PRA OVERVIEW

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June 2, 2010

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

NuScale Power

Outline

- NuScale PRA
  - PRA Background and Design Goals
  - PRA Scope and Approach
  - Multiple-module Analysis
  - Level-II
  - External Events
  - Overall CDF Results

Regulatory Background

- NUREG-0800 Chapter 19, “Severe Accidents”
- 10 CFR 50.34(f)(1)(i)
  - “Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant.”
- RG 1.206 C.I.19, Commission Safety Goals and Surrogate Objectives
  - Surrogate objectives of less than 1E-4/ry for core damage frequency (CDF) and less than 1E-6/ry for large release frequency (LRF) (without credit for the non-safety-related “defense-in-depth” systems)
  - Containment integrity is maintained for 24 hours following onset of core damage for more likely severe accident challenges
  - Containment conditional failure probability (CCFP) is 0.1 or less for composite of all core damage sequences assessed in PRA

NuScale-Specific Design Goals

- NuScale adopting individual safety goal for following events: Internal CDF, Seismic CDF, Other Externals CDF, Security CDF, and LRF Design Goals
  - Single modules
  - Cluster of 4-modules
  - Whole plant of 12-modules
  - Whole plant without credit to non-safety-grade systems

- Integrated result of design safety goals will exceed NRC's safety goals and surrogate objectives requirements

NuScale Approach: PRA Objectives

- Demonstrate NuScale reactor design is safe and active equipment, passive safety features, plant architectures, and expected operational practices offer adequate redundancy, diversity, and independence with sufficient defense-in-depth and safety margins
  - Evaluate potential design improvements for risk-significant scenarios in terms of core damage, containment failure, or large releases
  - Tool for decision-making in selecting among alternative design features, operational strategies, and design options
  - Human error prevention analyses tool for risk-significant human errors
  - Evaluate tolerances to severe accidents initiated by either internal or external events
  - Defense against risk-significant common-cause failures
  - Safety margins to address uncertainty
  - Single failure criteria

NuScale Approach: PRA Development Scope

- PRA will include all structures, systems, and components (SSCs); human actions; and common-cause failure considerations

- Standards / Guides / Requirements
  - Level-1 internal events: ASME/ANS RA-Sa-2009 “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” February 2009
  - R.G. 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Rev. 2, March 2009
  - External Events: Fire PRA (NUREG/CR-6850 and ANSI/ANS-58.23-2007), Seismic PRA (margins assessment); others (mainly internal floods and high wind-screening approach based on ANSI/ANS-58.21-2007)
  - Low power and shutdown (ANS-58.22-200x)

NuScale Approach: PRA Development Scope (cont.)

- Tools / Data
  - Generic data adapted to NuScale characteristics from NUREG/CR-6928 (February 2007) for hardware and initiating event frequencies
  - SPAR-H human reliability analysis methodology from NUREG/CR-6883 (2005)
  - Common-cause data from the NRC website
  - Fire Model: NUREG-1824, Fire PRA Methodology NUREG/CR-6850, Unit-Specific Fire Ignition Data from Inspection Manual Chapter (IMC) 0609
  - SAPHIRE 7.27 (PRA)
  - MELCOR 1.86 (Severe Accident)
  - RELAP5/Mod3.3 (Success Criteria)
  - AP1000 steam generator specification to scale NuScale SGTR initiating event frequency

Status of PRA Activities

- Completed (preliminary)
  - Single-Module Level-I PRA, Multi-Module Analysis (Quantitative Screening)
  - Low Power and Shutdown
  - Internal Fire Risk (Quantitative Screening), Internal Flood, High Wind, External Flood
  - Phase I SA Evaluation, “Storyboard” Approach
  - Single-Module Level-2 PRA (rough draft of plant damage states, critical severe accident sequences, containment event tree, release categories)
- In Progress / Future
  - Phase II Single/Multi-Module Level-1 PRA
  - Phase II SA Evaluation
  - Single/Multi-Module Level-2 PRA Quantification & Uncertainty
  - PRA-Based Seismic Margins Assessment
  - Single/Multi-Module Level-3 PRA

NuScale Initiating Events

1. CVCS Charging Line Break Between Containment and Vessel
2. CVCS Charging Line Break Outside Containment
3. CVCS Letdown Line Break Between Containment and Vessel
4. CVCS Letdown Line Break Outside Containment
5. Feedwater Line Break Downstream of FRV11
6. Feedwater Line Break Downstream of FRV12
7. Loss of Condenser Heat Sink
8. Loss of Main Feedwater
9. Loss of Offsite Power
10. Inadvertent Closure of MSIV11
11. Inadvertent Closure of MSIV12
12. Inadvertent Closure of Both MSIVs
13. Main Steam Line Break Downstream of MSIVs

NuScale Initiating Events (cont.)

14. Main Steam Line Break Upstream of MSIV11
15. Main Steam Line Break Upstream of MSIV11
16. Steam Generator Tube Rupture in Bundle 1
17. Steam Generator Tube Rupture in Bundle 2
18. SLOCA due to Spurious RVV11 Opening
19. SLOCA due to Spurious RVV12 Opening
20. SLOCA due to Spurious RRV11 Opening
21. SLOCA due to Spurious RRV12 Opening
22. General Transients

    (The Anticipated Transients Without Scram (ATWS) events were modeled as the integral part of applicable event trees.)

Event Sequence Modeling
Initiating Event Frequencies

- "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," NUREG/CR-6923, February 2007
  - Station Blackout Study – NUREG/CR-6890
  - Expert Elicitation for LOCA Frequencies – NUREG-1890

- Supporting Calculations
  - Frequency for CVCS line break estimated based on industry-average pipe failure rates for non-ESW (emergency service water) piping from NUREG/CR-6928, assuming a specific pipe length for NuScale reactor
  - NuScale SGTR initiating event frequency assessed based on comparisons with IRIS and AP1000

Event Sequence Modeling
Success Criteria Analysis

Function	Definition	System	Success Criteria
Reactivity Control	To achieve sub-criticality and maintain the reactor in a sub-critical state	- Reactor Protection System (RPS) - Control Rod Drive (CRD) - Chemical Volume and Control System (CVCS)	Automatic RPS actuation or manual CRD actuation; manual boration through CVCS
Decay Heat Removal	To remove reactor decay heat through the steam generators to the pool	- Steam Generators (SGs) - Decay Heat Removal System (DHRS)	1-out-of-2 SGs associated with 1 out of 2 DHRS trains
Containment Heat Removal	To remove reactor decay heat through the containment wall to the pool	-Containment Heat Removal System (CHRS) -Containment Cooling Pool (CCP)	1-out-of-2 CHRS trains
Coolant Injection	To provide makeup flow to the reactor pressure vessel	-CVCS	1-out-of-2 CVCS pumps

Event Sequence Modeling
Human Reliability Analysis

- Pre-Initiator Human Failure Events
  - Failure to Restore from Test or Maintenance (Including Failure to Restore Equipment Status, Failure to Set Setpoints correctly, etc.)
- Post-Initiator Human Failure Events
  - Failure to Align System
  - Failure to Isolate System
  - Failure to Control or Operate System
  - Failure to Generate Signals
  - Failure to Recover Hardware Failure or Offsite Power
- Errors of commission will be considered for certain special conditions (e.g., Post-Fire)

Event Sequence Modeling
Common Cause Analysis

- Basic Parameter Model – Alpha Factor Method
- NRC Website CCF Database
- Common-cause Failures Considered
  - Batteries, battery chargers – low power
  - RRVs, RVVs, DHRS valves/check valves – fail to open
  - MSIVs, CVCS isolation valves – fail to close
  - DHRS inlet screens, CVCS heat exchangers – plugging
  - CVCS pumps, EDGs – fail to start or run
  - Digital system components such as fiber optic receivers, fiber optic transmitters, digital input modules-fail to operate

Internal Flooding
Objective, Approach and Assumptions

- Considered single- AND multi-module plant, including LPSD phase of operation
- Submergence of safe shutdown equipment, water spray, dripping, or splashing
- Sources of flooding include pipe breaks and tank ruptures
- Level-1 internal PRA used to assess effects of flooding on safe shutdown equipment
- Approach includes:
  - Screen out areas not expected to be significant flooding risk
  - Estimate flooding frequency from flooding sources and piping segments
  - Analyze flood propagation scenarios for flooding areas based on Level-1 internal event PRA and assume that the worst scenario would be applicable
  - Apply the worst flood propagation scenario in each area and calculate the corresponding CDF/LRF, as well as total cumulative CDF/LRF, for all areas

Internal Flooding PRA
Preliminary Conclusion

- No flooding event will be risk-significant
- Overall internal flooding CDF for individual modules and multi-module plant will be very small
- Flooding during refueling shown to be insignificant
- Highest contributor to CDF is flooding events in the turbine building
- Flooding in remaining areas in reactor building are primarily due to fire protection pipe breaks

High Wind & External Flood PRA
Objective, Approach, and Assumptions

- Progressive screening approach of NUREG-1407 used for IPEEE assessment of risks due to high winds and tornado
- For NuScale design, a tornado or high wind event will not have significant impact on safety-related equipment
- Most limiting impact is loss of offsite power, which the plant is designed to cope with and is not a leading contributor to overall CDF
- External flooding analysis is highly site-specific
- No site specification for NuScale is available; hypothetical site characteristics are assumed and external flood analysis is performed to identify significant contributors to CDF resulting from damage caused by storms, dam failure, and flash floods

High Wind & External Flood PRA

Preliminary Conclusion

- Risks from high wind events on NuScale design are extremely low

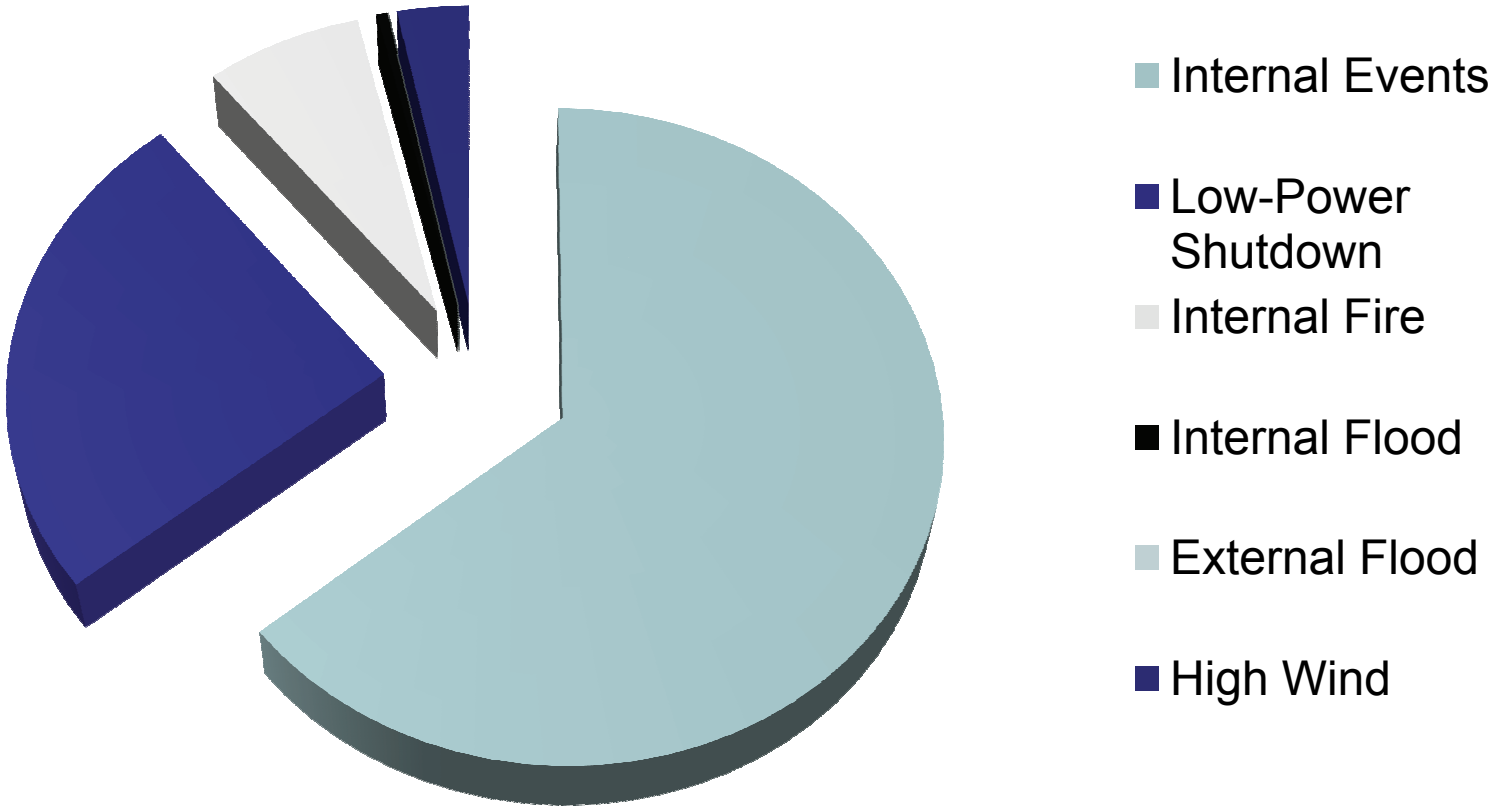
- External flooding design features of NuScale, in combination with requirements for building location relative to probable maximum flood (PMF) and maximum groundwater elevation, provide a robust design against potential external floods

NuScale Approach

Seismic Risk Analysis (planned)

- PRA-based seismic margin assessment (SMA)
  - Regulatory Guides 4.7, 1.198, 1.208; ASCE/SEI 43; ANSI/ANS 2.27, 2.26, 2.29; NUREG/CR-6926, etc.
- When large seismic margin exists, seismic risk contribution is considered acceptably small
- Seismic fragilities will be estimated based on seismic design and qualification criteria
- High confidence of low probability of failure (HCLPF) versus some peak ground acceleration (PGA) values will be calculated for important accident sequences identified from PRA
- For leading seismic scenario and/or certain single-failure critical structures with low HCLPF, limited-scope seismic PRA may be performed

Preliminary Single-Module CDF Results



Total Single-Module Mean CDF = 2.8 E-8 / Reactor-Year

NuScale Power

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