Developing a Low Power/Shutdown PRA for a Small Modular Reactor

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Overview

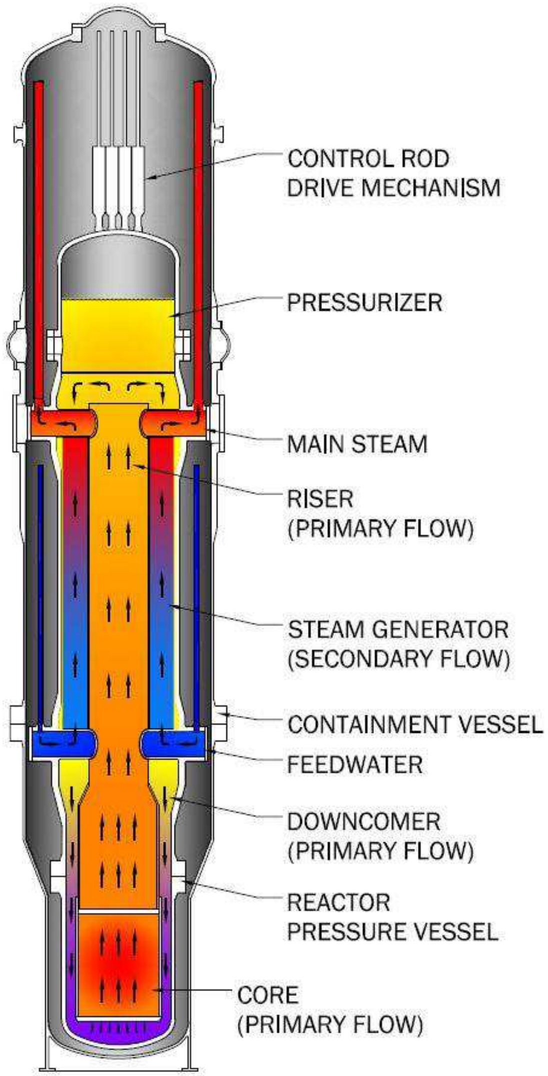
- Probabilistic risk assessment (PRA) has traditionally focused on events occurring during power operation
  - Internal events (transient, steam generator tube break, loss of coolant accident (LOCA))
  - Internal fire and flood
  - External events (earthquake, high winds, aircraft impact)
- Decay heat means risk of core damage is still present after shutdown
  - Core damage at Three Mile Island (1979) and Fukushima (2011)
- Increased activity may cause initiating events
  - Station blackout at Vogtle 1 (1990)

Agenda

- NuScale design overview
- NuScale refueling process overview
  - Plant operating states (POS)
- Developing low power/shutdown (LP/SD) PRA
  - Initiating event frequency
  - Accident sequences
  - Crane failure probability

NuScale Power Module

- **Factory built nuclear steam supply system:**
  - *Primary system and containment is prefabricated and shipped by rail, truck or barge*
- **Integral design with natural circulation cooling**
  - *Eliminates major accident scenarios*
  - *Eliminates many pumps, pipes, valves*
- **Immersed in large ultimate heat sink**
  - *Simplifies and enhances safety case*
- **Built on proven technology**
  - *Innovation is in the design and engineering*
- **Constructed below grade**
  - *Enhances security and safety*



Passive Safety

- All safety-related components are fail-safe valves that actuate cooling systems on loss of power
- Safety systems rely on passive processes of natural circulation and heat conduction
- Triple Crown for Nuclear Plant Safety™
  - The NuScale plant is able to safely shut down and self-cool, indefinitely, with:
    - No operator action
    - No AC or DC power
    - No additional water

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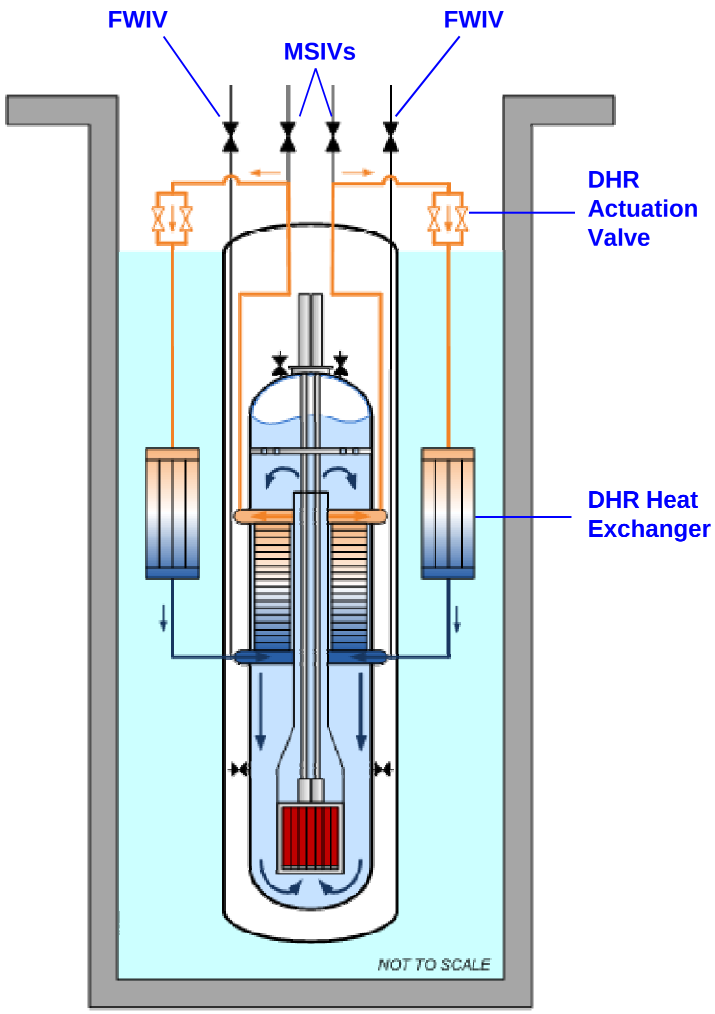
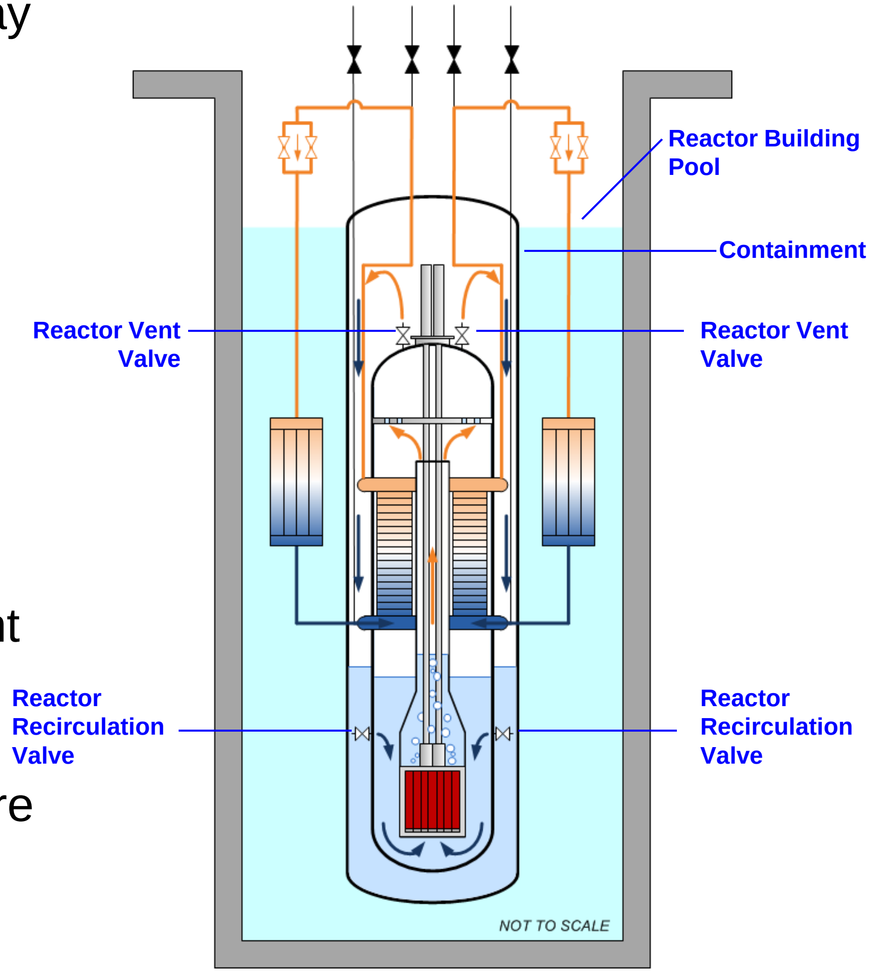
Decay Heat Removal System



- Two passive, independent single-failure-proof trains
- Closed loop system
- Two-phase natural circulation operation
- DHRS heat exchangers mounted directly on exterior of containment vessel--normally full of water
- Supplies secondary side coolant inventory
- Natural circulation of primary coolant is maintained
- Pool provides a >3 day cooling supply for decay heat removal

Emergency Core Cooling System

- 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 open to provide recirculation path through the core
- Provides >30 day cooling followed by unlimited period of air cooling



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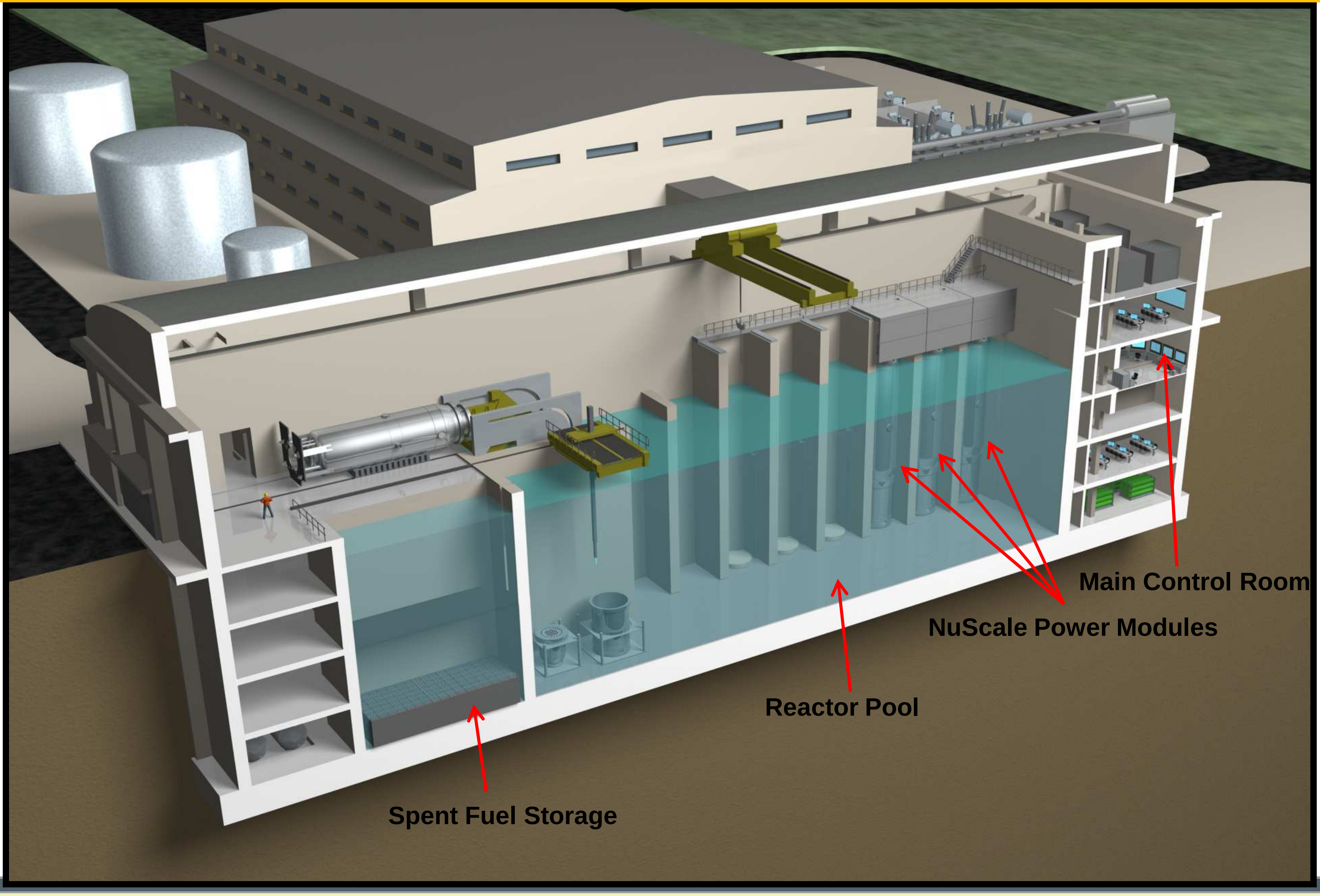
Reactor Pool

- Below-grade concrete pool with stainless steel liner provides seismic damping and radiation shielding during normal operation
- Ultimate heat sink for safety systems
  - DHR system heat exchangers
  - Condensation and conduction through containment vessel when ECC valves are open
- Inventory provides water cooling of containment for 30 days, air cooling indefinitely
- Spent fuel pool provides at least 30 days of passive cooling of fuel assemblies

Reactor Building



Refueling Procedure

- Module shutdown and initial cooling (POS1)
  - Cooling with normal secondary cooling (turbine bypass)
  - Begin containment flood
- Cooling through containment (POS2)
  - Open vent and recirculation valves
  - Cooling with heat conduction through containment to reactor pool
  - Module stable in this state indefinitely
- Disconnection (POS3)
  - Disconnect piping and power
  - Connect to reactor building crane

Refueling Procedure

- Transport (POS4)
  - Transport to refueling area with reactor building crane
- Disassembly (POS5)
  - Open containment flange and reactor vessel flange
  - Remove upper vessels and reactor vessel internals
- Refueling and core shuffle (POS6)
- Reassembly (POS5), Transport (POS4), Reconnection (POS3)
- Restart (POS7)

Total refueling outage time: ~11 days

Refueling Advantages

NuScale has:

- the ability to operate 11 modules while any module is being refueled
- permanent refueling personnel, reducing contract staff
- NO draindown events
- NO mid-loop operation
- NO active cooling after containment is flooded

Refueling Challenges

- Refueling takes place under ~60 feet of water
- No control over temperature and pressure in reactor after disconnection
- Transport of module while fueled

Initiating Events for LP/SD PRA

- For initiating events from Level 1 PRA that apply to one or more POSs, the frequency is adjusted to account for the duration of the POS.
- Initiating events involving systems not in service are screened
  - Example: when cooling through containment (POS2), secondary cooling systems are isolated. Initiating events including loss of feedwater, main steam line break, and steam generator tube breaks are not included for this POS.

Initiating Event Frequency

$$f_{\mathrm{LP}} = \frac{f_{\mathrm{FP}}}{\mathrm{CF}} \times f_{\mathrm{POS}} \frac{d}{8760}$$

Where

- f_{LP} low power frequency, per calendar year
- f_{FP} full power frequency, per reactor critical year
- CF module capacity factor, dimensionless*
- f_{POS} frequency of POS, per calendar year
- d = duration of POS, hours

* CF used as 0.844, industry average from 2012

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Initiating Event Frequency

Initiating Event	POS	Duration (hours)	f_{FP} (per rcry)*	f_{POS} (per year)	f_{LP} (per year)
LOCA outside containment	1	10	3.67E-4	2.5	1.24E-6
Loss of secondary cooling	1	10	1.28E-1	2.5	4.33E-4
Loss of offsite power	1	10	6.14E-2	2.5	2.08E-4
LOCA outside containment	2	15	3.67E-4	1.5	1.12E-6
Loss of secondary cooling	2	15	1.28E-1	1.5	N/A
Loss of offsite power	2	15	6.14E-2	1.5	N/A
LOCA outside containment	7	20	3.67E-4	2.5	2.48E-6
Loss of secondary cooling	7	20	1.28E-1	2.5	8.65E-4
Loss of offsite power	7	20	6.14E-2	2.5	4.15E-4

* From generic operating experience data (NuScale values proprietary)

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Reactor building crane

- Single-failure-proof (SFP) crane, as required for all critical load lifts
- Meets or exceeds all nuclear regulations
  - 10CFR50 Appendix B, NQA-1, NUREG-0554 and NUREG-0612, NOG
- Dedicated lifting device to interface with module lifting points
  - No temporary or moveable rigging
- Success criteria and accident sequences for a crane failure event are being developed

Crane Failure Probability

- Operating experience data for very heavy loads (greater than 30 tons) in NUREG-1774
  - 9 failure events (load slip or load drop) in estimated 54,000 lifts

$$\lambda = \frac{9}{54{,}000} = 1.67 \times 10^{-4} \text{ per lift}$$

  - Not all events are directly relevant: non-SFP crane, non-critical load
- Weighting factors adjust failure events for relevance
  - Identify consequence, cause, and crane for each failure to take credit for SFP and dedicated rigging device
  - Product of three factors is percent relevance for each event

Crane Failure Probability

Consequence	Factor	Cause	Factor	Crane	Factor
Slip	0.5	Human	1.0	SFP	1.0
Drop	1.0	Mechanical	0.1	Non-SFP	0.1
		Rigging	0.1		

- Most relevant event would be load drop with SFP crane caused by human error: 1.0 equivalent failures
- Least relevant is a load slip with non-SFP caused by rigging or mechanical failure: 0.005 equivalent failures

Crane Failure Probability

Weighting system applied to failure events

Date	Plant	Consequence	Cause	Crane	Equiv. Failures
11/1985	St. Lucie 1	Slip	Mechanical	Non-SFP	0.005
4/1990	Fort Calhoun	Slip	Rigging	SFP	0.050
9/1993	Arkansas 1	Slip	Human	SFP	0.500
12/1997	Byron	Slip	Human	Non-SFP	0.050
10/1999	Comanche Peak	Slip	Mechanical	Non-SFP	0.005
11/1999	Crystal River 3	Slip	Rigging	SFP	0.050
12/1997	Byron	Drop	Human	Non-SFP	0.100
5/2001	San Onofre	Drop	Rigging	Non-SFP	0.010
6/2001	Turkey Point 4	Drop	Rigging	Non-SFP	0.010
				Total	0.780

$$\lambda = \frac{0.780}{54,000} = 1.44 \times 10^{-5} \text{per lift}$$

Crane Failure Uncertainty

- Uncertainty sampling was performed with OpenBUGS
  - Assumed lognormal distribution with error factor of 10

Mean	Standard Deviation	5% Value	Median	95% Value
1.437E-5	3.411E-5	5.368E-7	5.383E-6	5.350E-5

Conclusions

- Internal events are not a major contributor to risk at a NuScale plant
  - Passive cooling with large quantities of water nearby
- Failure of reactor building crane deserves closer scrutiny
  - More detailed and design-specific analysis of NuScale's crane to determine failure probability
  - Deterministic thermal-hydraulic analyses to determine consequences of a crane failure

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