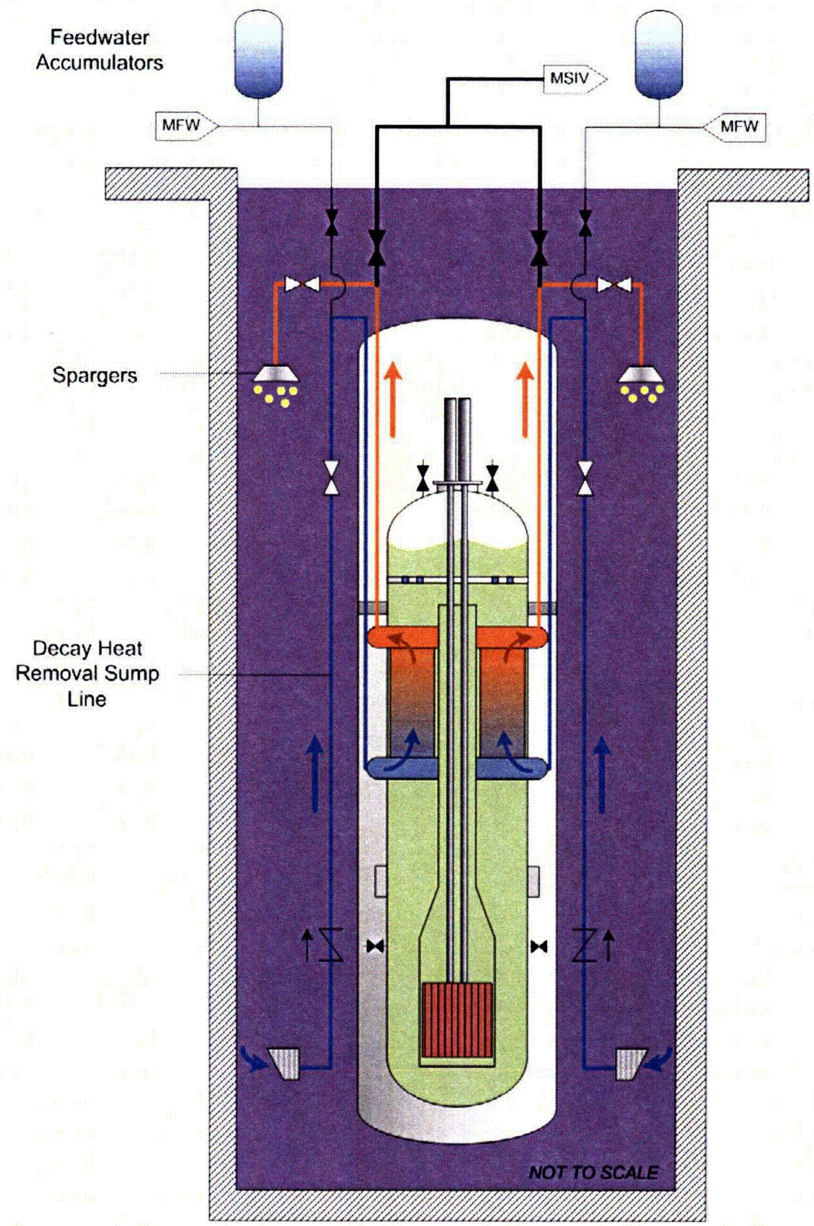
Enhanced Safety



- **Capable of ~3.1 MPa (450 psia)**
  - Peak equilibrium pressure between reactor and containment following any LOCA is always below containment design pressure
- **Insulating Vacuum**
  - Significantly reduces convection heat transfer during normal operation
  - No insulation on reactor vessel. ELIMINATES SUMP SCREEN BLOCKAGE ISSUE (GSI-191).
  - Improves steam condensation rates during a LOCA by eliminating air
  - Prevents combustible hydrogen mixture in the unlikely event of a severe accident (i.e., no oxygen)
  - Eliminates corrosion and humidity problems inside containment

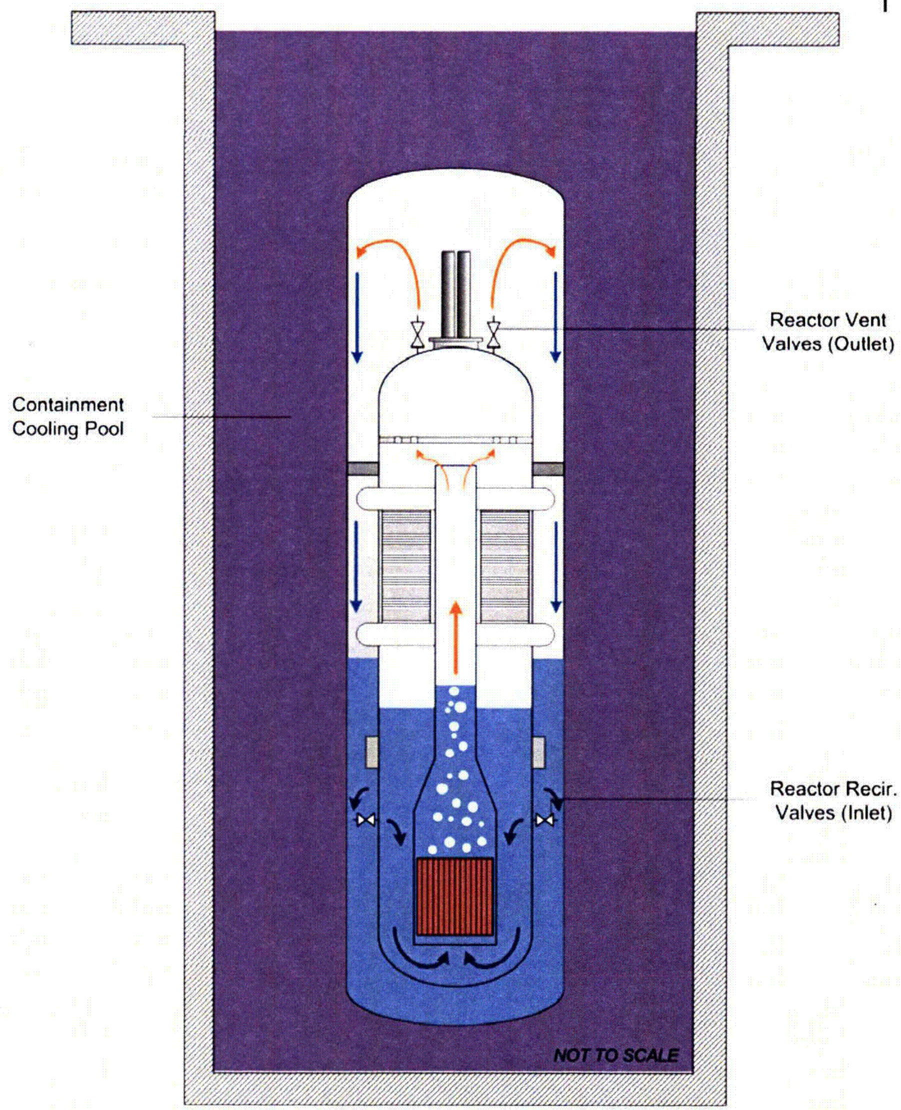
DHRS

- Two independent trains of emergency feedwater to the steam generator tube bundles
- Water is drawn from the containment cooling pool through a sump screen
- Steam is vented through spargers and condensed in the pool
- Feedwater accumulators provide initial feed flow while DHRS transitions to natural circulation flow
- Pool provides a 3 day cooling supply for decay heat removal



ECCS/CHRS

- Provides a means of removing core decay heat and limits containment pressure by:
  - Steam Condensation
  - Convective Heat Transfer
  - Heat Conduction
  - Sump Recirculation
- Reactor Vessel steam is vented through the reactor vent valves (flow limiter)
- Steam condenses on containment
- Condensate collects in lower containment region
- Reactor recirculation valves provide a path for natural circulation through the core



NuScale Power

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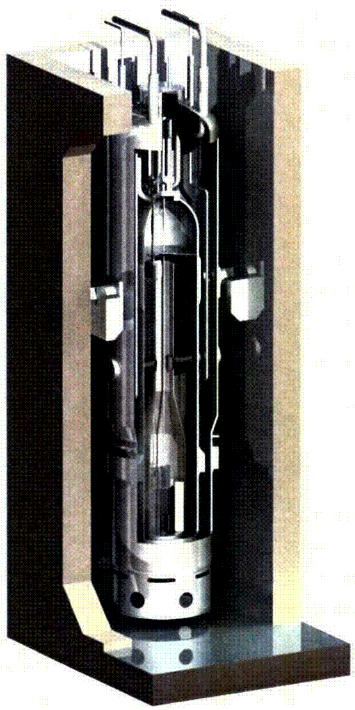
Overview of NuScale Codes and Methods



Dr. José N. Reyes, Jr.
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Outline

- Introduction
- List of Computer Codes
- Verification and Validation
- Sample Integral Assessment Benchmark
- OSU Integral Testing Matrix
- Code Approval Process
- Summary

Introduction

- NuScale will employ a suite of well-established computer codes using industry-standard methods
- Unique features of the NuScale plant need to be considered in selecting and using the codes and methods (e.g., integrated reactor)
- All these codes are conventional codes that have a long history of being used for LWR licensing – we will not be breaking new ground in the codes and methods we plan to use
- These codes will all be brought in house and verified and validated under a NuScale QA system, with the exception of the Studsvik suite
- RELAP5 will be modified using an Appendix K approach

List of Computer Codes

Event	NuScale Code(s) Used
Increase in Secondary Side Heat Removal (Includes Containment Flooding)	N-RELAP5/K, N-COBRA-TF
Decrease in Secondary Side Heat Removal	N-RELAP5/K, N-COBRA-TF
Decrease in Reactor Coolant System Flow Rate	N/A
Reactivity and Power Distribution Anomalies	N-RELAP5/K, SIMULATE-3K, N-COBRA-TF, N-FRAPTRAN
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	SIMULATE-3
Spectrum of Rod Ejection Accidents	N/A
Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory	N/A
CVCS Malfunction that Increases Reactor Coolant Inventory	N-RELAP5/K, N-COBRA-TF
Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	N-RELAP5/K, N-COBRA-TF
Steam Generator Tube Rupture	N-RELAP5/K
Loss-of-Coolant Accidents	N-RELAP5/K
ATWS	N-RELAP5/K, SIMULATE-3K, N-COBRA-TF, N-FRAPTRAN

Verification and Validation

- Adequacy of codes for conducting steady-state core/fuel design and transient and accident analysis should include considerations for:
  - Reduced-height core
  - Integrated system (e.g., internal steam generators and pressurizer)
  - Natural circulation flow conditions
  - Lower operating pressure
  - Lower operating temperature
- A great deal of LWR validation data was generated in reduced-scale facilities at reduced flow, pressure, and temperature conditions
- Validation suite will utilize existing and well establish LWR benchmark data sets, coupled with reliance on the prototype test facility at OSU

OSU Integral Test Facility

- Stainless steel integral system test facility operating at full system pressure and temperature
  - Reactor vessel
  - Electrically heated rod bundle
  - Core shroud with riser
  - Pressurizer
  - Reactor recirculation valves
  - Helical-coil steam generator
  - Variable speed feedwater pump
  - Containment vessel
  - Containment cooling pool
  - Instrumentation

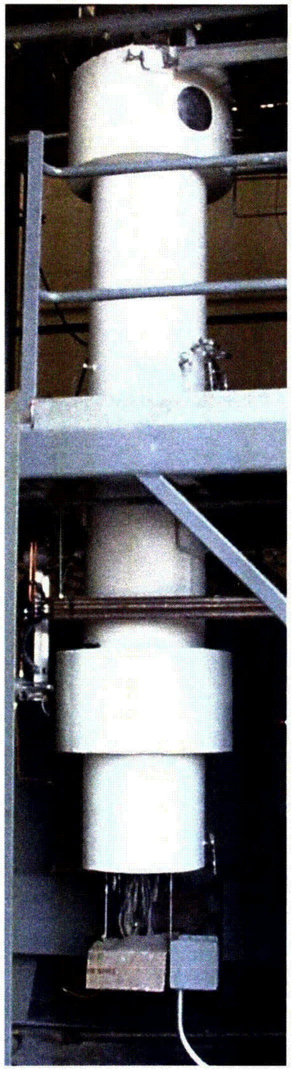
Integral System Test Facility

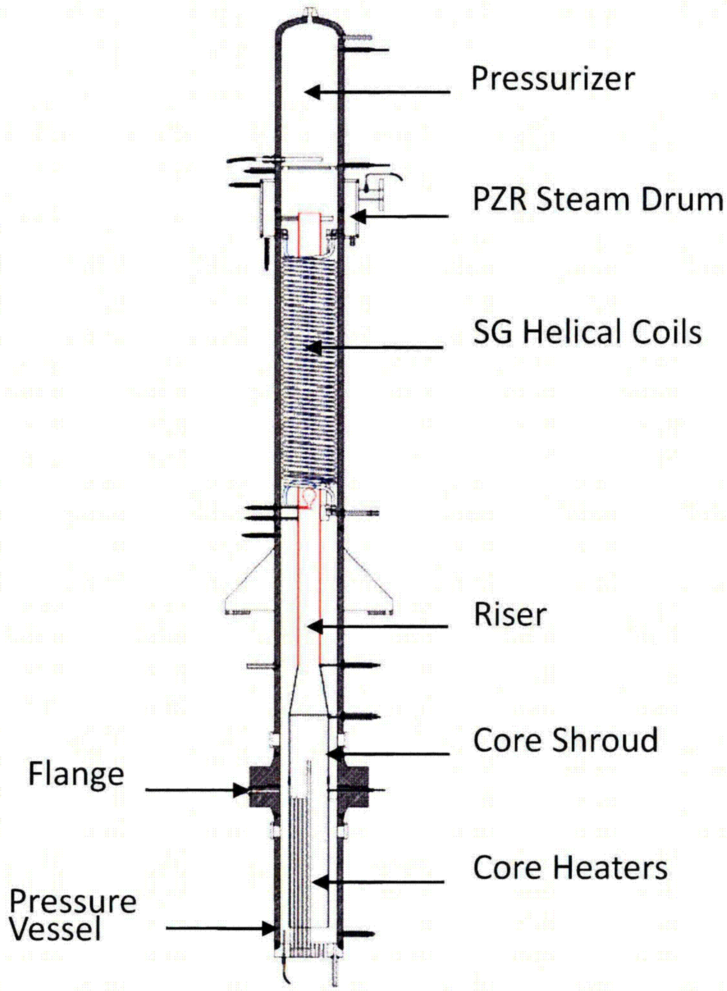
- A scaling analysis was used to guide the design, construction and operation of a 1/3-scale integral system test facility for the original MASLWR design
- NuScale has exclusive use of the test facility
- NuScale is modifying the facility to incorporate design improvements
- Facility can be used to:
  - Evaluate design improvements
  - Conduct integral system tests for NRC certification
- OSU has significant testing capability
  - Performed DOE and NRC certification tests for the AP600 and AP1000 designs
  - 10 CFR 50 Appendix B, NQA-1, 10 CFR 21

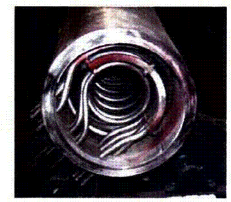
Scaling Ratios

Pressure Ratio	1:1
Temperature Ratio	1:1
Length Ratio	1:3.1
Time Ratio	1:1
Velocity Ratio	1:3.1
Cross-sectional Area Ratio	1:82
Volume Ratio	1:254.7
Power Ratio	1:254.7
Active Heat Transfer Area Ratio	1:254.7
Active Heat Transfer Wall Thickness Ratio	1:1

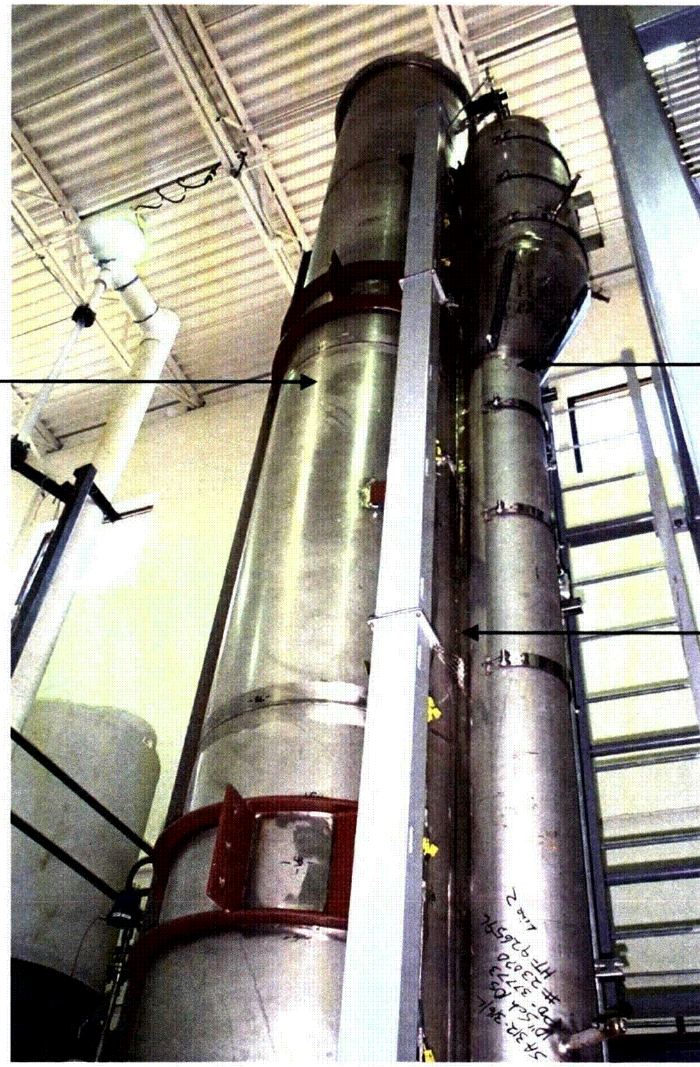
Integrated Reactor Test Vessel







Containment and Cooling Pool

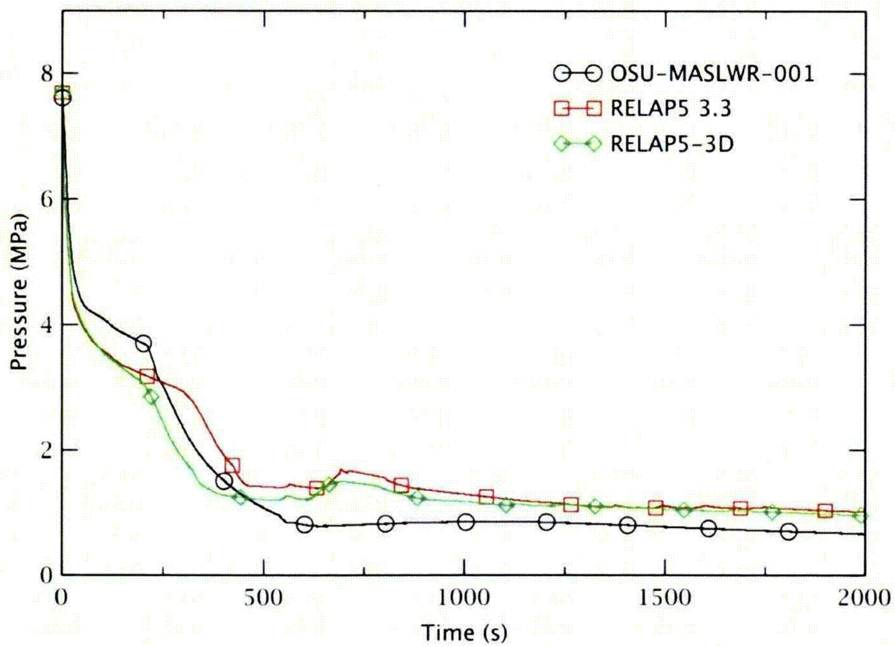


Sample Integral Assessment Benchmark

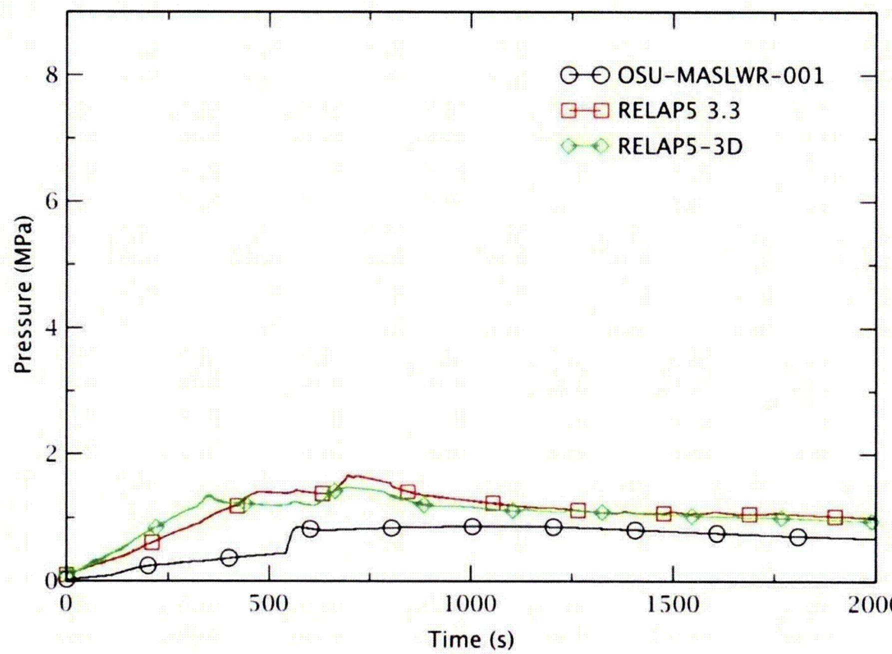
- Two codes were used in the comparisons: RELAP5-3D and RELAP5/MOD3.3. The codes were applied essentially “out-of-the-box”.
- Same input model was used for both codes that modeled all major facility components
- OSU-MASLWR-001 Integral Test
  - Inadvertent actuation of one ADS valve
  - Followed by staged blowdown of the facility

Sample Integral Assessment Benchmark: OSU-MASLWR-001

Primary system pressure

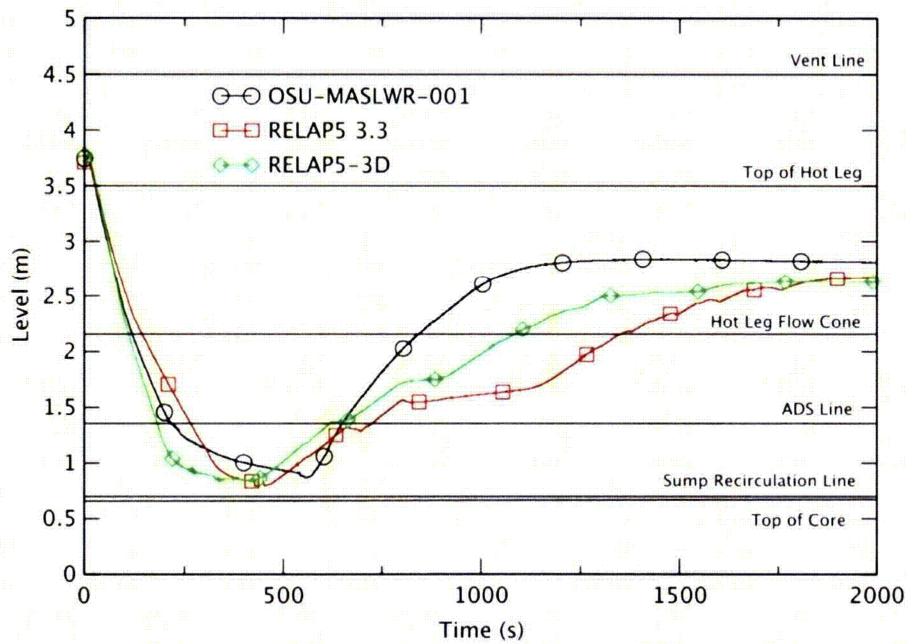


Containment pressure

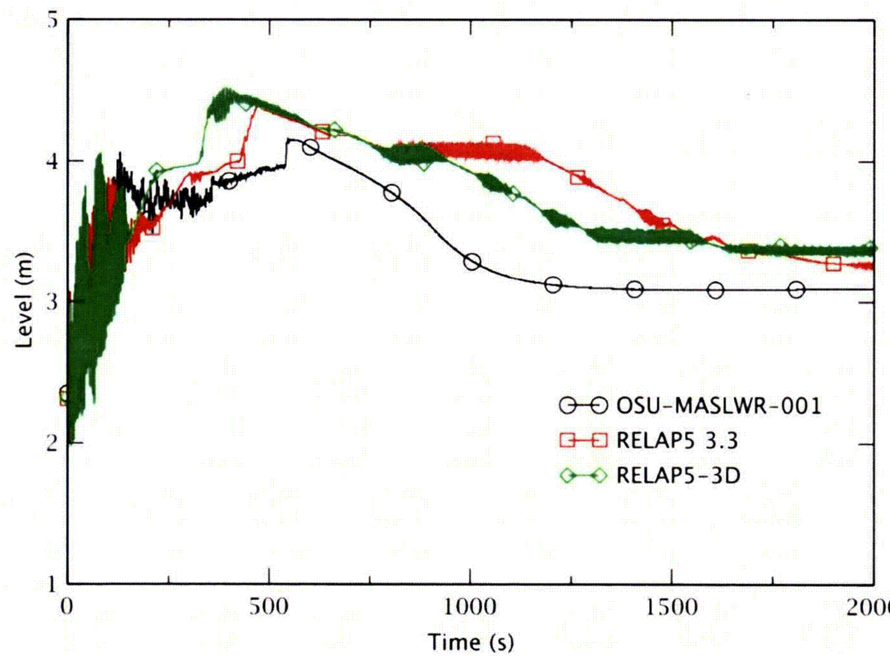


Sample Integral Assessment Benchmark: OSU-MASLWR-001

Liquid level above the core



Containment water level



OSU Integral Test Matrix to Support Design Certification

NuScale Power

- The following tests are planned (2009-2010):
  - Increase in feedwater flow rate
  - Loss of feedwater flow/feed line break
  - Decrease in feedwater temperature
  - Main steam line break
  - Station blackout
  - 5 LOCAs

Code Approval Process

- Developmental assessment plans for four in-house codes are being prepared: N-RELAP5/K, N-COBRA-TF, N-FRAPCON-3, and N-FRAPTRAN
- Limited model development for N-RELAP5/K is expected for predicting helical-coil steam generator heat transfer
- No development work is expected for the other three codes
- Some validation will be carried out in-house for the 3-D simulator codes, SIMULATE-3 and SIMULATE-3K, while code verification will be done by Studsvik under their NQA-1 Quality Assurance Program
- Licensing Topical Reports on the verification and validation of in-house codes will be submitted during the design certification review

Summary

- Suite of codes selected for based on decades of LWR experience
- Four codes will be brought in-house under NuScale's QA Program
- Codes are well-established and have been reviewed and approved by the NRC for analyzing typical PWRs
- Verification and Validation
    - NuScale design has several features that will need to be evaluated for code applicability
    - Many code models are based on data from reduced-scale facilities with very similar conditions as the full-scale NuScale design
    - Minor model development expected for only one code: N-RELAP5/K. *Preliminary integral assessments show current RELAP5 family of codes predict important data trends.*
    - Integral system performance will be confirmed by running new tests at OSU

Topics for Proprietary Session

- Core/Fuel Design Methods
- Non-LOCA Analysis
- LOCA Analysis
- Containment Performance Analysis
- Sample Calculations

NuScale Power