Reactor and Fuel Design Update

Larry Linik, Kent Welter

Sangyoun Jeon, Jongsung Yoo

February 4, 2014

NuScale Nonproprietary

NuScale Power™

PM-0114-5697-NP

Agenda

- Purpose
- Plant overview
- Fuel solution strategy
  - core design update
  - fuel design status (KNF)
  - fuel design limits – DSRS 4.2
- Nuclear Design – DSRS section 4.3
- Thermal Hydraulic Design – DSRS section 4.4
- Control Rod Drive System – DSRS 4.6
- Chapter 4 – information for NRC development of NuScale DSRS
- Results achieved and next steps
- Requested Feedback

Purpose

- Familiarize NRC with NuScale's open market fuel solution
- Establish fuel design scope to support DC and commercial fuel design for COL
- Establish scope of CRA conceptual design to support DCA
- Establish approach for evaluating flow instabilities
- Provide information for NRC development of NuScale DSRS for Chapter 4

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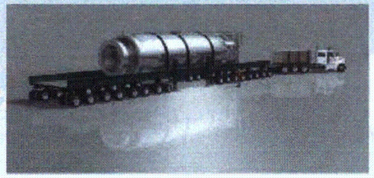
Plant Overview

Plant Design Overview

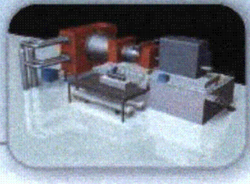
Factory Manufacturing

Module includes Containment and Reactor Vessel

Shipped by Truck, Rail, or Barge

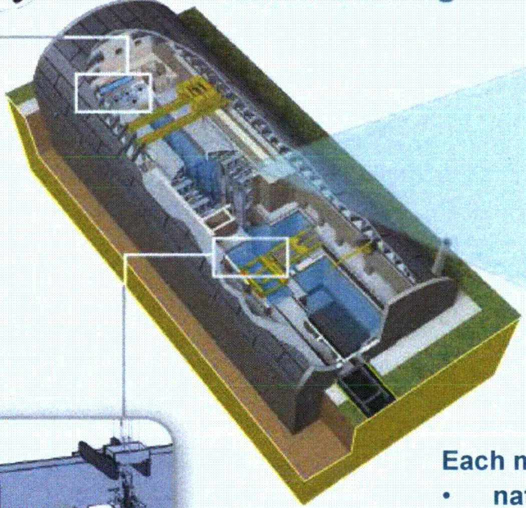


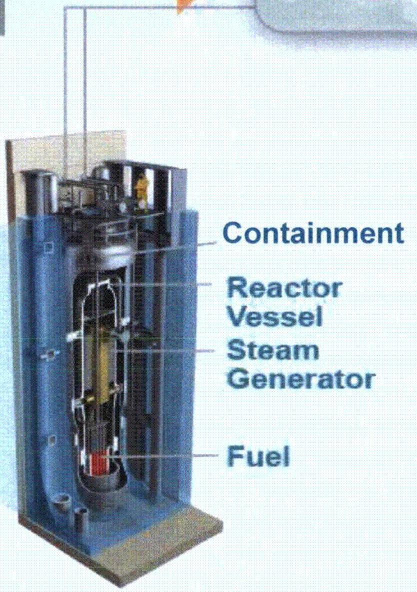
Skid-Mounted Steam Turbine/Generator

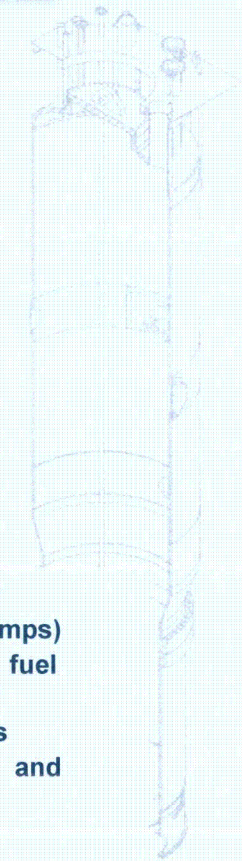


Below-Ground Control Room provides enhanced security and state-of-the-art controls

12 Module Reactor Building

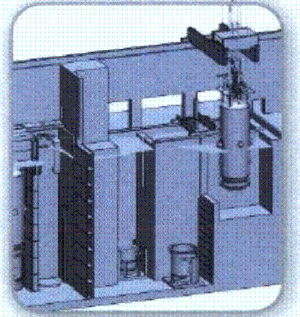






Each Module is refueled underwater while the remainder of the plant produces power

- Refueled once every 24 months
- 10 Day Refueling Target



Each module installed in its own isolated bay

- natural circulation (no reactor coolant pumps)
- 37 standard 17x17 PWR fuel (half-height) fuel assemblies
- standard magnetic jack control rod drives
- internal helical coil steam generators and pressurizer
- 45 MWe net power

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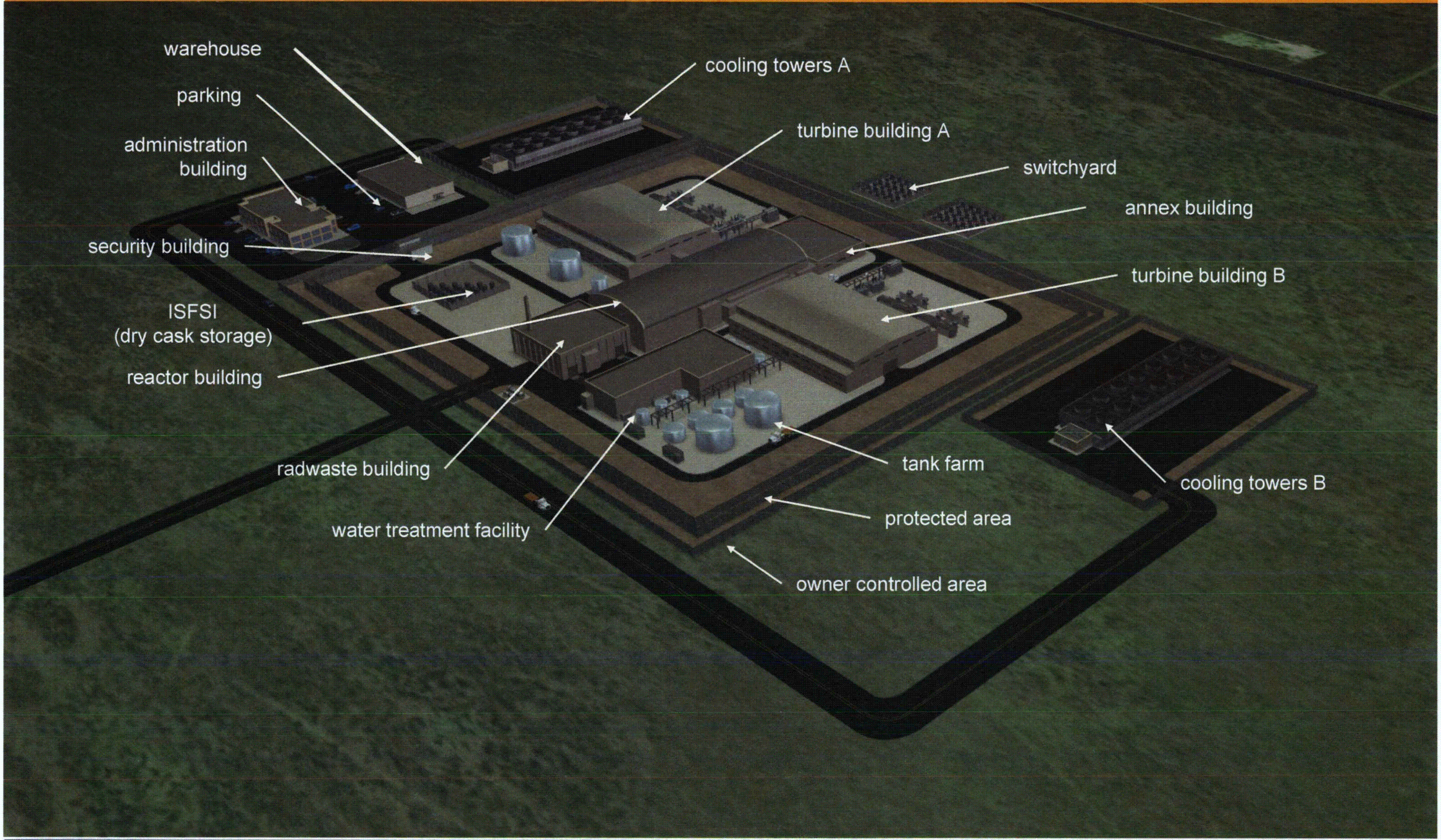
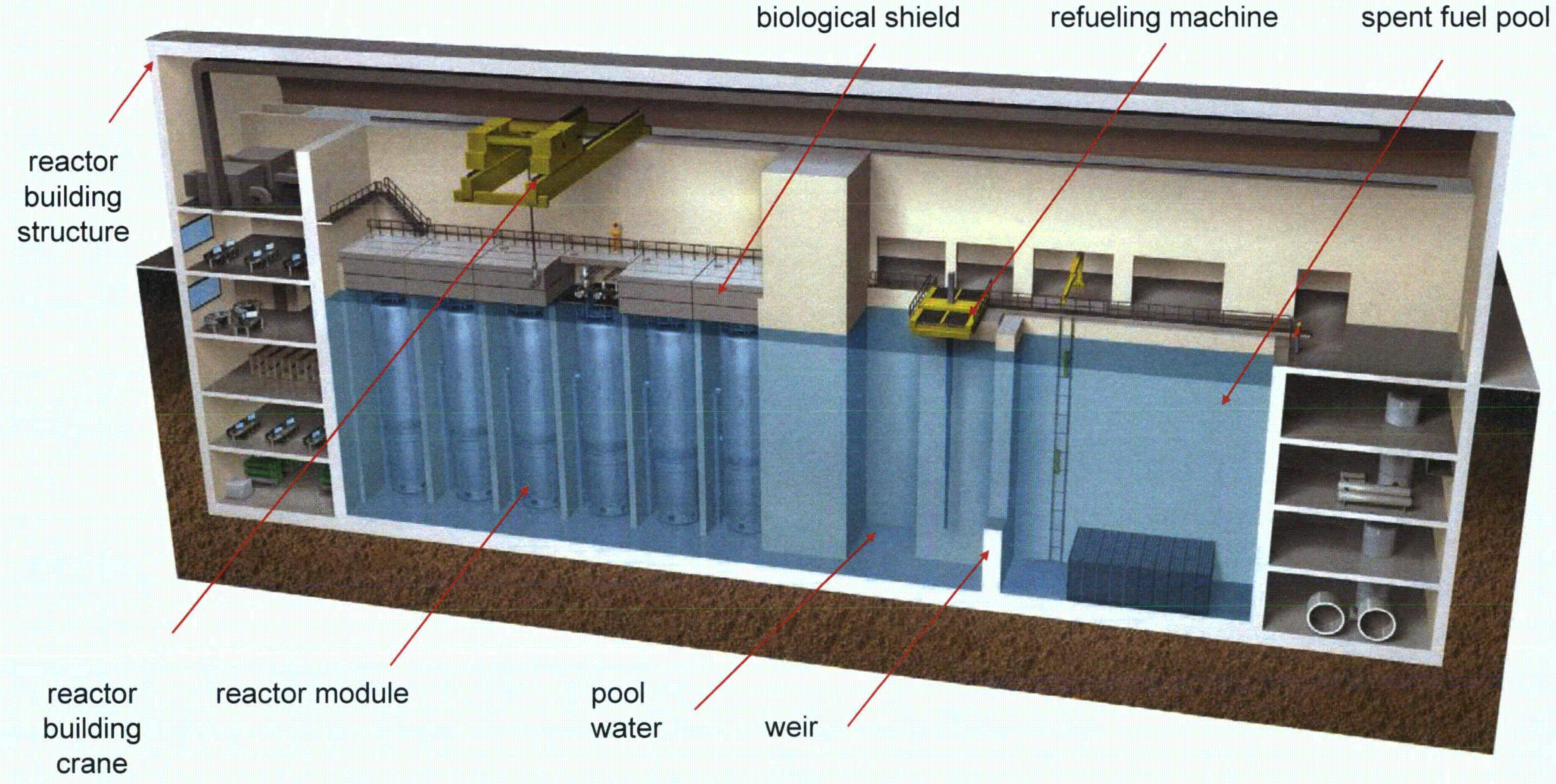
Site Aerial View



Reactor Building Cross-Section

Reactor building houses reactor modules, spent fuel pool, and reactor pool



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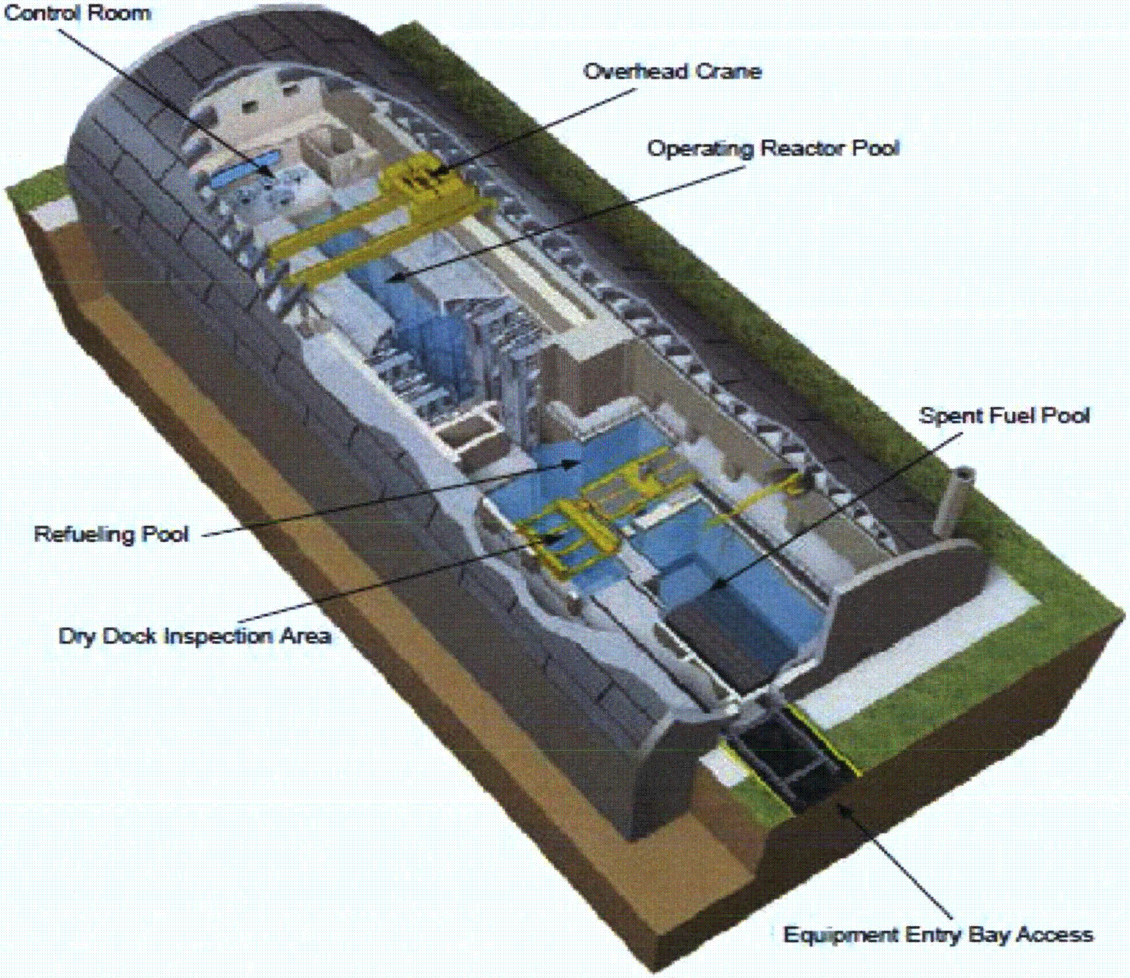
Reactor Building Overhead View



Basic Plant Parameters

Overall Plant	
• Net electrical output	Up to 540 MW(e)
• Plant thermal efficiency	> 30%
• Number of power generation units	Up to 12
• Nominal plant capacity factor	> 95%
• Total plant area	~44 acres
Power Generation Unit	
• Number of reactors	One
• Net electrical output	45 MWe
• Steam generator number	Two independent tube bundles (50% capacity each)
• Steam generator type	Vertical helical coil tube (secondary coolant boils inside tube)
• Steam cycle	Superheated
• Turbine throttle conditions	3.3 MPa (475 psia)
• Steam flow	67.5 kg/s (536,200 lb/hr)
• Feedwater temperature	149° C (300 °F)
Reactor Core	
• Thermal power rating	160 MWt
• Operating pressure	12.7 MPa (1850 psia)
▪ Fuel design	UO_2 (< 4.95% U^{235} enrichment); 37 half height 17x17 geometry lattice fuel assemblies; Zircaloy-4 or advanced cladding material; negative reactivity coefficients
▪ Refueling interval	24 months

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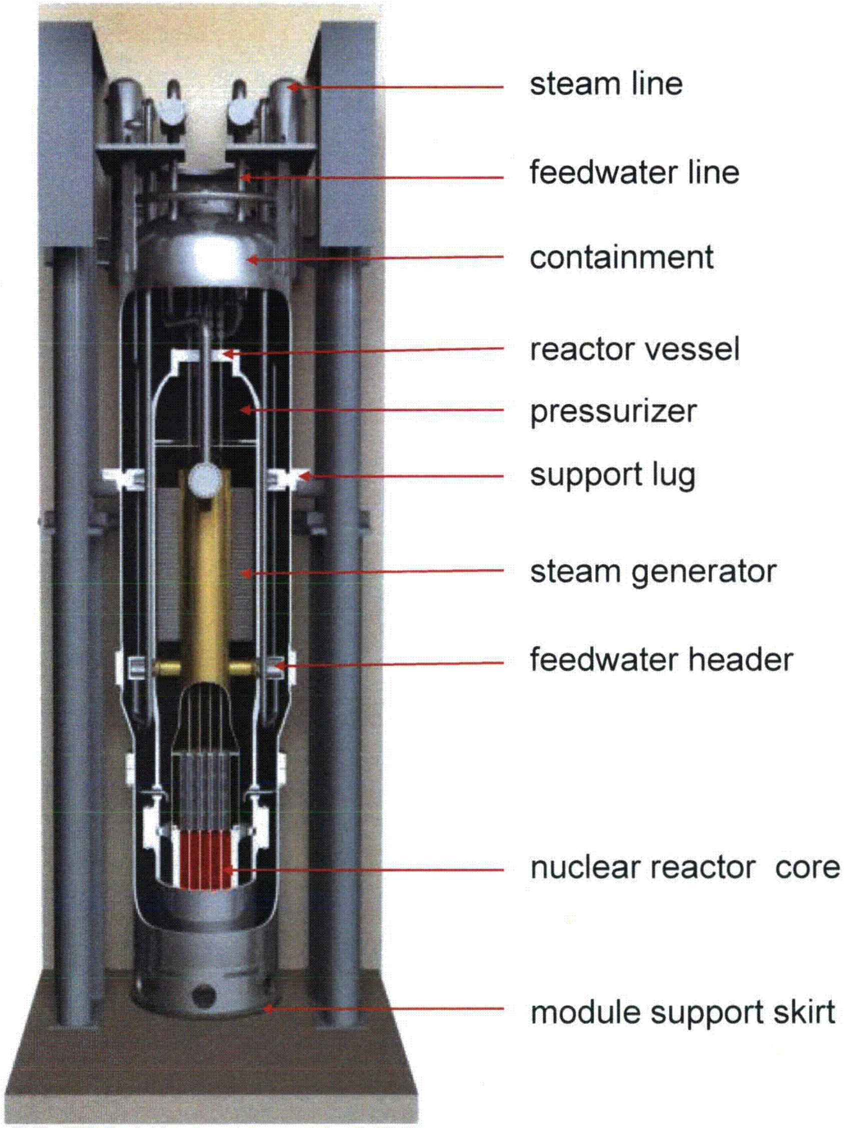
Reactor Module Overview

Natural convection for cooling

- passively safe, driven by gravity, natural circulation of water over the fuel
- no pumps, no need for emergency generators

Simple and small

- reactor is 1/20th the size of large reactors
- integrated reactor design, no large-break loss-of-coolant accidents



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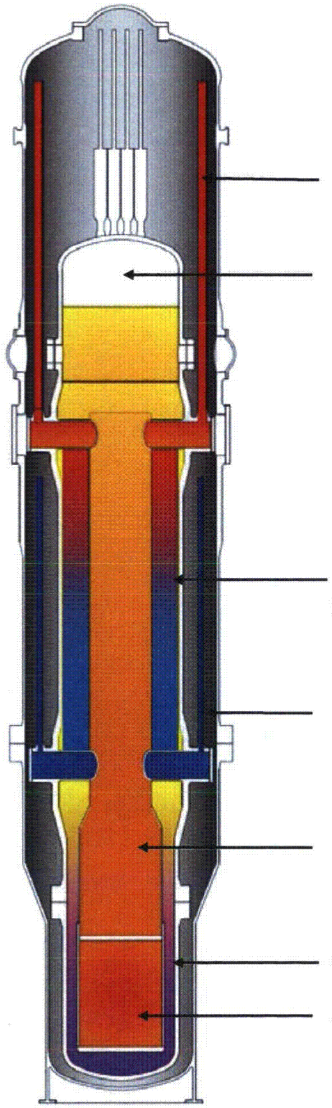
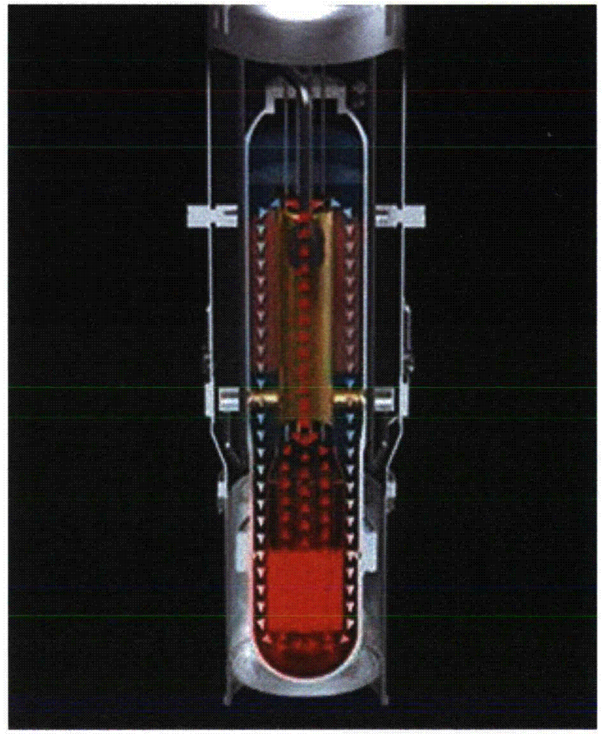
Module Normal Operation

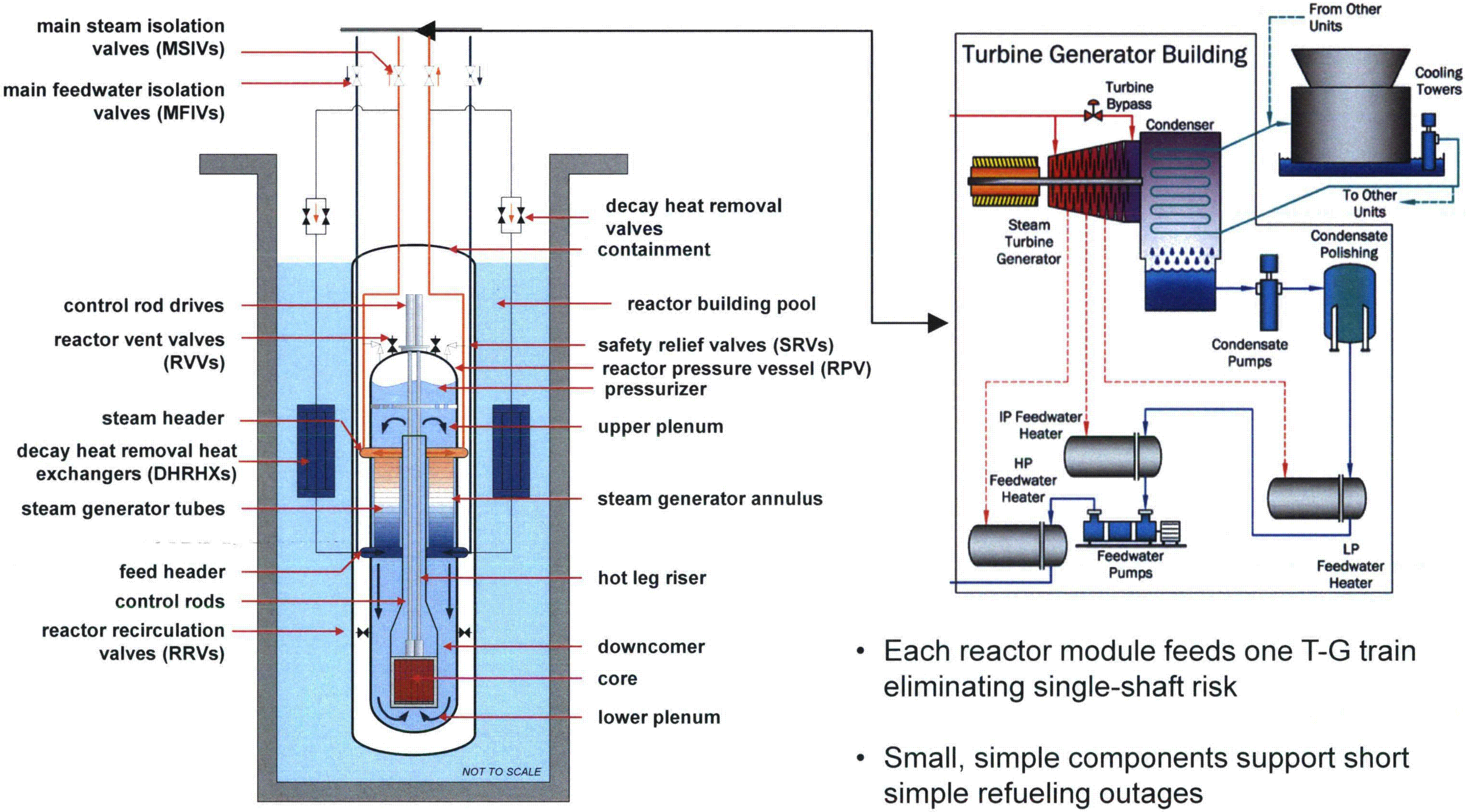
- Primary side
  - natural circulation
  - integral pressurizer
- Secondary side
  - feedwater plenums
  - two helical steam generators
  - steam plenums





primary coolant flow path

NuScale Power Train



- Each reactor module feeds one T-G train eliminating single-shaft risk
- Small, simple components support short simple refueling outages

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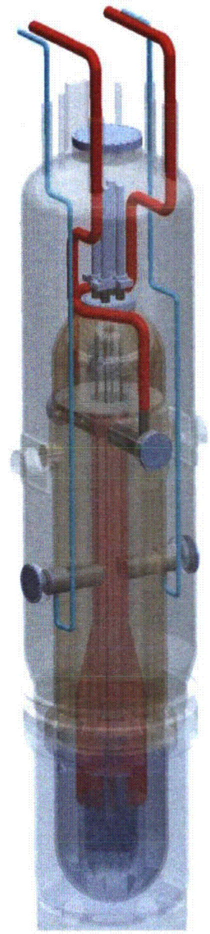
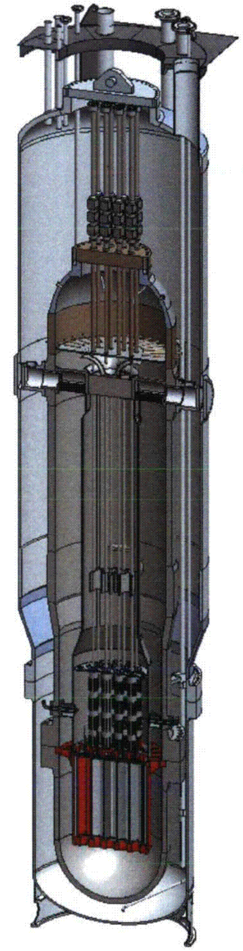
Containment Design

High Pressure Containment – Enhanced Safety

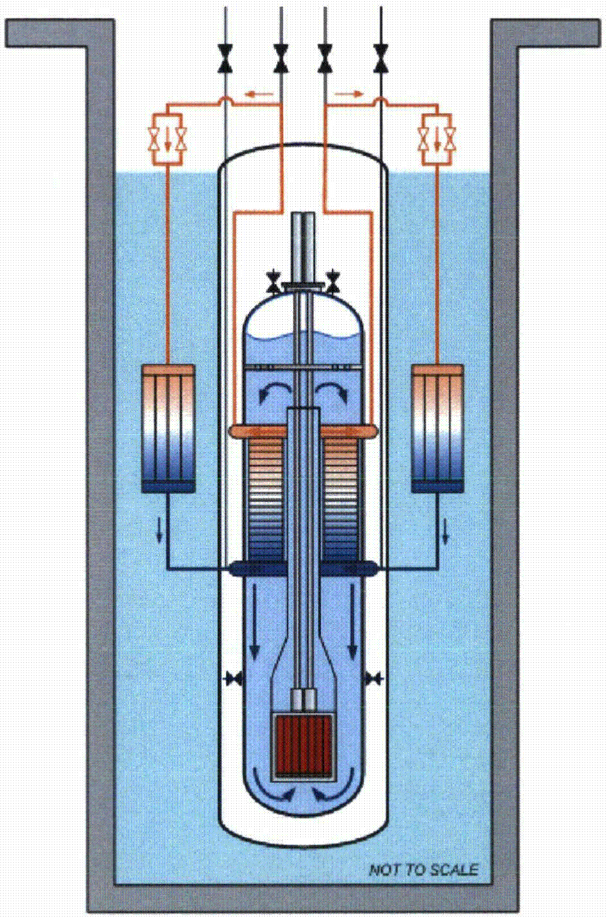
- Containment volume sized so that core does not uncover following a LOCA (prevents fuel heat-up)
- Large water pool keeps containment shell cool and promotes efficient post-LOCA steam condensation
- Insulating vacuum
  - significantly reduces conduction and convection heat transfer during normal operation
  - requires no insulation on reactor vessel. Eliminates sump screen blockage issue (GSI-191)
  - improves LOCA steam condensation rates by eliminating air
  - prevents combustible hydrogen mixture in the unlikely event of a severe accident (i.e., little or no oxygen)
  - reduces corrosion and humidity problems inside containment





Passive Decay Heat Removal System

- Main steam and main feedwater isolated
- Decay heat removal (DHR) valves opened
- Decay heat passively removed via the steam generators and DHR heat exchangers to the reactor pool
- DHR system is composed of:
  - four actuation valves (1 of 4 needed)
  - two heat exchangers (1 of 2 needed)
  - two independent single failure proof trains (1 of 2 trains needed)



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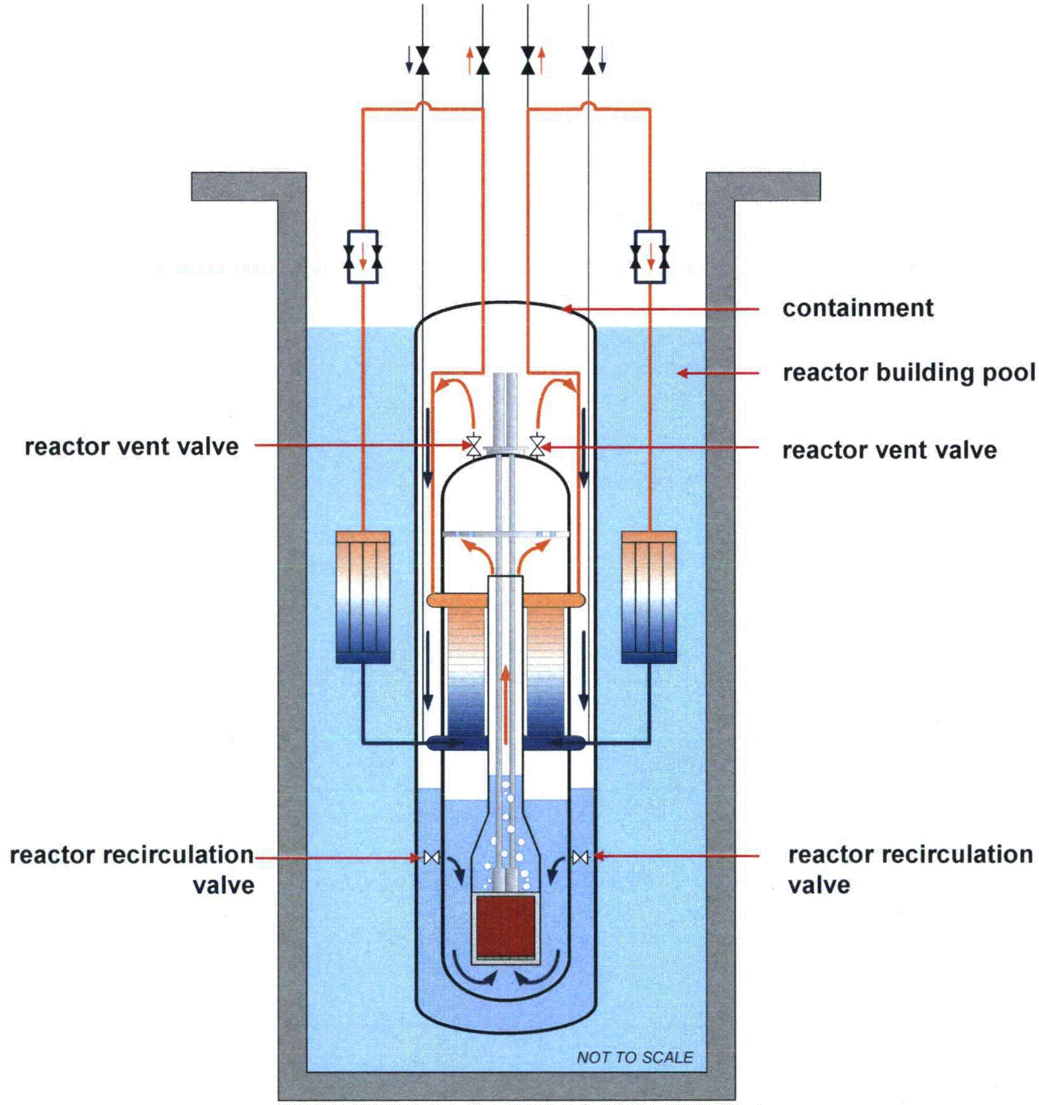
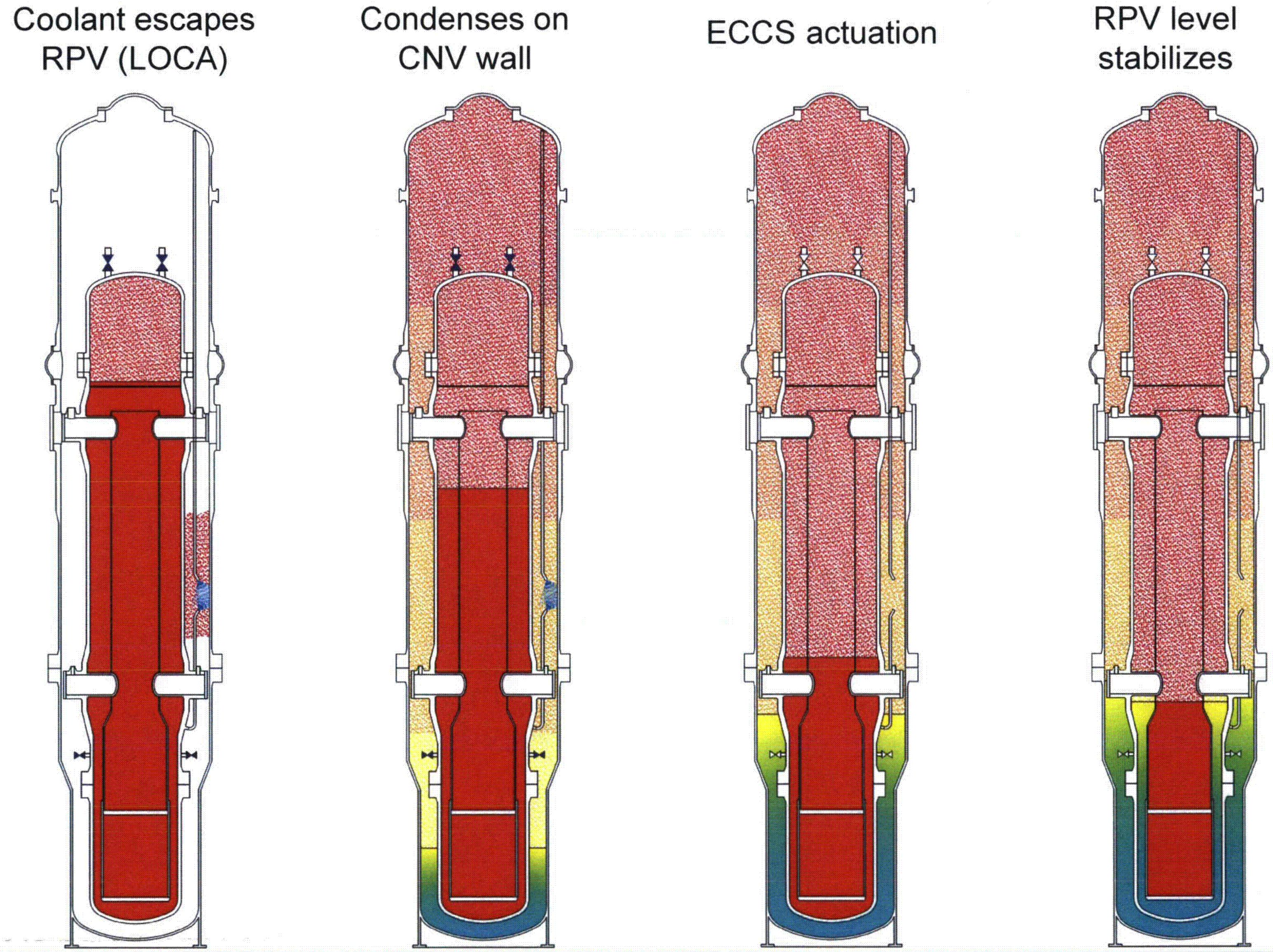
Emergency Core Cooling System (ECCS) and Containment Heat Removal System (CHRS)

- Design does not require safety injection. Reactor water inventory is protected by containment isolation.
- Reactor vent valves opened on safety signal
- When containment liquid level is high enough, reactor recirculation valves open
- Decay heat removed
  - condensing steam on inside surface of containment vessel
  - convection and conduction through liquid and both vessel walls



ECCS Accident Operation



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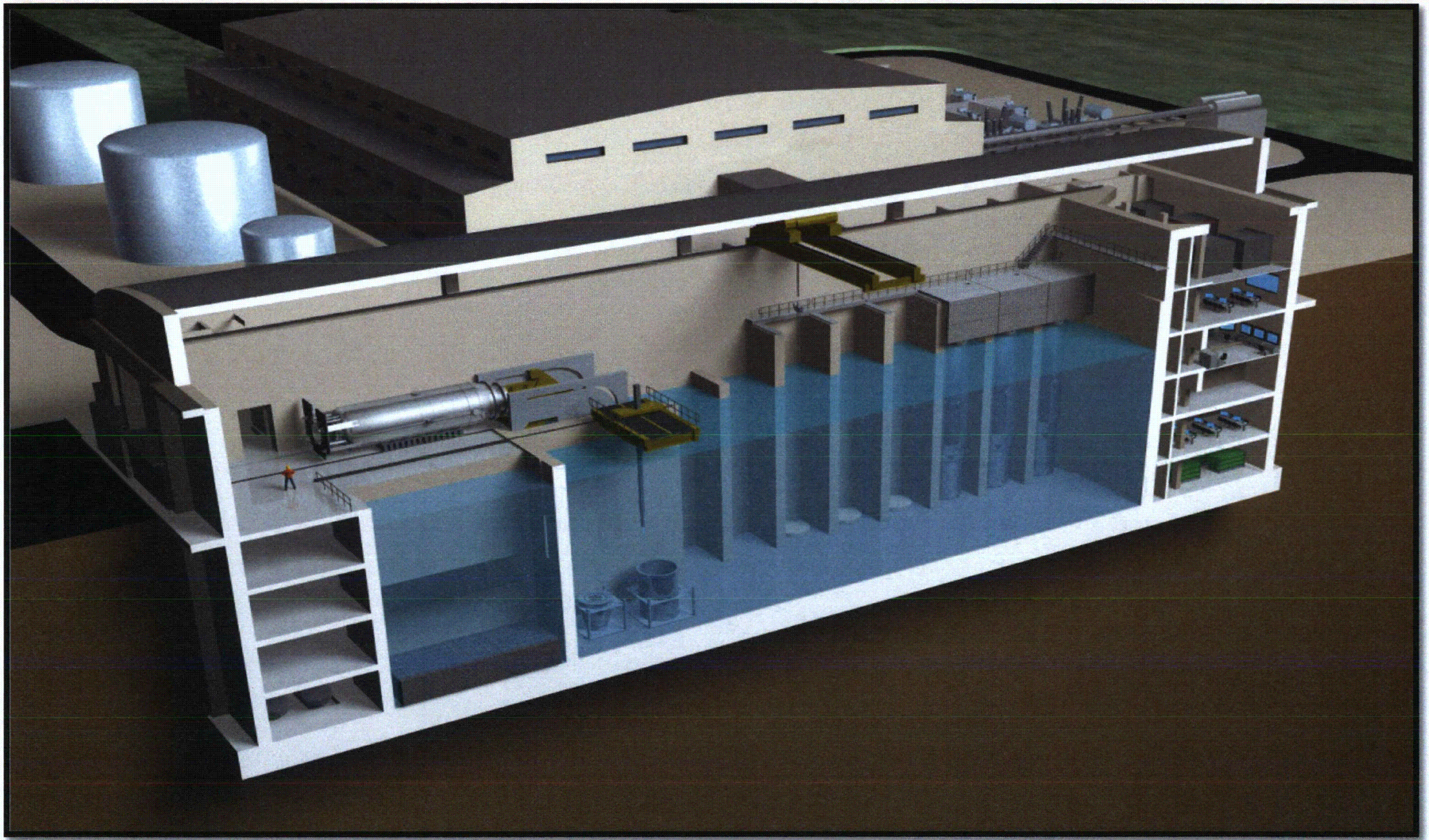
Module Component Assembly



Design Simplification

- **New system**
  - containment evacuation
- **Eliminated systems**
  - containment spray
  - containment fan cooler
  - auxiliary feedwater
  - ECCS injection and recirculation
  - steam generator blowdown
  - main plant electrical generator hydrogen supply

- **Eliminated components**
  - reactor coolant pumps
  - ECCS pumps, tanks, and RPV injection lines
  - containment sumps and tanks
  - refueling water storage tank
  - reactor coolant hot leg and cold leg piping
  - pressurizer surge line and relief tank
  - reactor vessel and primary coolant system insulation
  - safety-related emergency diesel generators

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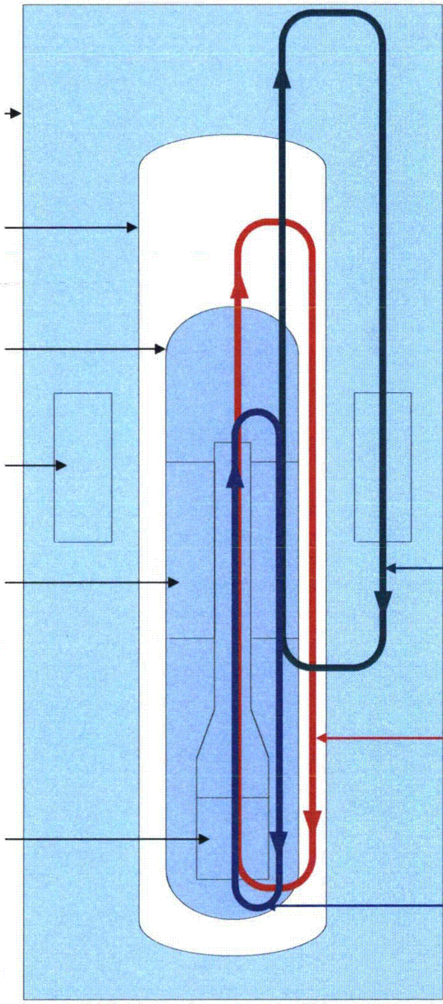
Passive Cooling Systems



- BC: boiling and condensation
- NC: natural circulation

- **Boiling and condensation in the decay heat removal system (DHRS)**
- **Boiling and condensation in the emergency core cooling system (ECCS)**
- **Natural circulation in the reactor coolant system (RCS)**

DHRS BC* **transient operation**

ECCS BC* **accident operation**

RCS NC* **normal operation**

Fuel Solution Strategy

Larry Linik
Nuclear Engineering Supervisor

Open Market Fuel Solution

- Generic Zr-4 design sufficient for design certification of base fuel performance capability
- Competent vendor (KNF, AREVA, Westinghouse, etc.) can supply fuel for the COLA
  - chosen by customer, supported by NuScale
  - meet or exceed the performance of the Zr-4 generic design

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Fuel Licensing Path to Operation

- Design limits defined as Tier 2* in previous applications
- Licensing options for commercial fuel design, from example given in DC/COL-ISG-011
  - Recommend departure from DC fuel design to COL application, or
  - process amendment request and FSAR updates after COL is issued.
- Approach similar to fuel vendor change process for the current fleet
- Submit topical reports separate from DCA for commercial fuel design

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Fuel Design Schedule – Being Revised

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Fuel Design Status

- Accomplishments since last meeting

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Core Design Update

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NuScale Core

- 37 typical 17x17 half-height fuel assemblies

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Reflector Design Change

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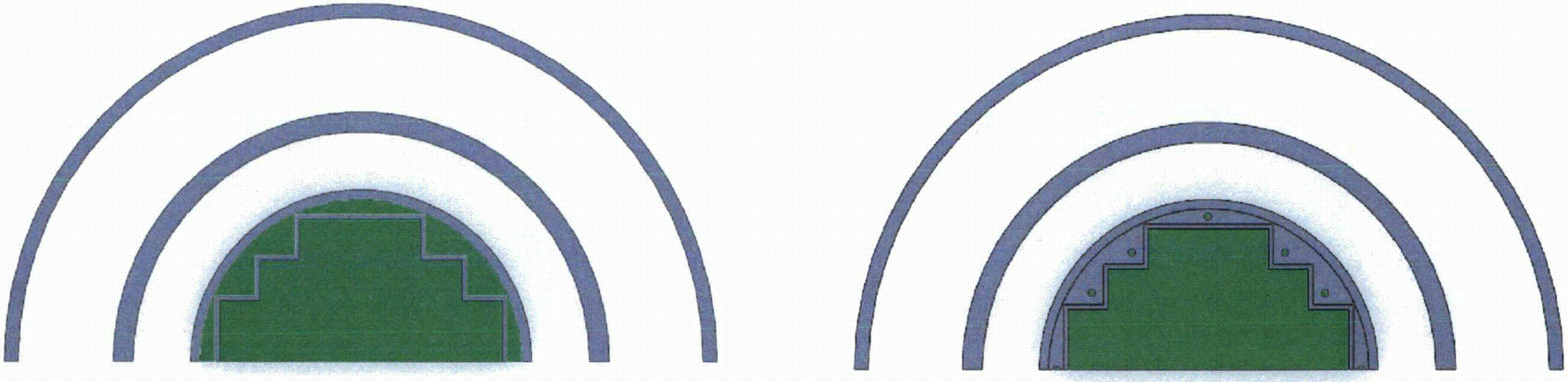
- Motivated by a desire to reduce neutron leakage
  - fuel utilization is increased
  - lower fuel enrichment is required
  - power distribution is flattened and core characteristics are improved
  - neutron dose to vessels is reduced



Heavy Reflector

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Fuel Design Status
KEPCO Nuclear Fuel (KNF)

Dr. Sangyoun Jeon

General Manager, Nuclear Fuel Development Section

Dr. Jongsung Yoo

General Manager, Fuel Rod Design Technology Section

Overview of NuScale Fuel Development

February 4, 2014

KEPCO Nuclear Fuel

Contents

1 Introduction

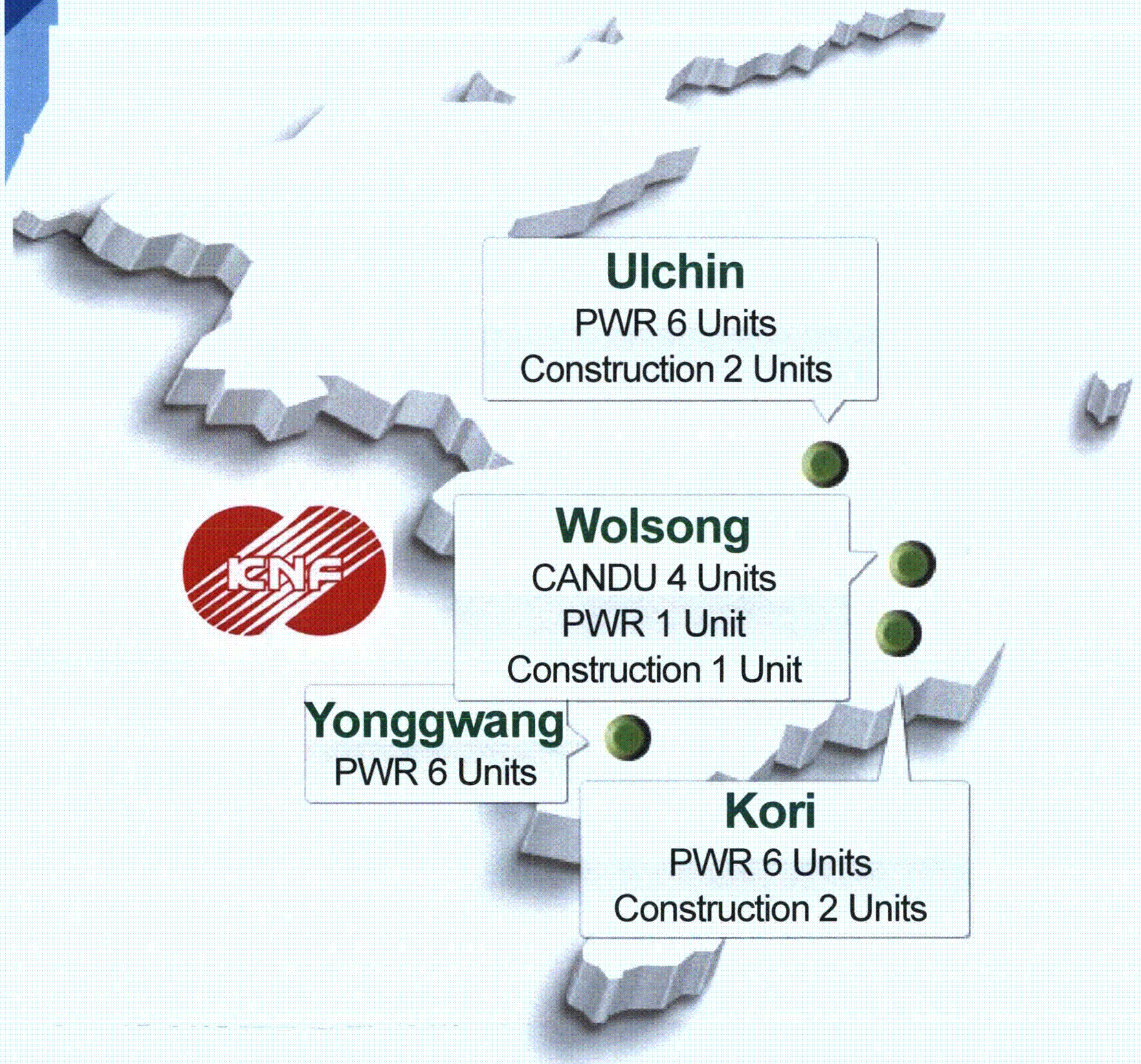
1. Introduction

Nuclear Fuel Assembly Development for NuScale SMR

- Develop a Zircaloy-4 clad based fuel design with all necessary analyses and tests
- Comply with acceptance criteria embodied in the latest revision of the U.S. Nuclear Regulatory Commission Standard Review Plan (NUREG-0800)

2 KNF Fuel Development History

2. KNF Fuel Development History

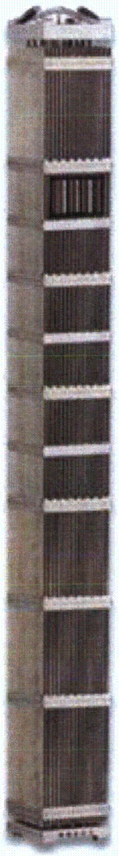


Number of Units

Type	In Operation	Under Cons.	Planning (~2030)
OPR1000	11	1	-
APR1400	0	4	7~10
W 14x14	1	-	-
W 16x16	1	-	-
W 17x17	6	-	-
CANDU	4	-	-
Total	23	5	7~10

2. KNF Fuel Development History

PWR

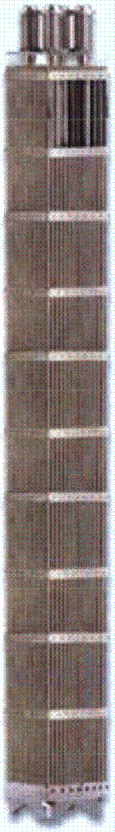


Westinghouse Type Plants

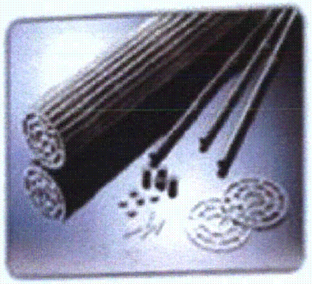
- 14X14 OFA
- 16X16 STD
- 17X17 RFA
- 16X16 ACE7™
- 17X17 ACE7™

OPR 1000 Plants
APR 1400 Plants

- 16X16 Guardian
- 16X16 PLUS7™

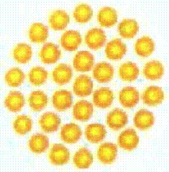


CANDU



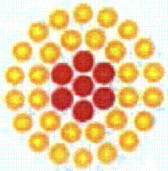
CANDU fuel

- 37- element standard fuel



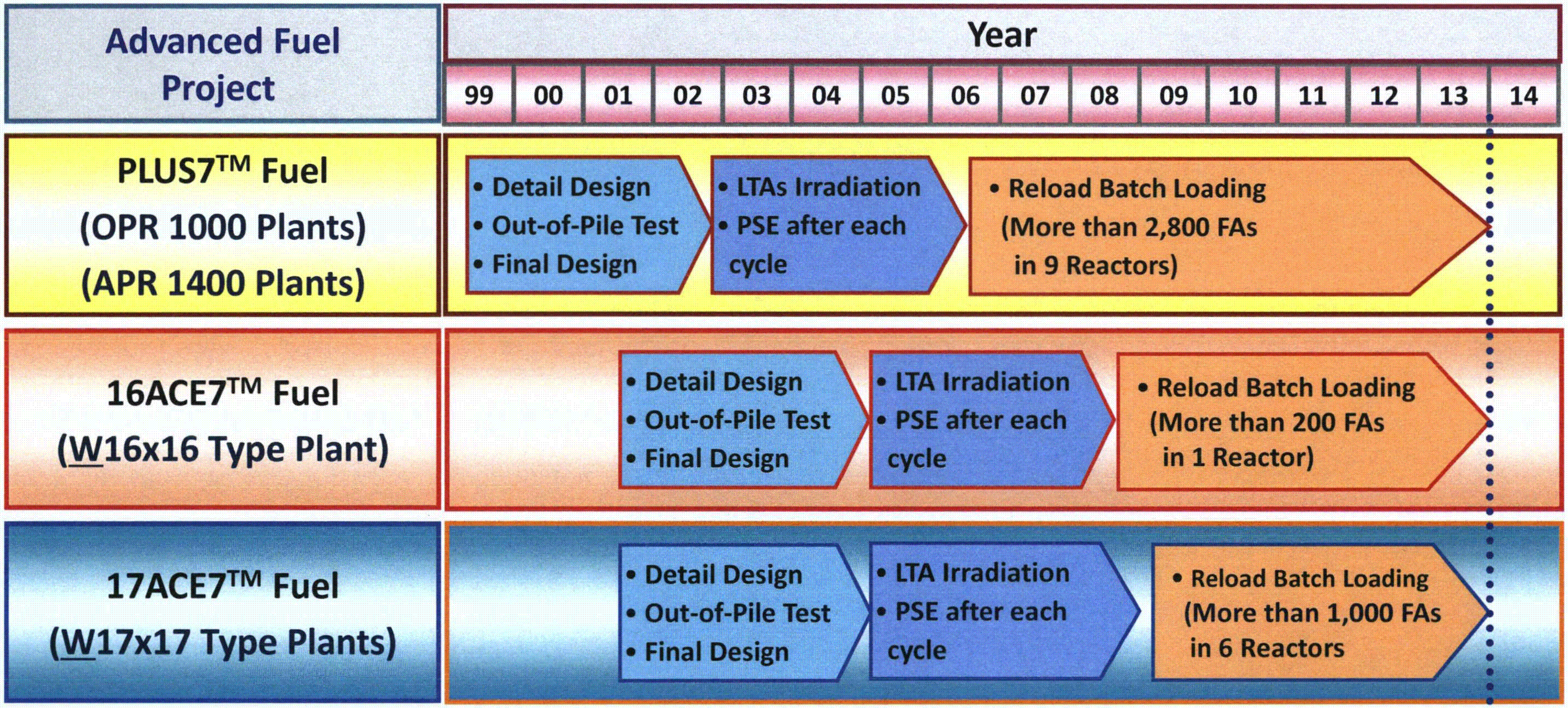
CANFLEX-NU

- 43 - element fuel



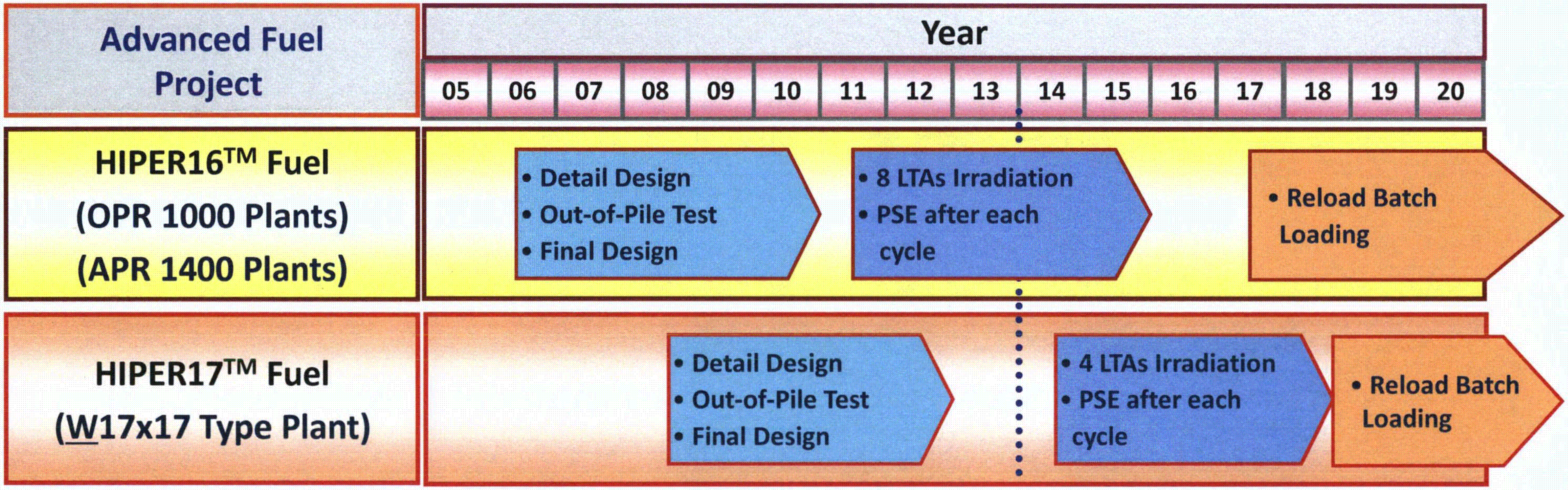
2. KNF Fuel Development History

- KNF Advanced Fuel Development



2. KNF Fuel Development History

- KNF Indigenous Fuel Development

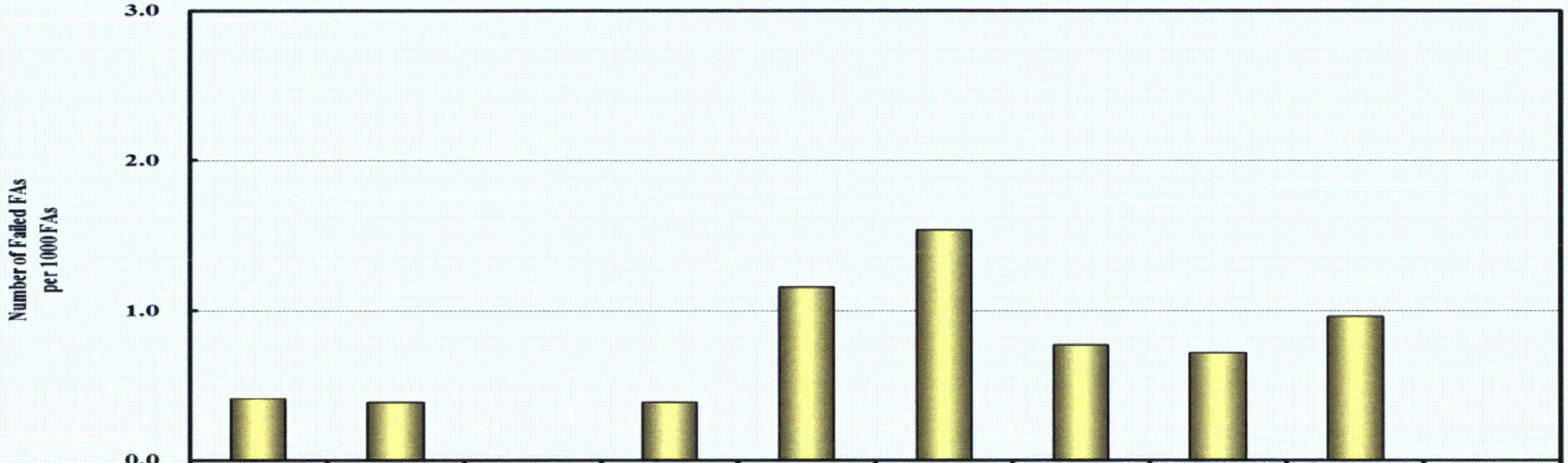


PLUS7™ Fuel will be replaced with HIPER16™ Fuel from 2017

17ACE7™ Fuel will be replaced with HIPER17™ Fuel from 2018

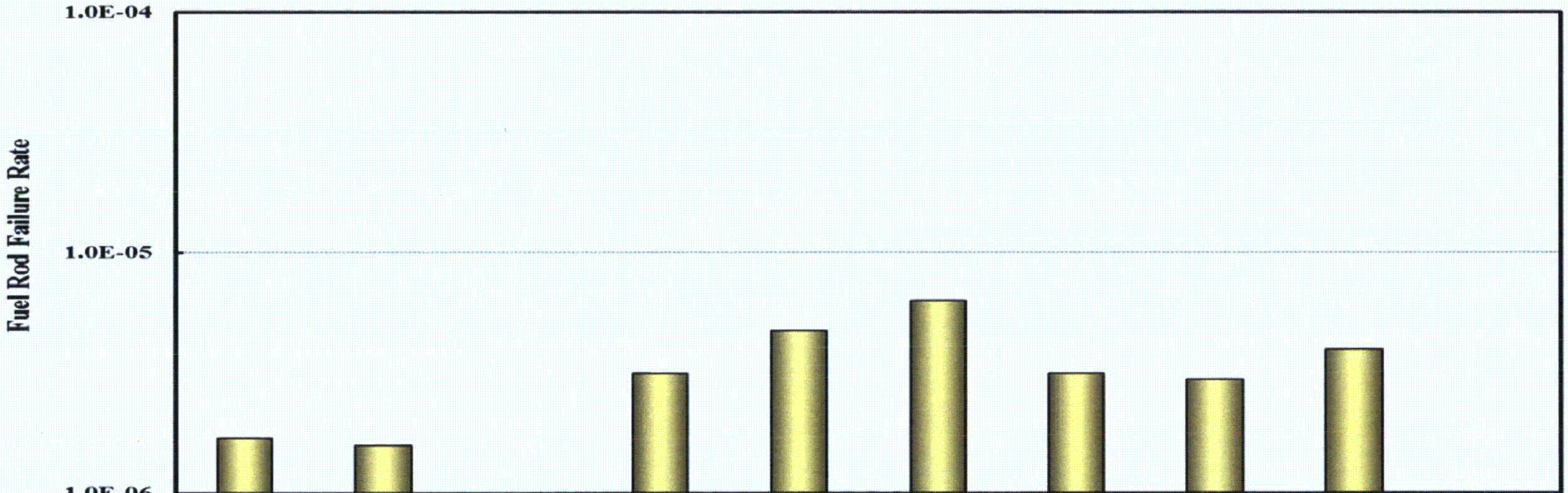
2. KNF Fuel Development History

- Fuel Failure Rate



	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013
Total	0.41	0.38	0.00	0.38	1.15	1.54	0.77	0.72	0.96	0.00

Fuel Assembly Failure Rate



	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013
Total	1.69E-06	1.58E-06	0	3.16E-06	4.74E-06	6.32E-06	3.16E-06	2.96E-06	3.96E-06	0

Fuel Rod Failure Rate

3 NuScale Fuel Design Characteristics

3. NuScale Fuel Design Characteristics

- Design Development
  - Commercial 17x17 Type FA Features and Dimensions
  - Component Materials

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  - No Debris Filtering Features
  - Reliability, Compatibility, Manufacturability

3. NuScale Fuel Design Characteristics

- NuScale Fuel – Design Characteristics

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3. NuScale Fuel Design Characteristics

- NuScale Fuel – Main Parameters

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3. NuScale Fuel Design Characteristics

- NuScale Fuel – Skeleton

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3. NuScale Fuel Design Characteristics

- NuScale Fuel - Top Nozzle

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3. NuScale Fuel Design Characteristics

- NuScale Fuel - Bottom Nozzle

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3. NuScale Fuel Design Characteristics

- NuScale Fuel - Inconel Grid

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3. NuScale Fuel Design Characteristics

- NuScale Fuel – Zircaloy Grid

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3. NuScale Fuel Design Characteristics

- NuScale Fuel – Fuel Rod

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3. NuScale Fuel Design Characteristics

- Summary of Design Characteristics

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4 Design Evaluation Results

4. Design Evaluation Results

- Preliminary drawings
  - preliminary drawings have been issued (for NuScale fuel assembly and its components)
- Detail design evaluation
  - fuel assembly structural evaluation
  - fuel assembly growth evaluation
  - fuel assembly lift-off evaluation
  - fuel assembly pressure drop evaluation
  - spacer grid growth evaluation
  - debris filtering evaluation
  - fuel assembly seismic analysis

4. Design Evaluation Results

- Fuel assembly structural evaluation
  - code and standard
    - 10 CFR 50 Appendix A. GDC 10 Reactor Design
    - NUREG-0800, SRP 4.2 Fuel System Design
    - ASME Code Section Ⅲ, Year 2004

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4. Design Evaluation Results

- Fuel assembly growth evaluation

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4. Design Evaluation Results

- Spacer grid growth evaluation

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4. Design Evaluation Results

- Fuel Assembly Lift-off Evaluation

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4. Design Evaluation Results

- Fuel assembly pressure drop evaluation

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4. Design Evaluation Results

- Debris filtering evaluation

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4. Design Evaluation Results

- Scoping seismic analysis - CPM generation

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4. Design Evaluation Results

- Scoping seismic analysis - FA models

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4. Design Evaluation Results

- Scoping seismic analysis – results

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5 Design Verification Plan

5. Design Verification Plan

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5. Design Verification Plan

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5. Design Verification Plan

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5. Design Verification Plan

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Thank you for your attention!

– Q & A –

Overview of ROPER Code Verification and Validation Plan

February 4, 2014

KEPCO Nuclear Fuel

Contents

1 Introduction

1. Introduction

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2 US NRC SQA Requirements

2. US NRC SQA Requirements

- Comparison of Software Quality Assurance (SQA) requirements between US NRC and NuScale/KNF

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2. US NRC SQA Requirements

- Software Quality Assurance (SQA) of US NRC is defined as (NUREG-1737 "SQA Procedures for NRC T-H Codes", Dec. 2000):
  - Planned and systematic actions necessary to provide adequate confidence that a software product meets established technical requirements

2. US NRC SQA Requirements

- SQA activities can be categorized as follows (NUREG-1737):
  1. Documentation of software or software modules as they are developed
  2. Software verification and validation activities
  3. Non-conformance reporting and corrective actions
  4. Acceptance testing and installation of software
  5. Configuration management
  6. Quality assessment and improvement

3 KNF SQA Requirements

3. KNF SQA Requirements

- KNF SQA requirements covers:

  1. Classification of software
  2. Software development or change
  3. Software error reporting and corrective actions
  4. Software maintenance and discarding
  5. Configuration management
  6. In-use tests
  7. Documentation

3. KNF SQA Requirements

- Classification of software

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3. KNF SQA Requirements

- Software development or change

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3. KNF SQA Requirements

- Software development or change (cont.)

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3. KNF SQA Requirements

- Software development or change (cont.)

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3. KNF SQA Requirements

- Software error reporting and corrective actions

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3. KNF SQA Requirements

- Software maintenance and discarding

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3. KNF SQA Requirements

- Configuration management

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3. KNF SQA Requirements

- In-use tests

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- Documentation

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4 ROPER Code V&V Plan

4. ROPER Code V&V Plan

- Classification of ROPER

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4. ROPER Code V&V Plan

- Key elements of ROPER V&V

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4. ROPER Code V&V Plan

- Key elements of ROPER V&V (cont.)

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4. ROPER Code V&V Plan

- Key elements of ROPER V&V (cont.)

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4. ROPER Code V&V Plan

- Contents of ROPER V&V Plan (SVVP) Document

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4. ROPER Code V&V Plan

- ROPER code validation methods

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4. ROPER Code V&V Plan

- Validation methods of ROPER models

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4. ROPER Code V&V Plan

- Validation methods of ROPER models (cont.)

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4. ROPER Code V&V Plan

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5 Summary

5. Summary

- ROPER code has been developed and maintained under KNF QA program, following ASME NQA-1-1994/1995a.
- KNF will be developing ROPER code for NuScale SMR fuel rod design in accordance with ASME NQA-1-2008/2009a.
  - V&V of ROPER code will be completed by 2nd quarter of 2014.

Thank you for your attention!

– Q & A –

Fuel Design Limits – DSRS 4.2

Larry Linik
Nuclear Engineering Supervisor

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Fuel Design Limits

- SRP 4.2 – I.1 Design Bases: “evaluate established (past) design-basis limits and associated SAFDLs to determine whether they remain applicable to the new fuel design”
- Design Basis limits address
  - Fuel System Damage - GDC 10
  - Fuel Rod Failure - 10 CFR Part 100
  - Fuel Coolability - 10 CFR 50.46, GDC 35 & GDC 27

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Fuel Design Limits

- Preliminary evaluation of SRP acceptance criteria presented December 2012
- For postulated accidents
  - core remains covered
  - clad heatup not expected

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Fuel Design Limits

- Appendix K white paper submitted December 2013 (WP-1013-5124-P)
  - requirements are met because core remains covered
  - no clad heatup, and quench conditions are not encountered
- Planned pre-application submittals
  - white paper on SRP 4.2 acceptance criteria evaluation

Nuclear Analysis – DSRS 4.3
Code Qualification

Code Qualification Process

- NuScale size reactor core-follow and criticals do not exist

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Code Qualification Process

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Thermal-Hydraulic Design DSRS 4.4

Dr. Kent Welter

Nuclear Safety Engineering Manager

Outline

- Regulatory requirements
- Thermal-hydraulic analysis – code qualification
- Core and reactor coolant system instabilities
  - design and operational requirements
  - evaluation method
- DSRS implications
- Pre-application submittals

Regulatory Requirements

- SRP Section 4.4 review confirms that the thermal and hydraulic design of the core and RCS
  - uses acceptable analytical methods
  - is equivalent to or is a justified extrapolation from proven designs
  - provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and anticipated operational occurrences (AOOs)
    NOTE: references AOO and ATWS evaluations in Chapter 15
  - is not susceptible to thermal-hydraulic instability

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T/H Analysis – Code Qualification

- Using and qualifying NuScale proprietary sub-channel analysis code - SCANR
- SCANR implements correlations appropriate for NuScale natural circulation conditions
- Analytical methods are to a large extent equivalent to those used in proven designs
- Establishes acceptable safety margin during normal operation and AOOs
- Establishes operating regime to prevent thermal-hydraulic instabilities

Core and Reactor Coolant System Instabilities

- SRP Chapter 15 review confirms that core and RCS instabilities during AOOs and ATWS events do not lead to unacceptable plant behavior
- Regulatory precedents
  - AP1000, EPR, and US-APWR design certification submittals evaluated core and RCS thermal-hydraulic stability in DCD Section 4.4 in a similar fashion, focusing on
    - demonstrated margin to density wave oscillations (e.g., percent power increase to initiate instability)
    - impossibility of Ledinegg instability due to the system pressure drop characteristics (“pressure difference vs. flow rate” comparison between the pressure gain and pressure loss)

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Design and Operational Requirements

- Core Functional Specification

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Design and Operational Requirements

- Core Functional Specification

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Design and Operational Requirements

- Reactor Coolant System Functional Specification

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Evaluation Approach

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DSRS Implications

- No changes to Section 4.4
- No changes to Chapter 15
  - Section 15.9 not applicable, PWR instabilities during transients are event dependent
  - instabilities will be addressed on a case-by-case basis within each event class or type in Chapter 15
  - focus on detecting and suppressing instabilities during AOOs

Pre-Application Submittals

- Critical heat flux (CHF) test program technical report
- CHF correlation technical report
- SCANR engineering report
- SCANR V&V technical report

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Control Rod Drive System – DSRS 4.6

Larry Linik

Nuclear Engineering Supervisor

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Reactivity Control Functions

- CVCS
  - normal startup and shutdown sequence
  - chemical shim
  - nonsafety-related
- CRDS
  - normal startup and shutdown sequence
  - safe shutdown for DBEs, ATWS, and SBO
  - safety-related

Control Rod Drive System

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Control Rod Assembly

- Conceptual design sufficient to demonstrate fuel design performance
  - basic dimensions
  - masses
  - materials
- CRA Licensing Path to Operation
  - Final design controlled by design acceptance criteria (DAC)
  - Commercial vendor (e.g., KNF, AREVA, Westinghouse) will supply CRA for the COLA
  - Selected by customer, supported by NuScale

NuScale Chapter 4 DSRS

Proposed Application of NRC Review Guidance to NuScale DCD Chapter 4				
Section	Section Title	Gap Analysis Conclusion re SRP	Proposed Basis Document for NuScale DSRS	NuScale DSRS Section Needed
4.2	Fuel System Design	See Dec 2012 presentation	Use mPower DSRS Section w/Modification	Yes
4.3	Nuclear Design	Use As-Is	Use mPower DSRS Section w/Modification	Yes
4.4	Thermal and Hydraulic Design	Use As-Is	Use mPower DSRS Section w/Modification	Yes
4.5.1	Control Rod Drive Structural Materials	Use With Modification	Use mPower DSRS Section w/Modification	Yes
4.5.2	Reactor Internal and Core Support Structure Materials	Use As-Is	Use mPower DSRS Section w/Modification	Yes
4.6	Functional Design of Control Rod Drive System	Use As-Is	Use mPower DSRS Section w/Modification	Yes
BTP 4-1	Westinghouse Constant Axial Offset Control (CAOC) (Former Section 4.3 BTP has been separated into an individual section.)	NA	-	No

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Next Steps

- Pre-application submittals
  - White papers
    - Appendix K - submitted December 2013 (WP-1013-5124-P)
    - Fuel acceptance criteria evaluation
  - Technical reports
    - CMS to MCNP benchmark application
    - CHF test program
    - CHF correlation
    - SCANR engineering report
    - SCANR V&V
    - Preliminary fuel design
    - ROPER V&V plan
- Future meetings on review of pre-application submittals as needed

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Requested Feedback

- Fuel design schedule – completing out-of-pile testing after DC submittal
- Open market fuel solution – initial fuel design for DC with COL departure for different commercial fuel design
- Fuel design limits – redefine or appropriately treat Section 4.2 acceptance criteria from 50.46 and Appendix K that do not apply, for example clad oxidation
- Flow instability – acceptability of approach
- CRA conceptual design – sufficient to demonstrate fuel design performance

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