NuScale US460 Risk-Informed Performance-Based (RIPB) Design and Licensing

Kent B. Welter, Ph.D.

Chief Engineer, Testing and Analysis

Acknowledgement and Disclaimer

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Presenter Bio – Dr. Kent Welter

Chief Engineer, Testing and Analysis

- 20+ years' experience in technology development, systems engineering, operations management, safety analysis, risk assessment, and testing.
- Overall responsibility to provide cross-functional technology leadership and oversight for NuScale nuclear power plant testing and analysis activities including:
  - Design basis events / safety analysis
  - Severe accidents
  - Probabilistic risk assessment (PRA)
  - Radiological
  - Shielding
  - Criticality safety
  - Core and fuel analyses
- Joined NuScale in 2008 as a member of the original startup team. Before joining NuScale, worked at the U.S. Nuclear Regulatory Commission in the Office of Research for 6 years.
- Ph.D. in Nuclear Engineering from Oregon State University with a research focus on advanced thermal-hydraulics of new reactors.
- Member of the American Nuclear Society (ANS) Standards Board and Chair of ANS 30.3 Standard, "Advanced Light Water Reactor Risk-Informed Performance-Based Design," Working Group.

Outline

- Design overview
- US460 risk-informed performance-based (RIPB) approach
- Design basis safety analysis
- Probabilistic risk assessment
- Performance-based decision making
- Defense-in-depth


Artistic concept of the NuScale Power Plant

NuScale's Mission

NuScale Power provides scalable advanced nuclear technology for the production of electricity, heat, and clean water to **improve the quality of life for people around the world.**

We will achieve this mission by providing technology that is:

- SMARTER
- CLEANER
- SAFER
- COST COMPETITIVE

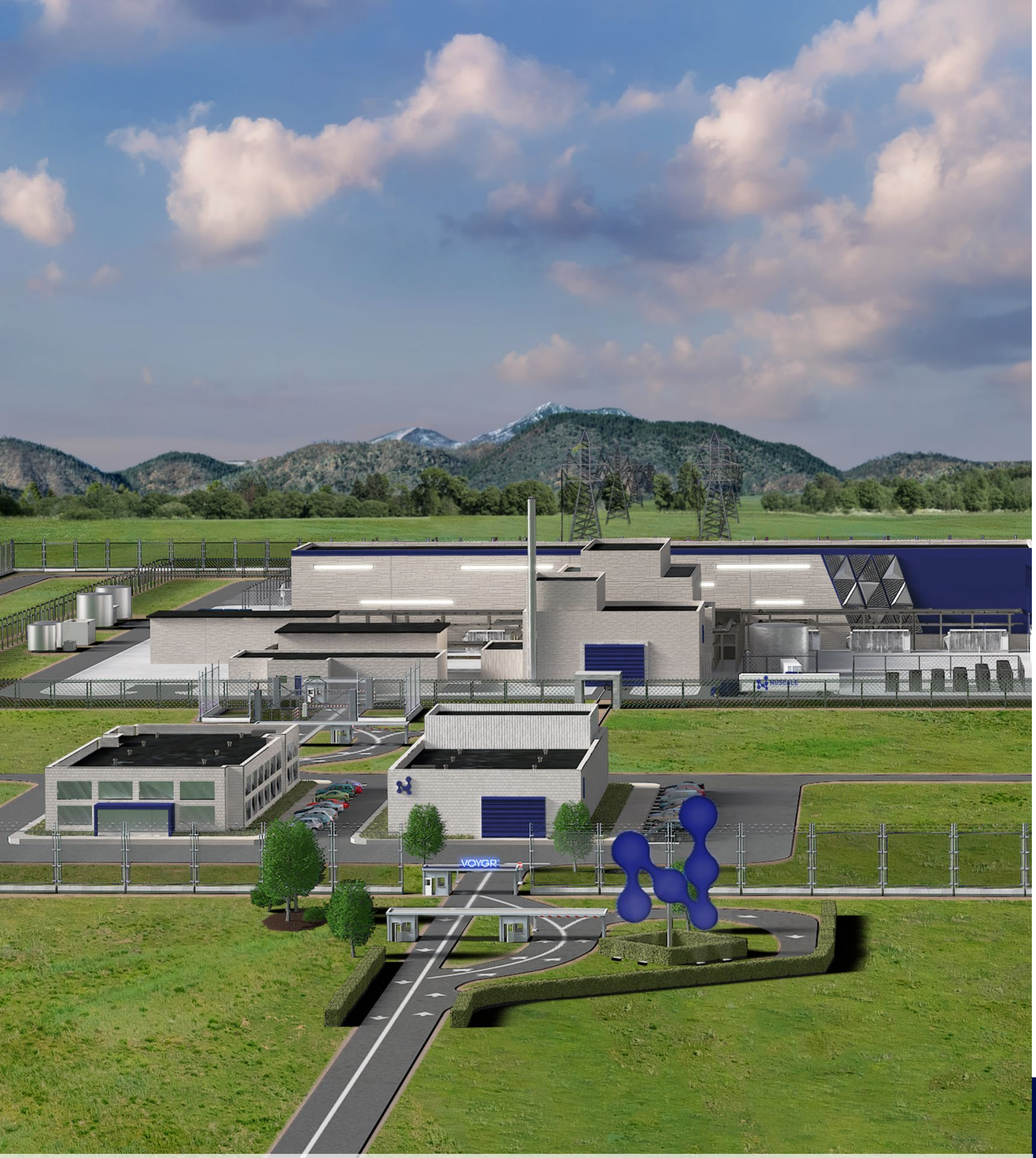
SMR Configurations

- Each plant is comprised of a different configuration of NuScale Power Modules (NPM) and output

12-Module Plant	6-Module Plant	4-Module Plant
924 MWe 3,000 MWt	464 MWe 1,500 MWt	308 MWe 1,000 MWt

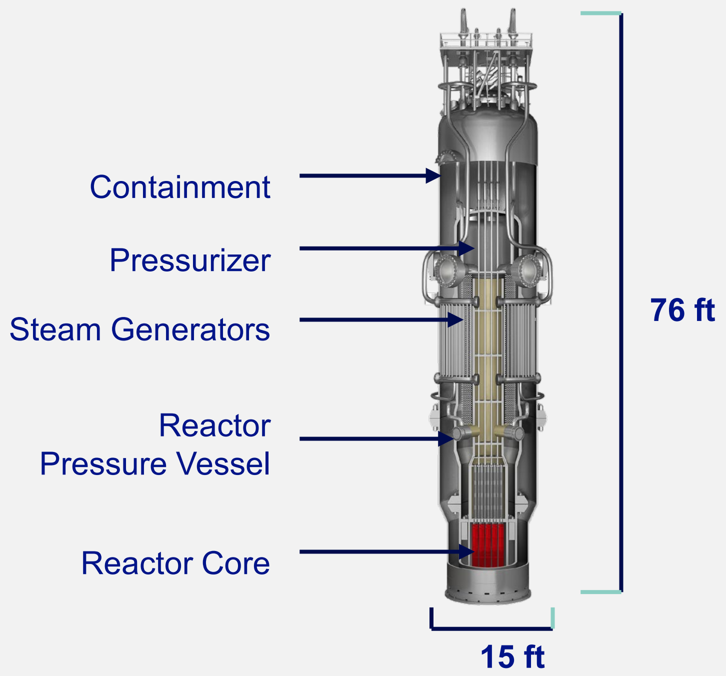
- Reference plant design
  - 924 MWe 12-module plant
  - Design approved by U.S. NRC in August 2020
  - Certified in 2023
- 4-module and 6-module plants contain all features and capabilities of reference 12-module plant

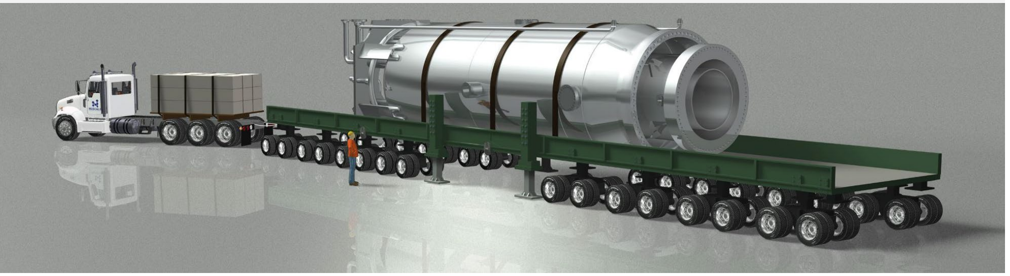

6-Module Plant Reactor Building



Artistic concept of the NuScale Power Plant

The NuScale Power Module™





- Groundbreaking technology features a **fully factory fabricated** NuScale Power Module™ consisting of an **integral nuclear steam supply system** in which the reactor core, steam generators and pressurizer are all contained in a single vessel housed in a containment vessel.
- **Simple design** eliminates reactor coolant pumps, large bore piping and other systems and components found in conventional reactors
- Modules individually refueled once every 18-22 months. Staggered refueling. 10 day refueling, inspection and maintenance outage.
- Modules can be incrementally added to match load growth

NuScale Power Module™ Specifications	
Electrical Capacity	77 MWe
Modules per Plant	Up to 12 (924 MWe)
Design Life	60 years
Fuel Supply	Existing light water reactor nuclear fuel
Safety	No AC/ DC Power, No Operator Actions, No additional water to keep reactors cooled for unlimited period.
Emergency Planning Zone (EPZ)	Supports EPZ as small as the site boundary

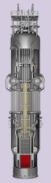
Inherently Safe Design Sets New Industry Standards

Unlimited Coping Period for Reactors

Comparison of Reactor Coping Period Following an Extreme Station Blackout
(loss of both AC and DC power)

Generation II Reactors:
4-8 Hours with Significant Operator Actions Required

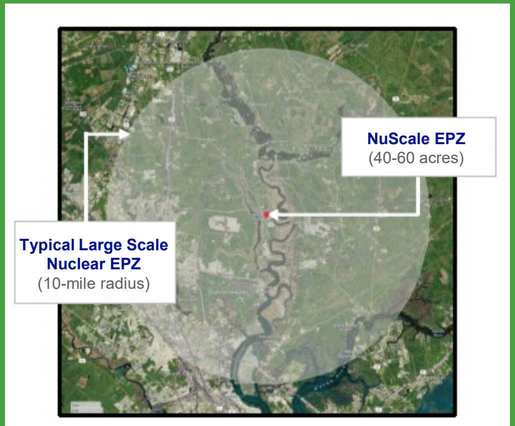
Generation III & III+ Reactors:
Up to 72 Hours with No Operator Actions



UNLIMITED WITH NO OPERATOR ACTIONS OR EXTERNAL SUPPORT

U.S. NRC-Approved Methodology to Support Emergency Planning Zone (EPZ) As Small As the Site Boundary

The smaller EPZ enables NuScale SMR power plants to better accommodate siting in close proximity to end-users, which is of particular importance to process heat off-takers and for repowering retiring coal-fired generation facilities



Williams Power Station (Coal),
South Carolina
Announced Retirement Date of 2028

Unparalleled Capability and Performance

"Black-Start" and "Island Mode" Capabilities
A NuScale SMR power plant can be started without the need for power from the grid and can operate disconnected from the grid – a first for a nuclear power plant

First Responder Power
A NuScale SMR power plant can start-up without power from the grid and can inject power back into the system to support grid restoration

Delivering Highly Reliable Power
Under a microgrid connection, a NuScale SMR power plant can provide 154 MWe of power to mission-critical installations at 99.95% reliability over the 60-year plant lifetime

Adaptable Siting Broadens Opportunity
A NuScale SMR power plant can be sited at the "end of the line" with only a single grid connection or off-grid

Risk-Informed Performance-Based (RIPB) Design and Licensing

What is risk-informed and performance-based (RIPB) design?

Risk-Informed

Combination of deterministic and risk-based approaches through a deliberative process

Deterministic

- Unquantified probabilities
- Design-basis accidents (prescriptive)
- Defense-in-depth and safety margins
- Can impose unnecessary regulatory burden
- Incomplete

Risk-Based

- Quantified probabilities
- Thousands of accident sequences
- Realistic / best estimate
- Incomplete

+

Performance-Based

Measurable results or performance outcomes based on objective criteria rather than a prescriptive process, technique, or procedure

- *"The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."* – PRA Policy Statement (1995)
- *"A performance-based requirement relies upon measurable (or calculable) outcomes (i.e., performance results) to be met, but provides more flexibility to the licensee as to the means of meeting those outcomes."* – SECY-98-144 (1999)

US460 RIPB Approach

- Built upon a robust systems engineering infrastructure (e.g., requirements engineering, product life cycle management, testing/validation)
- Updated D-RAP process to include DID adequacy evaluations and performance-based decision making to align with ANS-30.3-2022 (LWR Risk-Informed Performance-Based Design) provisions
- Updated (from US600 design) safety and risk performance objectives, along with key performance indicators for engineering change control
- Defense-in-depth philosophy
    - Prevent accidents or lessen the effects of damage if an accident or malfunction occurs
    - Multiple-barrier approach against fission product releases
    - Provide redundancy in safety function performance
    - Assure that safety will not be wholly dependent on any single design feature
    - Preference for selecting engineered over administrative controls
    - Risk analysis issued to improve engineering and operational decisions by identifying and taking advantage of opportunities to reduce risk
    - Assure that the public is adequately protected on the intended site and that well-conceived, workable emergency plans surround the nuclear facility
    - Utilize modern risk-informed performance-based (RIPB) principles and practices
- Conducted target RIPB evaluations to address specific issues or solutions (e.g., exemption requests, design changes)

ANS-30.3-2022, LWR Risk-Informed Performance-Based Design

ANSI approved July 21, 2022 and available for use/purchase

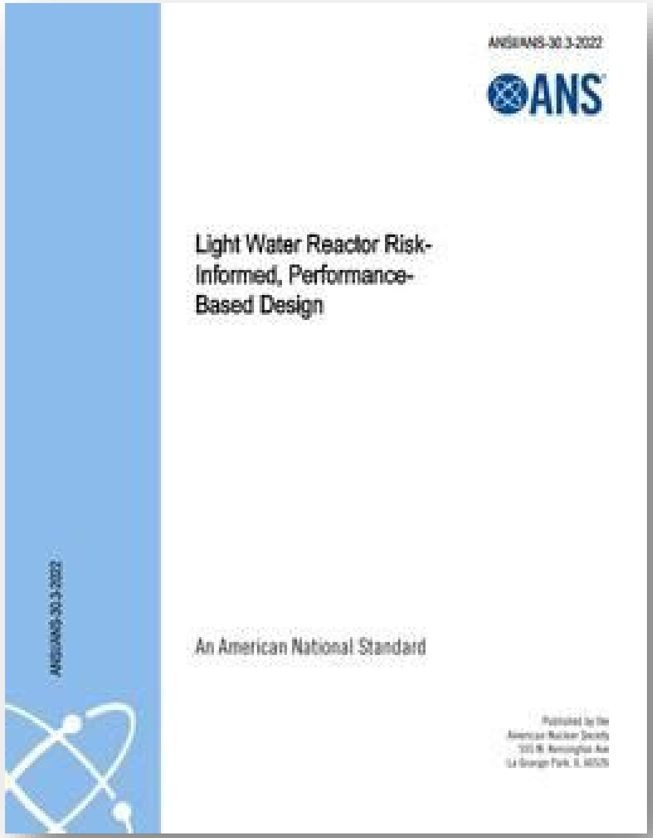
Working Group Chair: Kent Welter, NuScale Power

Purpose

- provides requirements for the incorporation of risk-informed, performance-based (RIPB) principles and methods into the nuclear safety design of new commercial light water reactors (LWRs)
- establishes a minimum set of requirements for the designer to follow in order to appropriately combine deterministic, probabilistic, and performance-based methods during design development

Scope

- definition of safety requirements
- licensing-basis event (LBE) selection
- design-basis safety analysis
- probabilistic risk assessments (PRAs)
- severe accident analysis
- classification and categorization of structures, systems, and components (SSCs)
- systematic defense-in-depth (DID) evaluations
- performance-based decision analysis



ANS 30.3-2022, Light Water Reactor Risk-Informed, Performance- Based Design

Key concept – hierarchy of functions, requirements, and physical systems

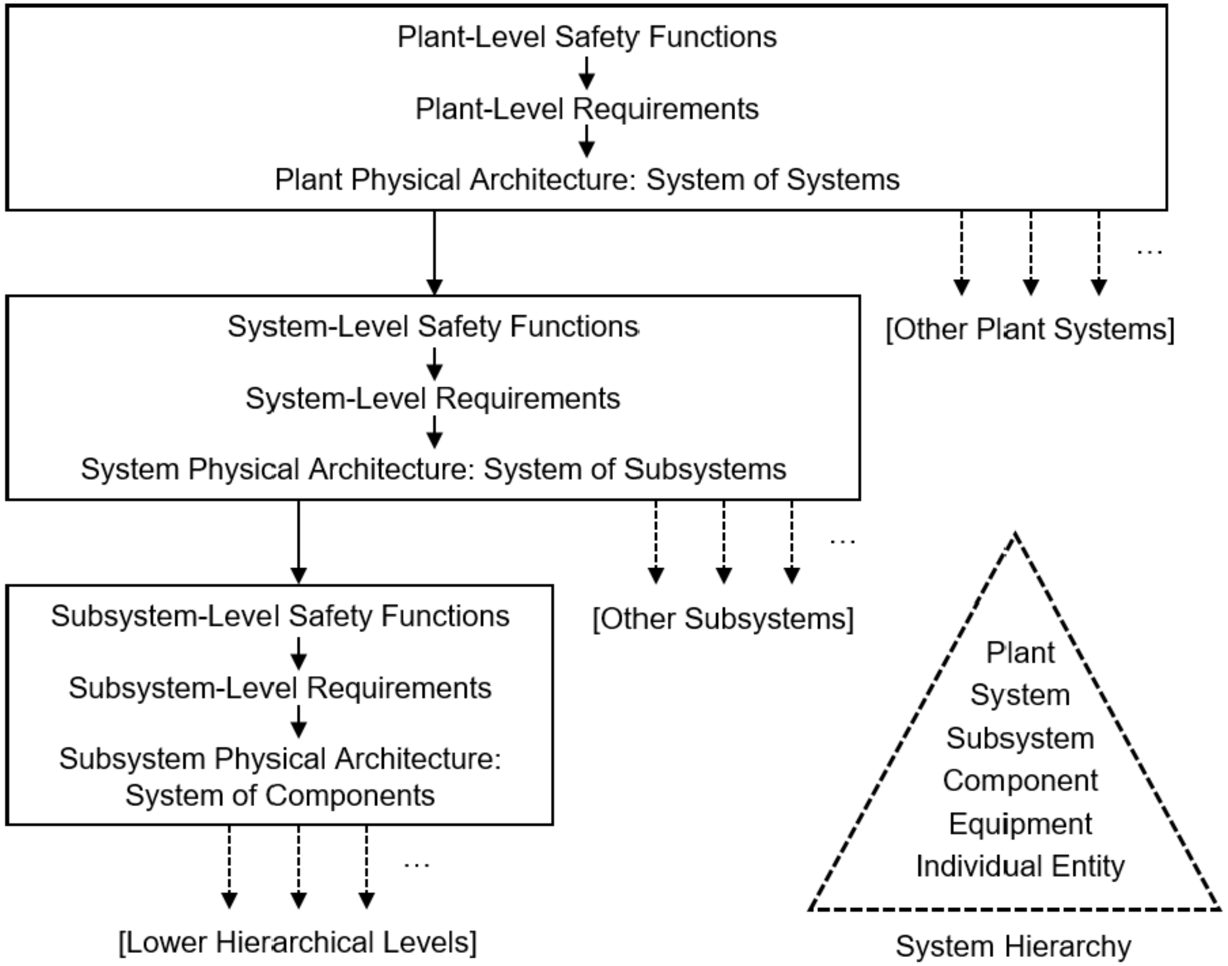
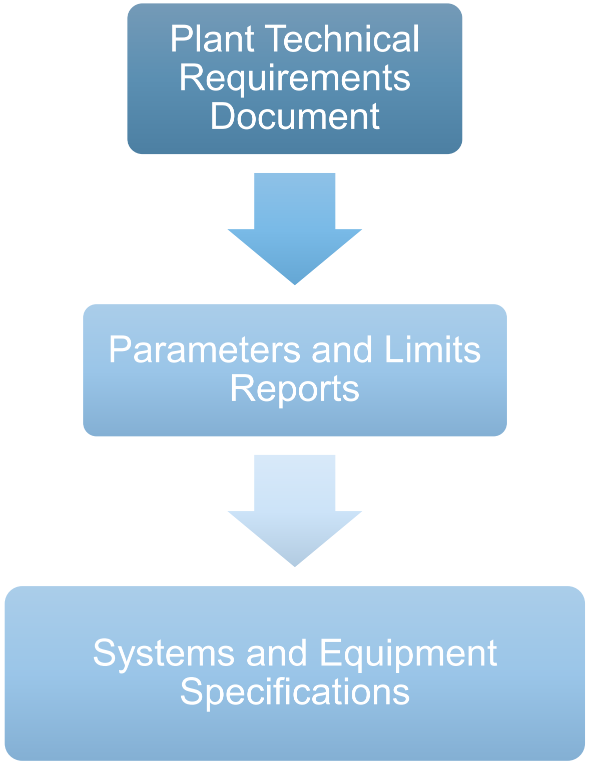


Figure 1 – Relationship structure for safety functions, safety requirements, and physical architecture

ANS-30.3-2022, LWR Risk-Informed Performance-Based Design

Performance-based design requirements

Requirements flow down



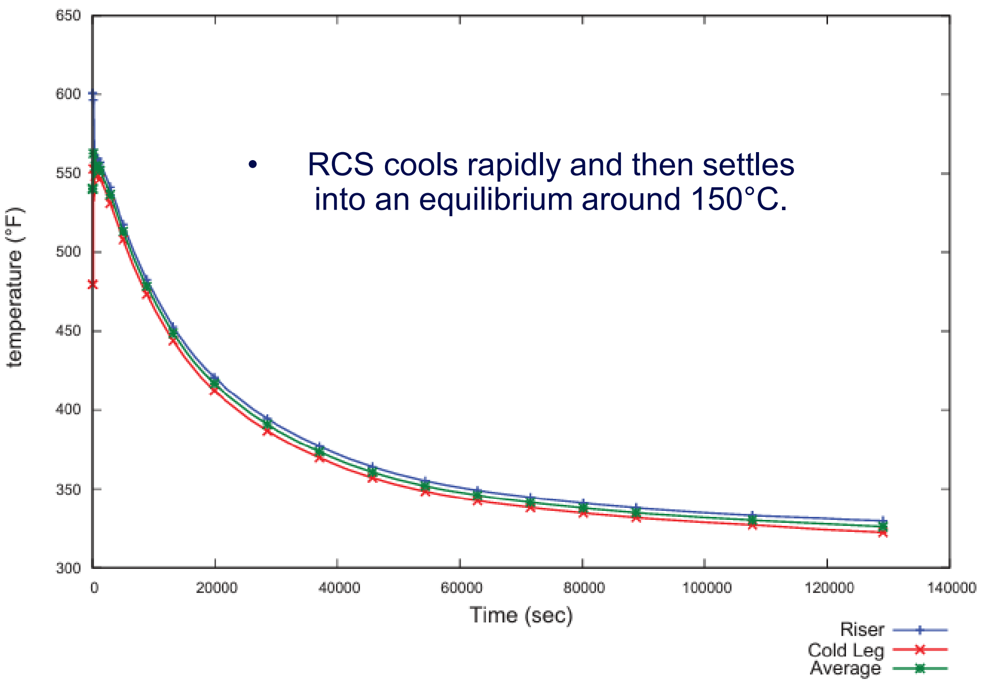
Example performance-based safety objectives from PTRD:

- The plant shall be capable of cooling the NPM without external makeup water to replace UHS boil off for a minimum of 30 days.
- After a design-basis event, the plant shall provide automated actions that place and maintain the NuScale Power Module (NPM) passively cooled with keff < 1 for at least 72 hours.
- Fuel thermal margin for normal operation and all anticipated operational occurrences (AOOs) shall be 5 percent greater than specified fuel design limits.
- The plant shall maintain core damage frequency for internal beyond design basis events during full power operation, below 1E-7 per module critical year (mcyr).
- Conditional containment failure probability shall be less than or equal to 0.1.

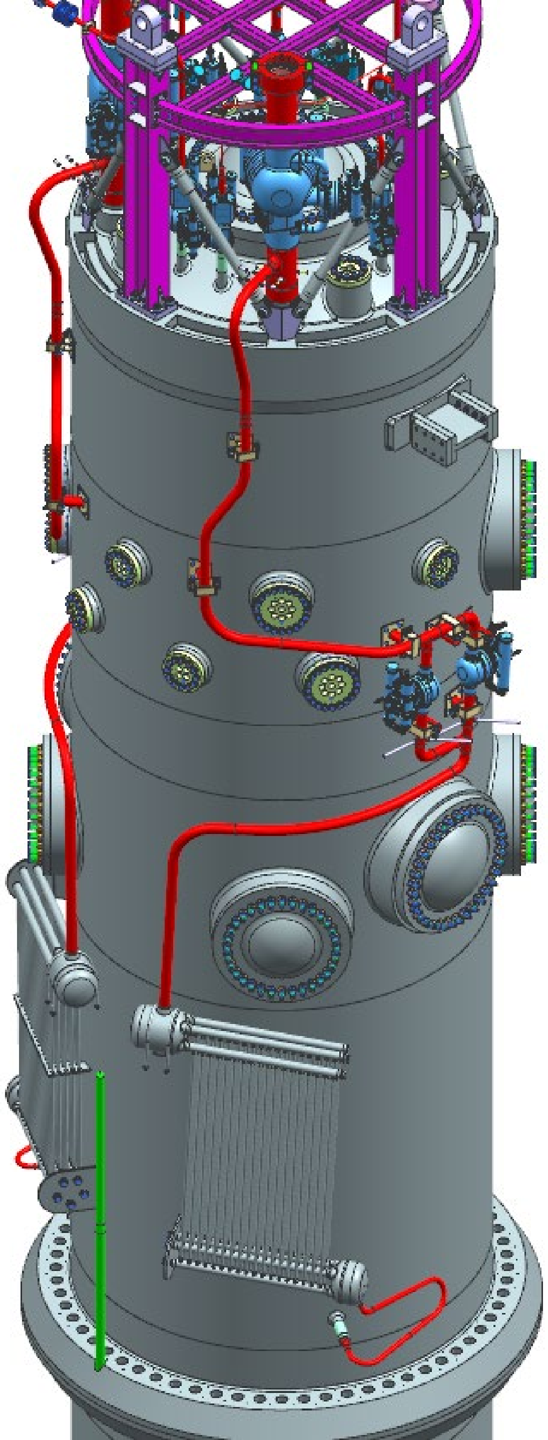
NuScale conformance assessment against ANS-30.3-2022

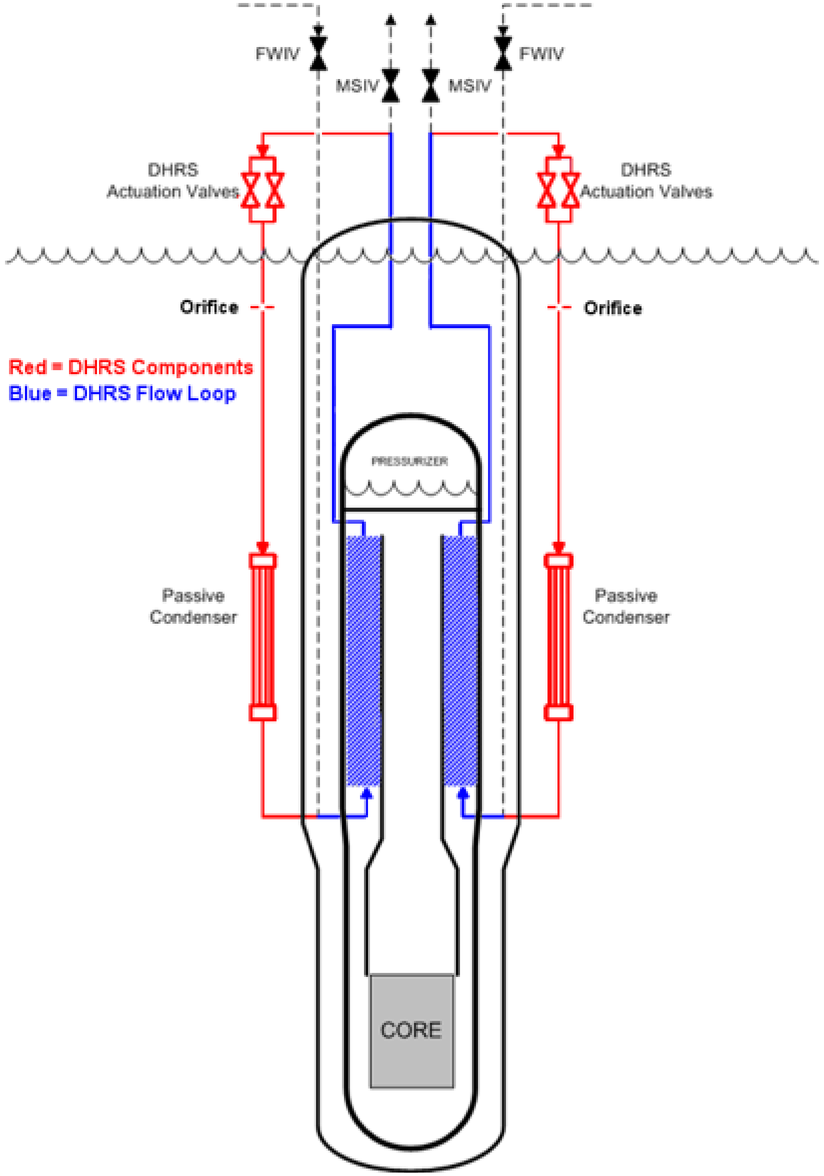
- 156 specific requirements in the standard (“shalls”)
  - 75 conform
  - 65 partially conform
  - 11 does not conform
  - 5 indeterminate
- Overall, NuScale conforms well to the ANS-30.3-2022 (90% conforms or partially conforms)
  - “Partially conforms” are generally because we meet the requirement on a project- or product-specific basis, but the requirement is not in a procedure or standard
    - NOTE: Requirements don’t have to be in procedure or standard if they are being effectively implemented on a consistent basis
  - “Does not conform” generally can be considered to be beyond the “state-of-practice” and venture into the “state-of-the-art”. In other words, they are advanced/modern approaches that may not be fully accepted yet by the LWR community.
  - Several requirements as written in ANS 30.3 were difficult to assess conformance (i.e., indeterminate):
    - “Any additional or alternative design requirements shall be justified in the design basis.”
    - “Incorporation of RIPB principles, methods, and results into the design shall consider the maturity of the design as it progresses through the phased design process.”
    - “<u>Proper</u> development of safety functional specifications shall be developed based on the derived safety requirements in order for the definition, organization, and validation of lower tier safety requirements to be achieved.”

DHRS Performance



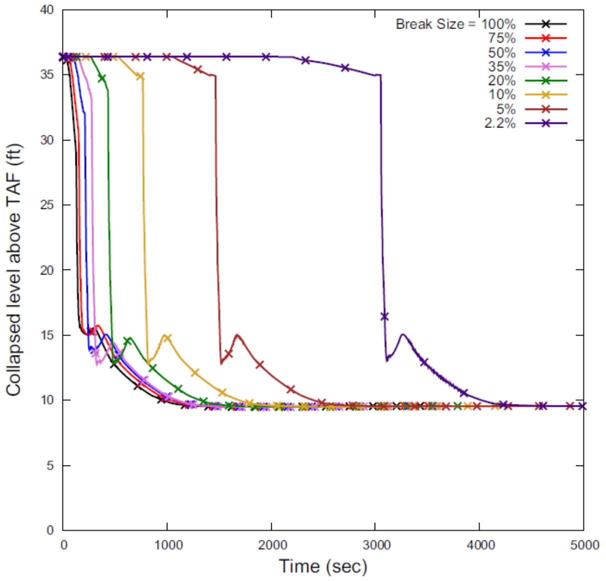
- The NuScale decay heat removal system (DHRS) is a safety-related, passive system that removes residual and decay heat
- Two redundant DHRS trains, each consisting of:
  - A passive condenser submerged in the reactor pool
  - TWO parallel actuation valves above the condenser (only one required)
  - Piping that connects the condenser to the main steam and feedwater piping for the steam generators



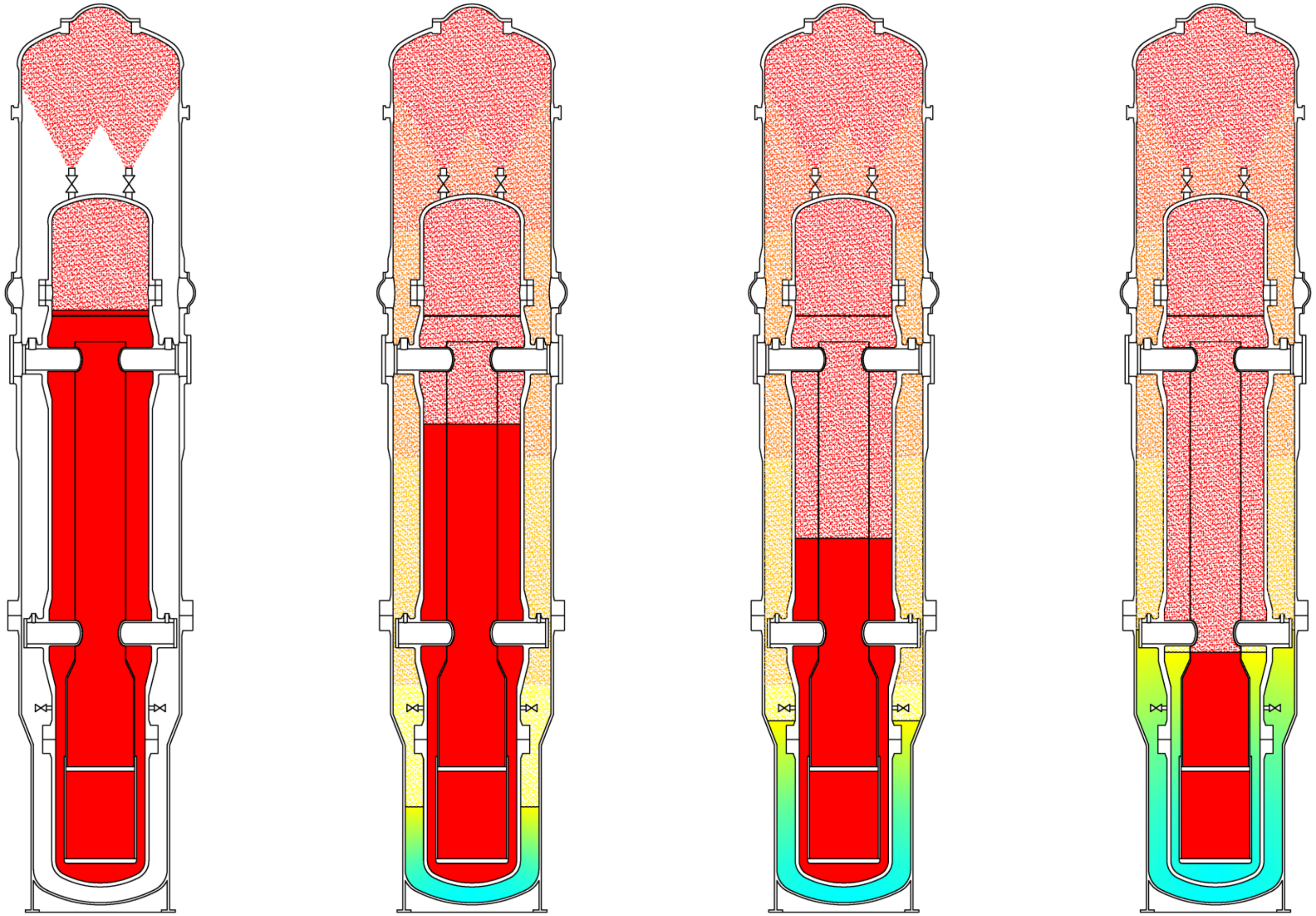


LOCA progression / decrease in RCS inventory transients

1. Pipe break
2. Reactor trip
3. CNV isolation
4. DHRS actuation
5. ECCS actuation
6. RCS depressurization
7. Steam condenses on CNV wall
8. ESB boron dissolved
9. Condensate fills bottom of CNV
10. Water returns through RRVs back into RCS
11. Core remains covered
12. Transition to long-term cooling to reactor pool



Collapsed level above TAF for RCS discharge line break



PRA Models Maintained with Design

- PRA models periodically updated to reflect design changes
- All operating modes and all hazards
  - full-power internal-events
  - low-power/shutdown
  - internal fire
  - internal flood
  - high winds and hurricanes
  - seismic
  - other external events

NRC ADAMS: ML20224A508

US460 PRA Summary Results

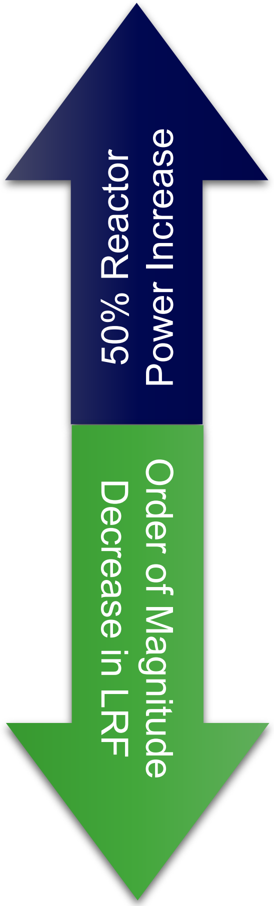
Hazard	CDF (per mcyr)	LRF (per mcyr)
Single Module		
Full power internal events	6.0E-09	6.6E-13
Low power and shutdown	4.0E-11	3.5E-12
Low power and shutdown (module drop)	1.8E-08	N/A
Internal fires	4.6E-09	1.3E-11
Internal floods	1.6E-10	3.4E-14
External floods	9.5E-09	1.4E-12
High winds (tornado)	2.6E-09	1.6E-13
High winds (hurricane)	1.9E-08	1.3E-12
Seismic (SMA)	0.92g	0.92g
Multi-Module		
	CCDP	CLRP
Multi-module factor	0.21	0.03

CCDP: Conditional core damage probability
CLRP: Conditional large release probability

PRA Key Safety Insights for US460

- Ultimate heat sink is co-located with NuScale Power Modules and protected by engineered, Seismic Class-1 structure
  - not susceptible to environmental hazards or disruptions in heat transfer systems
  - under conservative, bounding conditions will last many months
- Relevant safety systems consist of about a dozen fail-safe valves
  - heat transfer to UHS completely passive
- Small core size and self-limiting nature of heat transfer processes enhance reliability for maintaining core cooling
- Severe accident phenomena that may challenge containment in typical current generation plants (e.g., hydrogen combustion, molten-concrete interactions, high-pressure melt ejection, fuel-coolant interactions) do not challenge containment integrity in a postulated severe accident
- ***As demonstrated by the exceedingly small CDF and LRF coupled with the low multi-module factors, <u>radiological risk to the public beyond the standard site boundary is negligible</u>***

US460 Compared to US600



Structured DID adequacy evaluation

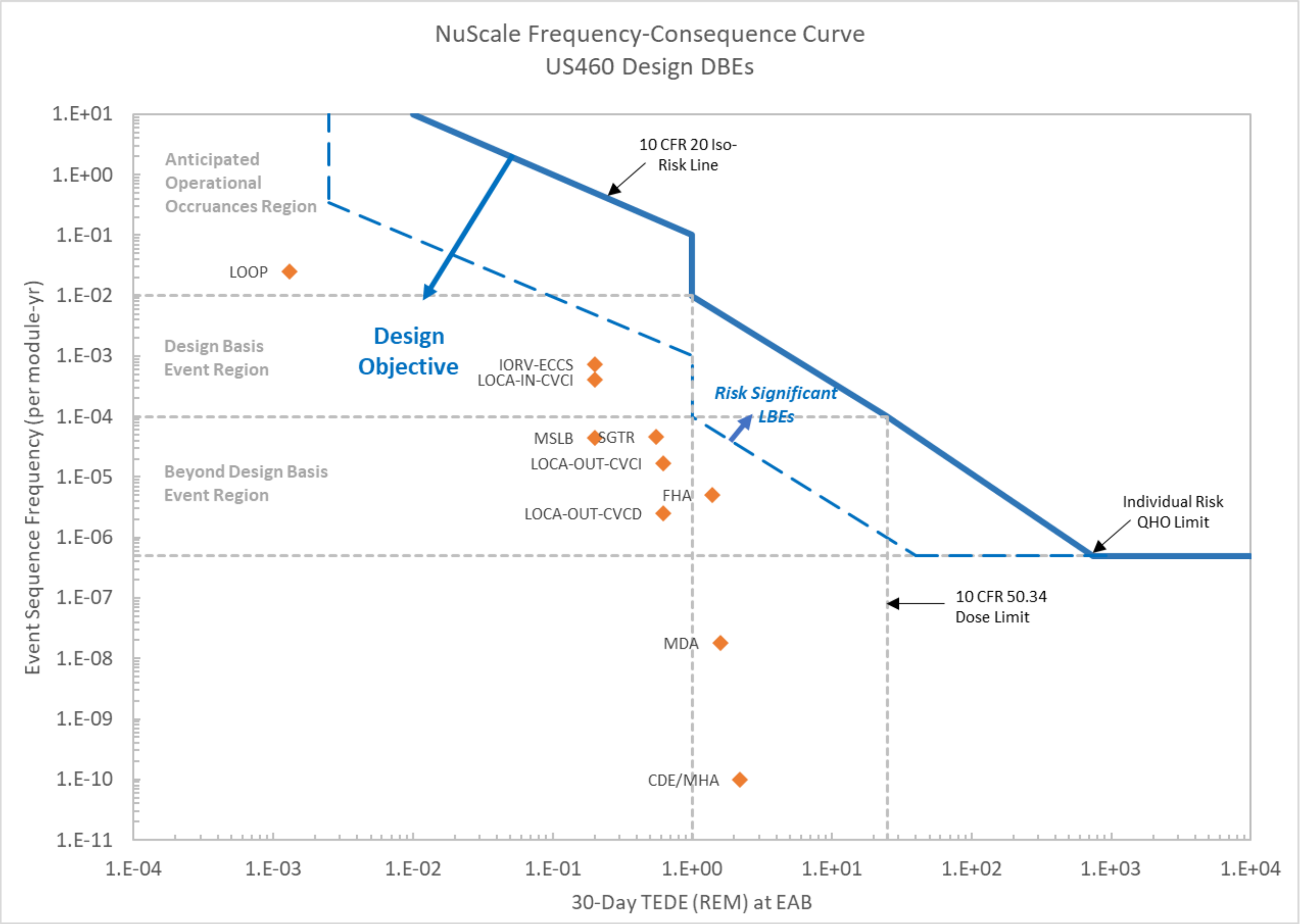
- The objective of a structured DID is to segregate this philosophy of redundancy and multiple barriers to be specific in its prevention or mitigation roles.
- During the structured DID adequacy evaluation, the D-RAP expert panel evaluated the following criteria:
  - Confirm whether any single feature is excessively relied upon to achieve public safety objectives, and, if so, identify options to reduce or eliminate such dependency.
  - Confirm that adequate technical bases for classifying SSCs as safety-related and risk-significant exist and their capabilities to execute safety functions are defined.
  - Confirm that the effectiveness of physical and functional barriers to retain radionuclides in preventing or limiting release is established.
  - Review inherent safety features that ensure the likelihood of an accident entailing severe core damage and the magnitude of radioactive releases in a severe plant condition are both kept as low as (is) reasonably achievable.
  - Confirm that a balance between event prevention and mitigation is reflected in the layers of defense for risk-significant licensing basis events (LBEs).
  - Review the technical bases for important characteristics of the LBEs with focus on the most risk-significant LBEs, and LBEs with relatively higher consequences. The technical bases for relatively high-frequency LBEs that are found to have little or no release or radiological consequences is also a focus of the review.
  - Evaluate whether sources of uncertainty that need to be addressed via programmatic and plant capability DID measures have been adequately addressed.

Review of licensing basis events

The D-RAP expert panel concluded that:

- No additional programmatic or plant capability DID was needed to address the LOOP event since highly reliable passive safety-related SCCs are relied upon.
- No additional programmatic or plant capability DID was needed to address the CDE/MHA event, since the dose at the EAB is low while only needing to rely on highly reliable, passive safety-related SSCs for mitigation and the event has an extremely low probability.
- No risk-significant LBEs.
- No systems were classified as risk-significant simply due to DID considerations as there are so many layers of redundancy.

NOTE: there are a cluster of events between the 1E-6 to 1E-3 frequency and 0.1 to 1.5 rem dose that are relatively close to the design margin target on the F-C curve. These events represent the central events analyzed in the Chapter 15 safety analysis and have been extensively evaluated; no additional scrutiny by the D-RAP expert panel was deemed necessary or warranted, since only safety-related systems are relied upon to mitigate these events and margin to these events are adequate.



CDE/MHA: Core damage event/maximum hypothetical accident
FHA: Fuel handling accident
IORV-ECCS: Inadvertent actuation of a reactor vent valve (RVV)
LOCA-IN-CVCI: Loss-of-coolant accident, break of the CVC injection line inside containment
LOCA-OUT-CVCI: Loss-of-coolant accident, break of the CVC injection line outside containment
LOCA-OUT-CVCD: loss-of-coolant accident, break of the CVC discharge line outside containment
LOOP: Loss-of-offsite power
MDA: Module drop accident
MSLB: Main steam line break
SGTR: Steam generator tube rupture

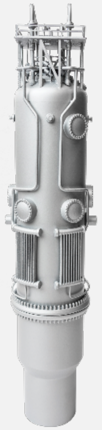
Defense-in-depth comparison

Defense-in-Depth Level	Conventional LWR	NuScale SMR
1 Prevent abnormal operation and failures	• ~20 safety-related systems • Historical set of design-basis events • Active safety systems that require electrical power	• Fewer than 10 safety-related systems required for safe operation* • Reduction in design-basis events due to simplified design • Passive safety systems*
2 Control of abnormal operation and detection of failures	• Manual control and performance monitoring • Multiple active systems required to protect critical assets	• Highly-automated control and performance monitoring • Simple, passive systems to protect assets*
3 Control of accidents within the design basis	• Significant reliance on design-basis operator actions • Several design-basis events lead to core damage	• No design-basis operator actions • ***No design-basis events lead to core uncovery****
4 Control of severe accident conditions	• Numerous active systems and operator actions requiring power • Core damage frequency (CDF) ~$1x10^{-5}$ • Minimal fission product barriers beyond containment	• Passive systems requiring no power or operator action • CDF < $1x10^{-7}$* • Additional fission product barriers beyond containment
5 Mitigation of consequences of significant radiological releases	• Large early releases • Emergency planning zone = 10 mi	• Small delayed releases* • Emergency planning zone as small as the site boundary

**IAEA DID improvement recommendation (INSAG-12, Basic Safety Principles for Nuclear Power Plants)*

U.S. Licensing Status

- Design Certification Application (DCA) for 12-module (US600) plant (50 MWe NPM) completed in December 2016
- Docketed and review commenced by U.S. Nuclear Regulatory Commission in March 2017
- NuScale received Standard Design Approval (SDA) in September 2020
- Design certification received in February 2023 for US600
- **New SDA received in May 2025 for 6-module (US460) plant (77 MWe NPM)**



NuScale Power Makes History
First Ever Small Modular Reactor to Receive U.S. Nuclear Regulatory Commission Design Approval and Certification

RIPB excerpts from NRC SER on NuScale US460 design

- "In accordance with SECY-11-0024,1 the NRC staff utilized a risk-informed approach for its review of instrumentation and controls by considering both the safety classification and risk significance of each structure, system, and component (SSC) to help determine the appropriate level of review for each SSC."
- "For the onsite DC systems, in SER Section 8.3.2, the staff used a risk-informed, graded approach to evaluate the quality aspects of the augmented DC power system (EDAS)."
- "Using a risk-informed approach, the NRC staff based its review of the radioactive material content of only a single NPM operating at the Technical Specification coolant specific activity limit, with the other five units operating with realistic coolant activity concentrations."
- "Using a risk-informed approach, the NRC staff's evaluation of radiation shielding focused on areas of the facility that could contain high concentrations of radioactive materials during normal operation or following accidents."
- "The applicant has not requested approval of, nor is the staff approving, the post-CHF models in NRELAP5. Nevertheless, for the purposes of a risk-informed approach supporting the staff's evaluation of the rationale provided to support a regulatory exemption, unvalidated PCT results can provide insights into the magnitude of consequences, and due to the large margins to the acceptance criteria, the staff has determined that this approach is reasonable for this beyond design basis event."
- "The staff's determination regarding the adequacy of the justification for the exemption request considers the overall safety significance of LOCAs for the US460 design at the subject locations and is based on a set of integrated risk-informed principles, such as, likelihood, consequences, safety margin, defense in depth, and performance monitoring."

Summary and conclusions

- New (relatively) RIPB guidance has been developed to support more efficient design and licensing of new and advanced reactors
- RIPB is a balance of deterministic and probabilistic analysis to help reduce uncertainties and focus on what is important (from a safety perspective) in order to make better decisions
- NuScale has implemented RIPB principles and methods in the design and licensing of its reactors that have resulted (in part) to an extremely safe design
- NuScale continues to use RIPB methods in the design and licensing of new product offerings and sees it as an important part of new reactor deployment in the US and around the world

Kent Welter

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kwelter@nuscalepower.com

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THANK YOU!

Background slides

Development of RIPB Regulation

- WASH-1400, The Reactor Safety Study (1975)
- PRA Implementation Plan (1994 to 1999)
- PRA Policy Statement (1995)
- White Paper on Risk-Informed and Performance-Based Regulation, SECY-98-144 (1999)
- Risk-Informed Regulation Implementation Plan (RIRIP) (2000 to 2007)
- NUREG/BR-0303, Guidance for Performance-Based Regulation (2002)
- Risk-Informed and Performance-Based Plan (RPP) (2007 to Present)
- Next Generation Nuclear Plant: A Report to Congress (2010)
- Use of Risk Insights to Enhance Safety Focus of Small Modular Reactor Reviews, Staff requirements memorandum (SRM), COMGBJ-10-0004/COMGEA-1 0-0001 (2010).
- Fukushima Near Term Task Force Recommendations (2011)
- NUREG-2150, A Proposed Risk Management Regulatory Framework (2012)
- Licensing Modernization Project for Advanced Reactor Technologies (2018)
- NEI 18-04, Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development (2019)
- RG 1.233, Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Application for Licenses, Certifications, and Approvals for Non-Light-Water Reactors (2020)

Historical change in CDF for the US600 design