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Enclosure:

"ACRS Subcommittee Presentation: NuScale FSAR Topic – Resolution of Chapter 15 Phase 2 Open Items," PM-0220-69062, Revision 0

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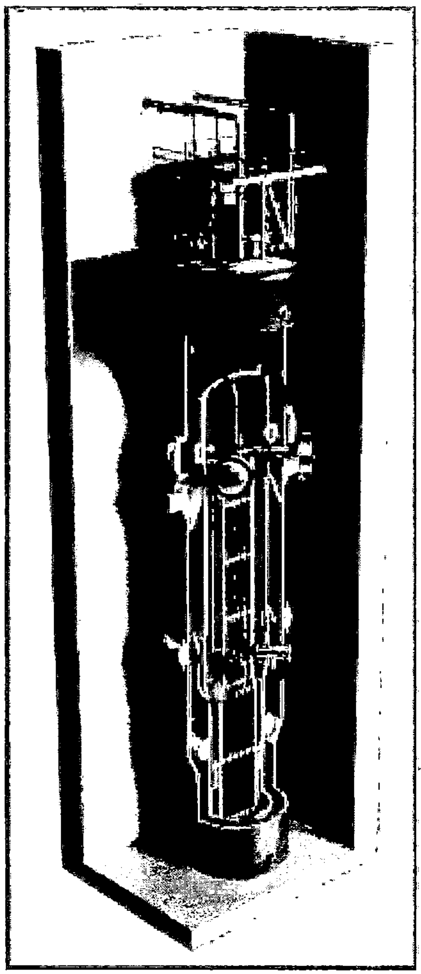
ACRS Subcommittee Presentation



NuScale FSAR Topic

Resolution of Chapter 15 Phase 2 Open Items

March 2-4, 2020

PM-0220-69062
Revision: 0

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Agenda

- NuScale Design objectives and long term shutdown implications
- FSAR 15.0.6 Return to power analysis
- Boron transport
  - Design basis ECCS cooling
  - Design basis DHRS cooling
  - Beyond design basis conditions
- Changes from FSAR Rev. 2 to FSAR Rev. 4
  - Incorporates NRELAP5 v1.4
  - Minor module model update
  - DHRS actuation logic changes
  - ECCS changes
- Overall changes in Chapter 15 analysis results FSAR Rev. 2 to FSAR Rev. 4

Completely Passive Design Basis

- Fundamental design characteristics enable passive design objectives
  - Low core power and large RCS volume
  - Simple decay heat removal systems
- Actual plant capabilities for heat removal and reactivity control are much more reliable than existing fleet
  - Fail safe valve positions activate passive heat removal
  - NPM can reach cold shutdown using CRAs alone
    - Accommodates reactivity insertion from complete Xenon burnout
    - NuScale PDC-27 commitment for all future core designs
- No active safety systems – no requirement for safety related power and or safety operator mitigation actions

Traditional Analysis Limitations

- Design basis events are analyzed considering highest worth CRA fails to insert.
  - 1 of 16 as opposed to 1 of 53 (AP1000)
  - Small core (larger leakage) leads to proportionally more excess reactivity and larger CRA worth for exterior assemblies.
  - Application of WRSO is uniquely penalizing for the NuScale design
- Origins of GDCs indicate no intention of application of stuck rod margin for the purposes of long term hold down.
  - Redundant system intended to be used to compensate for Xenon burnout (GDC 26 and 27).
  - Not required for the NuScale design due to CRA worth and natural Boron redistribution phenomena during extended ECCS operation

Boron Addition Considerations

- Mechanisms of ECCS cooling result in natural Boron accumulation in the core region
  - Same phenomena as typical PWR post LOCA Boron accumulation
  - Lack of continuous boron source supports sufficient margin to precipitation limits
  - Boron accumulation phenomena enhances long term shutdown margin, except late in cycle
- Consequences of late in cycle loss of SDM at low temperatures and power levels due to very slow Xenon burnout are not a safety concern.
  - GDC-27 exemption request
- NuScale conclusion: An active or passive safety Boron addition system does not make the design safer.

Principle Design Criteria 27

- DCA includes an exemption request from GDC-27
  - NPM design does align with precedent based compliance with GDC-27 due to lack of second safety reactivity control system
- Principle Design Criteria 27
  - Passive reactor GDC-27 equivalent
  - Ensures the safety related reactivity control system is designed to achieve and maintain subcritical core
  - Ensures fuel integrity for an extended overcooling in combination with a partial failure of reactivity system (stuck rod)

PDC-27 Clarification

- RAI-9498 Q# 15-9S1
  - Revised PDC language consistent with Staff interpretation of acceptable consequences for the return to power condition.
  - FSAR was updated committing to SAFDLs acceptance criteria for all DBEs.
  - Ensures an accident with fuel failure does not precede a return to power where additional source term would need to be analyzed.

The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted.

~~Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods, provided the specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.~~

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Compliance with PDC-27

- Immediate shutdown is sufficient to protect RCPB and SAFDLs with margin for the worst rod stuck out of the core
- Cold shutdown is achieved with all control rods fully inserted
- Loss of Shutdown Margin Consequences Benign
  - Evaluated with single highest worth control rod fully withdrawn
  - Critical power level does not challenge DHRS or ECCS heat removal or SAFDLs
- Probability of the combination of conditions that results in a loss of shutdown return to power with a single rod stuck out of the core is small

Return to Power Mechanisms

- Moderator overcooling
  - ECCS and DHRS designed to removed decay and residual heat
  - Under cold conditions DHRS or ECCS can cause a fairly rapid temperature decrease and increased moderation
- Fission product decay
  - Xenon decay causes a slow post shutdown reactivity insertion
- Boron redistribution
  - Boiling/condensing systems cause boron redistribution
    - Boric acid is not readily volatilized to the vapor phase and would be expected to recondense
    - Results in increasing concentration in boiling region and decreasing concentration in condensing region
  - Conclusion: Boron redistribution during extended ECCS operation increases SDM in the core (neglected in OCRP analysis)

Loss of Shutdown Margin

- FSAR 15.0.6 evaluation of return to power conditions
- Updated method described in FSAR 15.0.6
  - Statepoint analysis with SIMULATE5
  - NRELAP5 quasi-steady analysis
  - Critical power level at overlap
  - SIMULATE5 uncertainties accounted for
  - Control rod ejection with additional stuck rod only analyzed for short-term response
- Updated results presented in FSAR 15.0.6
  - Recriticality precluded for DHRS cooling with riser uncovery
  - DHRS cooling with covered riser non-limiting
  - ECCS cooling limiting

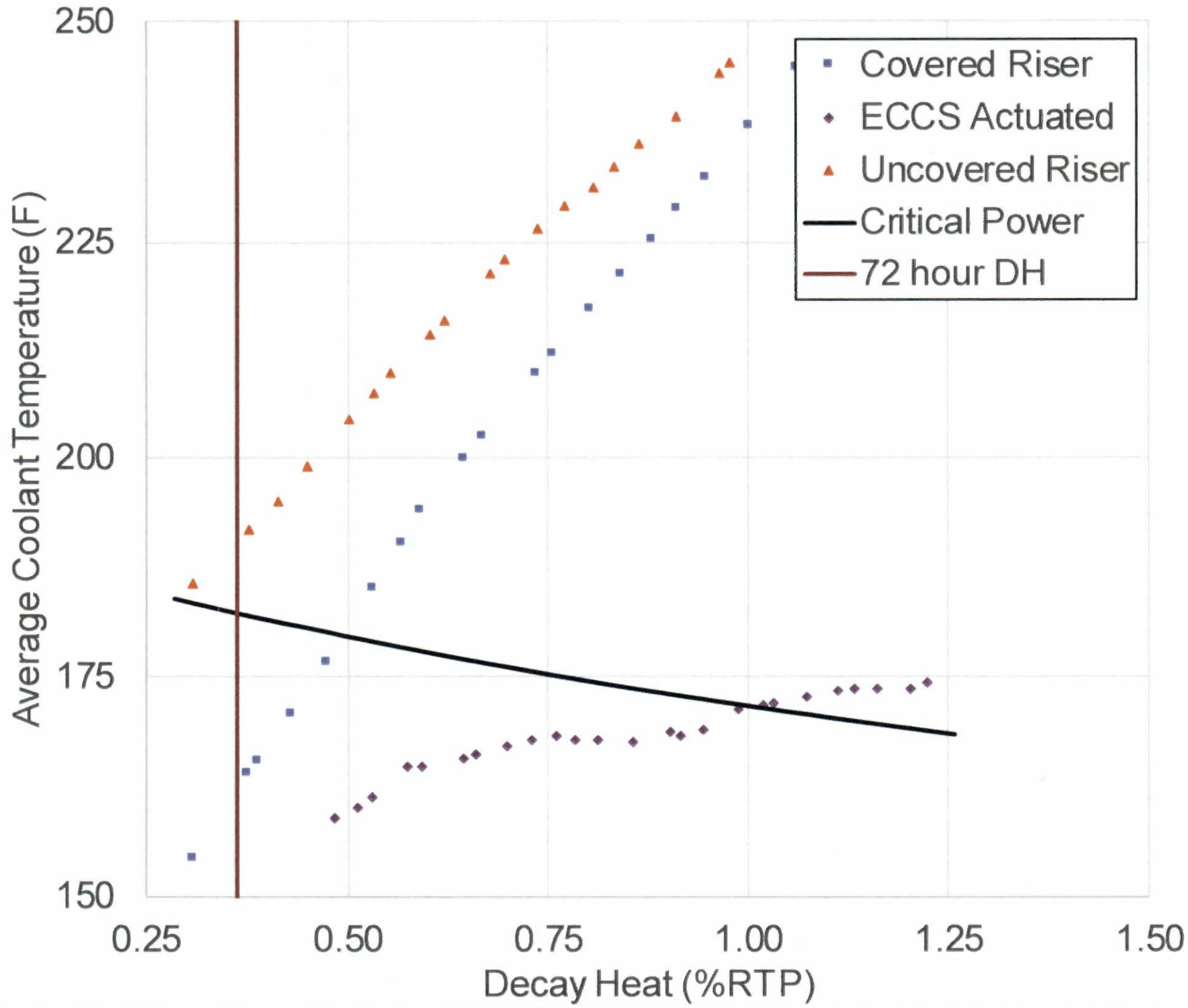
Loss of SDM Evaluation

- Average core temperature determined with the NRELAP5 state-point method described in LTC LTR
  - Performed for spectrum of initial conditions and cooling modes
- Critical power level determined using the SIMULATE5 core model with WRSO
  - Performed for a spectrum of boundary conditions (pressure, temperature, flow)
- CHF is evaluated using the zero flow CHF correlation described in the LOCA LTR
  - Margin is reported to the appropriate analytical limit also described in the LOCA LTR

Loss of SDM Evaluation

- Limiting Initial Conditions
  - Minimized Boron (Hot Full Power, Eq. Xe, EOC Core)
  - Maximized Cooling (Max pool level, min pool temp, biased high SG & DHRS heat transfer coefficient, max ECCS capacity)
- Additional Penalties for MCHFR Evaluation
  - Reactivity bias applied to SIMULATE5 to account for methodology uncertainty (Increases critical power level)
  - Conservative local peaking factor applied to core heat flux
  - Dynamic return to power factor of 2.0 applied

Equilibrium Power Results



Results – Return to Power Analysis

- ECCS cooling most limiting with equilibrium power limited to 1-2% RTP.
- Core temperature must be <200°F for recriticality
- Increased pool temperature decreases the magnitude of the return to power, with 140°F precluding a recriticality
- Earliest recriticality determined to occur approximately 40 hours post-scram
- MCHFR for most limiting results non-limiting relative to other events
- Other AOO acceptance criteria met
- Other SAFDLs demonstrated with OCRP conditions bounded by existing analyses developed for the DCA

Control Rod Ejection, GDC 28, and PDC-27

- RAI 9647/q15-29
- Return to power analysis
  - Performed to demonstrate compliance with PDC-27
  - Is bounding with respect to long-term holddown with a single control rod ejected
- Control rod ejection analysis
  - Performed to demonstrate compliance with GDC 28
  - GDC 28 imposes core design limits distinct from reactivity control system capabilities addressed by GDC 27
  - REA is non-mechanistically postulated for the purpose of evaluating the consequences of a limiting reactivity insertion event as required by GDC 28
  - Extension of PDC 27 to REA not warranted by unique design considerations

Control Rod Ejection, GDC 28, and PDC-27

- GDC 27 has historically not been applied to a rod ejection accident
  - GDC 27 is not cited in SRP 15.4.8 or RG 1.77
  - Application of GDC 27 not required in other approved rod ejection methodologies
  - Extension of PDC 27 to NuScale REA is not warranted by unique design considerations
- Control rod ejection is non-mechanistically assumed in NuScale design
  - FSAR 3.9.3.1.2: CRDM pressure housing is a Class 1 appurtenance per ASME BPVC, Section III, NCA-1271
    - As with other Class 1 vessels and appurtenances, gross failure is not considered credible
  - CRDM nozzles are integral parts of reactor pressure vessel closure head forging
  - CRDM nozzle to Alloy 690 safe-end welds are full penetration butt welds
  - Estimated likelihood of rod ejection, failure of a control rod to insert, failure of boron addition system that could result in return to power is ~ 1E-10 per year

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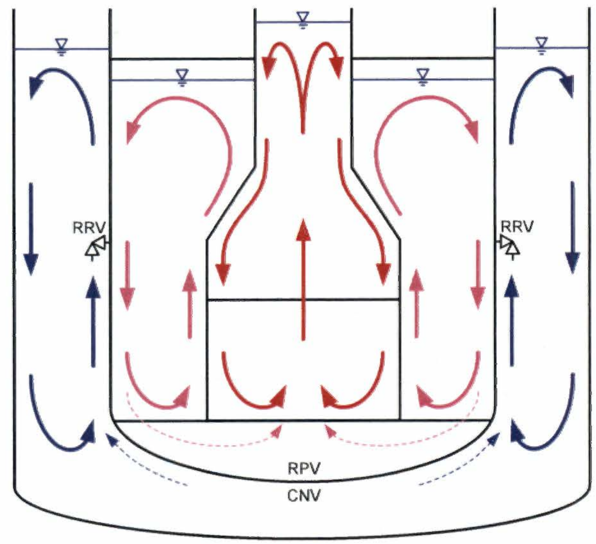
ECCS Boron Transport – Context

Context for ECCS boron transport analysis:

- As boron accumulates in the core/riser region, boron concentration in the CNV and DC decreases
  - Boron precipitation analysis performed as part of ECCS long term cooling analysis
- Boron dilution analysis performed to:
  - Evaluate potential for lower boron concentration fluid in core or near core inlet
  - Confirm appropriate scope of return to power analysis by demonstrating that core region concentration remains above initial concentration
  - Response to RAI 8930



Boron transport governed by:

- boiling in the core
- condensation in the containment vessel

ECCS Boron Transport – Method

- Method summary for dilution analysis:
  - LTC PIRT high ranked phenomena affecting boron transport evaluated
  - Control volume approach to analyze transport between regions
  - NRELAP5 used to provide volume fluid masses, flow rates as input for boron transport calculation
    - Volatility, entrainment calculated separately
  - Boron transport calculation performed separate from NRELAP5
  - Cotransport out of RCS hot region
  - Demonstrate that RCS hot region concentration remains above initial concentration
- Key areas of NRC review:
  - Treatment of boron volatility
  - Mixing
- CR opened to evaluate scope of scenarios considered
- Additional discussion in closed session
  - conservatively model transport between regions:
    Boron distribution factors applied to minimize boron transport in, maximize boron

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ECCS Boron Transport – Results

- Boron transport evaluated during ECCS cooling
  - Results summarized in RAI 8930 show core boron concentration remains above initial concentration
    - No net core boron dilution is expected even with biased transport assumptions
    - More realistic analysis of boron transport indicates boron concentration in RCS core region is 2-3 times the initial concentration at 72 hours. Core boron concentration remains above initial concentration for at least 7 days.
- Realistically, long term, high boron concentration expected in RCS hot region, with low concentration in RCS cold region, containment
- Recovering the riser and establishing Mode 3 conditions will take multiple deliberate operator actions following appropriate procedures
- Procedures are developed on a site-specific basis (COL commitments 13.5-2 and 13.5-7.)

Comparison of ECCS and DHRS Conditions

ECCS

- Cooling established by boiling/condensing mode
- Will tend to redistribute boron into the RCS hot region and out of RCS cold, CNV regions
- RCS level well below top of the riser
- Recovering the riser and establishing Mode 3 conditions will take multiple deliberate operator actions following appropriate procedures

DHRS

- Riser uncovery sustained by significant convective heat transfer through the riser wall
  - RCS level significantly higher than during ECCS operation – top of steam generator relatively limited condensing potential compared to CNV
  - When riser remains covered, primary side natural circulation maintained and boron distribution should remain close to initial well-mixed condition
- Minimal drivers to redistribute boron in RCS compared to ECCS cooling
- Recovering the riser and establishing Mode 3 conditions will take multiple deliberate operator actions following appropriate procedures

Additional discussion of extended DHRS cooling in closed session.

PRA Considerations for ATWS

- ATWS is typically postulated due to common cause failure of I&C systems to generate a reactor trip signal
  - Events would easily be resolved by removal of power from CRDMs
- Focus of ATWS analysis is generally limited to short term RPV/Secondary pressurization analysis.
  - Short term effects are not challenging due to small core power, large RCS volume, and large RSV capacity in NuScale module design.
- Analysis of long term effects of an ATWS are less meaningful for risk insight due to low combined likelihood of mechanical CCF of 16 CRAs
- In the NuScale PRA, ATWS frequency is conservatively based on mechanical CCF of 3 CRAs.
  - N-3 transients will immediately shutdown similar to design basis events where WRSO is considered (Ch. 15)
- PRA supporting T/H calculations evaluate failure of all CRAs to insert for 72hr coping period.

PRA Insights for ATWS

- LOCA/IORV Events
  - Short term response
    - BOC/EOC – Break flow cause sufficient depressurization and void feedback to make core subcritical.
  - Long term response (riser uncovery)
    - BOC – Boron accumulation in core leads to complete shutdown (no power return)
    - EOC – Equilibrium power level achieved due to balance ECCS cooling with reactivity feedback mechanisms from void, temperature, and Xenon.
- nonLOCA Events
  - Short term response
    - BOC – MTC is less negative but still sufficient to reduce reactor power to match DHRS heat removal.
    - EOC – Large negative MTC cause a quick stabilization of core power and DHRS heat removal.
  - Long term response (no riser uncovery)
    - BOC – Equilibrium power level achieved due to balance DHRS cooling with reactivity feedback mechanisms from temperature, Xenon, and Boron accumulation. RSV venting and subsequent boron concentrating identified as important factor in overall reactivity balance.
    - EOC – Equilibrium power level achieved due to balance DHRS cooling with reactivity feedback mechanisms from temperature, and Xenon.
- T/H results support the conclusion that no operator action is required to mitigate event and prevent core damage for short or long term ATWS mitigation.

Conclusions

- Inherent design characteristics provide ample safety
  - Low core power, large RCS inventory, small high pressure containment, and large ultimate heat sink
- Compliance with intent of GDCs is demonstrated for reactivity control systems
  - Conservative analysis of the low probability return to power condition demonstrates safety margin
- Boron redistribution is evaluated and demonstrated to not be a safety concern
  - Naturally accumulating boron in the core adds to shutdown margin for design basis event and severe accidents.

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Ch 15 Changes FSAR Rev. 2 to Rev. 4

- Results from FSAR Rev. 2 presented to ACRS in June, July 2019 in subcommittee and full committee meetings for Chapter 15
- Changes in FSAR Rev. 3 include
  - Update from NRELAP5 v1.3 to v1.4
  - Updated NRELAP5 base model input
  - More conservative core design input in some cases
  - DHRS actuation signal changes, addition of secondary side isolation signal
  - ECCS actuation signal changes
- Changes in FSAR Rev. 4 include
  - ECCS IAB threshold/release pressure changes

NRELAP5 v1.4

- Modifications made from v1.3 to v1.4 were due to routine code maintenance
- 26 specific code Fixes (documented in error reports) with most notable being:
  - Condensation correlation error corrections (< 2 psi increase in CNV pressure calculations)
  - Correction to choking model quality factor (little to no impact)
  - Updated Windows executable to 64-bit version (not used for production calculations)
- 5 new Features – None of which impact DCA calculations
  - Added proprietary classifications marking to source files
  - Expanded number of elements allowed in water property file (no water property file update)
  - Interpolation update for CHF correlation not used in DCA calculations
  - Added warning message to users if mass error stop (1%) is disabled
  - Removal of Developmental Options from user access

NRELAP5 Base Model

- Revision 0 released 12/2015 (DCA submittal 12/2016)
- Revision 1 released 8/2017
  - Updates for design consistency
    - Minor geometry changes based on drawing updates
    - Minor RCS flow loss updates (changes in best estimate values)
  - Updates for analysis consistency and ease of downstream use
    - Minor nodalization changes to match LOCA model
    - Added passive heat structures defined in LOCA model
  - Other changes
    - Change from elevation based to volume based calculation of collapsed liquid level
    - Error correction when specifying lower CNV material (had been previously corrected in impacted analysis calculations)
- Revision 2 released 01/2019 (FSAR Rev. 3 submittal 8/2019)
  - Removed ECCS actuation on RCS riser level signal
  - Minor RCS flow loss updates
  - Minor geometry error corrections

Neutronics Range Changes

- For FSAR Rev 3, analyzed more bounding ranges of core design input, including 2 additional depletions for high and low flow rates.
- Different parameter ranges included:
  - Most negative DTC (from -2.25 pcm/°F to -2.5 pcm/°F)
  - Delayed neutron fraction (β_{eff})
  - Augmentation factors for asymmetric reactivity events
- No changes to MTC range

DHRS Actuation Changes

- Summary of change:
  - Add secondary side isolation actuation for range of signals that indicate upset in normal secondary side cooling conditions
  - DHRS actuation limited to subset of signals indicating insufficient secondary side cooling
  - DHRS actuated following secondary side isolation
- Purpose of change: Support expected plant startup progressions
- Effect of change on transient analyses:
  - Heatup events – No change to expected DHRS actuations on high pressurizer pressure or high RCS hot temperature
  - Cooldown events – Secondary side isolation may be actuated first; DHRS actuated afterwards on high steam pressure
  - Reactivity events, inventory increase, inventory decrease events not significantly impacted

DHRS Actuation Changes

FSAR Rev. 2	FSAR Rev. 3, Rev. 4
DHRS actuation on: - High pressurizer pressure - High RCS hot temperature - High CNV pressure - Low pressurizer pressure - Low-low pressurizer level - Low main steam pressure - Low-low main steam pressure - High main steam pressure - High main steam superheat - Low main steam superheat - High under bioshield temperature - Low AC voltage	SSI actuation on: - High pressurizer pressure - High RCS hot temperature - High CNV pressure - Low-low pressurizer pressure - Low-low pressurizer level - Low main steam pressure - Low-low main steam pressure - High main steam pressure - High main steam superheat - Low main steam superheat - High under bioshield temperature - Low AC voltage DHRS actuation on: - High pressurizer pressure - High RCS hot temperature - High main steam pressure - Low AC voltage

DHRS Actuation Changes

- Example impact on cooldown event:
  Decrease in FW Temperature MCHFR Case

Event	Time (sec) FSAR Rev. 2	Time (sec) FSAR Rev. 3, 4
Feedwater temperature begins to decrease	0	0
Feedwater temperature reaches 100°F	160	86
High RCS hot temperature limit reached	125	184
High reactor power limit reached	131	187
Reactor trip (high reactor power)	133	189
DHRS actuation (high RCS hot temp)	133	192
SSI actuation (high RCS hot temp)	n/a	192

Limiting case occurs where high power, high RCS hot temperature occurs ~ same time

DHRS Actuation Changes

- Example impact on cooldown event:
  Increase in Steam Flow Rate MCHFR Case

Event	Time (sec) FSAR Rev. 2	Time (sec) FSAR Rev. 3, 4
Steam flow begins to increase	0	0
High RCS hot leg temperature reached	60	n/a
High reactor power limit reached	n/a	63
Reactor trip	68	65
Low pressurizer pressure limit reached	n/a	123
SSI actuation (low pressurizer pressure)	n/a	125
High steam pressure	n/a	1692
DHRS actuation	68	1697

Maximum power in both cases ~ 200 MW
Limiting case occurs where high power, high RCS hot temperature occurs ~ same time

DHRS Actuation Changes

- Example impact to heatup event:
  FWLB Limiting DHRS Case

Event	**Time (sec) FSAR Rev. 2**	**Time (sec) FSAR Rev. 3, 4**
Large FW line break inside CNV	0	0
High CNV pressure limit reached	1	1
RTS actuated (high CNV pressure)	3	3
Secondary system isolation actuated (high CNV pressure)	n/a	3
High pressurizer pressure limit reached	*{ does not cause additional actuations }*	7
DHRS actuation	3 (high CNV pressure)	9 (high PZR pressure)
RSV lift point reached	25	25
DHRS actuation valves open	33	39

DHRS Actuation Changes

- Example impact on reactivity event:
  Uncontrolled bank withdrawal at power – MCHFR case

Event	**Time (sec) FSAR Rev. 2**	**Time (sec) FSAR Rev. 3, 4**
CRA bank begins to withdraw	0	0
High RCS hot temperature limit reached	178	144
High pressurizer pressure limit reached	184	150
Reactor trip actuated	186	152
SSI actuated	n/a	152
DHRS actuated	186	152

Limiting case occurs where high power, high RCS hot temperature occurs ~ same time when different signal delays are accounted for

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ECCS Valve Operation

- Emergency core cooling system (ECCS) valves receive actuation demand on:
  - ECCS actuation signal on high CNV level, or
  - Loss of DC power to ECCS trip valves
- Inadvertent actuation block (IAB) feature prevents ECCS valve opening if the differential pressure between the reactor coolant system (RCS) and containment (CNV) is above the IAB threshold pressure
  - This feature prevents opening due to spurious signals or equipment failures at normal operating pressures but permits opening in loss-of-coolant accident (LOCA) conditions
- If IAB is actuated by ECCS demand at high differential pressure, IAB releases at lower pressure and then ECCS valves open

ECCS Actuation Changes

FSAR Rev 2

- ECCS actuated on
  - High CNV level (220-260 in)
  - Low RCS riser level (350-390 in)
  - Loss of DC power to valve actuators
- RCS riser level is post-accident monitoring Type B and Type C variable
  - 4 total divisions of RPV riser level
- IAB threshold/release
  - Threshold: Block if ECCS actuated above threshold pressure that is in the range of 1000-1200 psid
  - Release: If IAB blocks, release between 1000-1200 psid

FSAR Rev 4

- ECCS actuated on
  - High CNV level (264-300 in)
  - Loss of DC power to valve actuators
- RCS riser level is post-accident monitoring Type B and Type C variable
  - 4 total divisions of RPV riser level
- IAB threshold/release
  - Threshold: Block if ECCS actuated above 1300 psid; does not block below 900 psid
  - Release: If IAB blocks, release at 950 psid +/- 50 psi

ECCS Changes - Revised FSAR Analyses

- Impacted FSAR Sections
  - FSAR 6.2 Peak CNV Pressure
  - FSAR 15.6.5 Loss of Coolant Accidents
  - FSAR 15.6.6 Inadvertent Operation of ECCS
- Revised assumptions
  - Assumes all ECCS valves remain closed due to IAB block function above 1300 psid
  - Evaluated ECCS valves opening on IAB release pressure between 900 and 1000 psid
- Revised analysis results submitted in September 2019 and reviewed in NRC October audit in Corvallis
- DCA Revision 4, including revised FSAR analysis results, formally submitted January 2020

ECCS Changes - Updated Analysis Results

Event / Acceptance Criteria	DCA Rev 3 Results	Updated DCA Rev 4 Results	Comments
Peak CNV Pressure (RRV Opening) CNV Design Pressure - 1050 psia	986 psia	994 psia	Change in peak pressure due to staggered IAB release (2nd RRV at 1000 psid, RVVs at 900 psid)
LOCA - Minimum Water Level Above Top of Active Fuel	1.7 ft	1.5 ft	Change due to lower IAB minimum release pressure 900 psid
Inadvertent ECCS valve opening – MCHFR limit 1.13	1.41	1.32	Change due to model revisions not IAB threshold change

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Conclusions – ECCS Valve Changes

- CNV peak pressure results slightly more limiting (8 psi) due to explicit evaluation of ECCS valves opening at different IAB release pressures
- LOCA minimum water level above fuel results slightly more limiting (~0.2 feet difference) due to lower minimum IAB release pressure of 900 psid
- Inadvertent ECCS valve opening MCHFR slightly more limiting due to evaluation of error corrections and more bounding model input, not from IAB change
- All updated event results demonstrated margin to acceptance criteria

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FSAR 15 Limiting Transient Results

Parameter	Event	Acceptance Criterion	Limiting Result FSAR Rev. 2	Limiting Result FSAR Rev. 4
Maximum RCS Pressure	Several	< 2315 psia (110% P_{design}) or < 2520 psia (120% P_{design})	~ 2170 psia	~ 2170 psia
Maximum SG Pressure	Inadvertent Operation DHRS	< 2315 psia (110% P_{design})	1582 psia	1592 psia
	SG tube failure	< 2520 psia (120% P_{design})	1806 psia	1871 psia
MCHFR	Single rod withdrawal	> 1.284	1.614	1.375
	Inadvertent Opening RRV	> 1.13	1.41	1.32
	LOCA	> 1.29	1.796	1.74
Level above top of core	LOCA	> 0 ft	1.5 ft	1.5 ft

Conclusions

- Revised return to power analysis shows ECCS cooling conditions result in equilibrium power at 1-2% RTP
- ECCS boron transport analysis demonstrates that core boron concentration remains higher than initial concentration
- Changes incorporated into FSAR Revision 3:
  - Several minor changes in NRELAP5 code, NPM plant base model
  - DHRS, ECCS actuation changes
- ECCS IAB changes incorporated into FSAR Revision 4
- FSAR Ch 15 limiting transient results consistent between FSAR Rev. 2 and Rev. 4
- FSAR Ch 15 analysis results demonstrate margin to acceptance criteria

Acronyms

AOO – Anticipated Operational Occurrences

CHF – Critical Heat Flux

CNV – Containment Vessel

COL – Combined License

COLR – Core Operating Limits Report

CRDM – Control Rod Drive Mechanism

CVCS – Chemical and Volume Control System

DHRS – Decay Heat Removal System

DTC – Doppler Temperature Coefficient

ECCS – Emergency Core Cooling System

EOC – End of Cycle

GDC – General Design Criteria

IAB – Inadvertent Actuation Block

LCO – Limiting Condition for Operation

LOCA – Lossof Coolant Accident

MCHFR – Minimum Critical Heat Flux Ratio

MTC – Moderator Temperature Coefficient

NPM – NuScale Power Module

OCRP – Overcooling Return to Power

PDC – Plant Design Criteria

PIRT – Phenomena Identification and Ranking Table

RCPB – Reactor Coolant Pressure Boundary

RCS – Reactor Coolant System

REA – Rod Ejection Accident

SAFDL – Specified Acceptable Fuel Design Limits

SDM – Shutdown Margin

WRSO – Worst Rod Stuck Out

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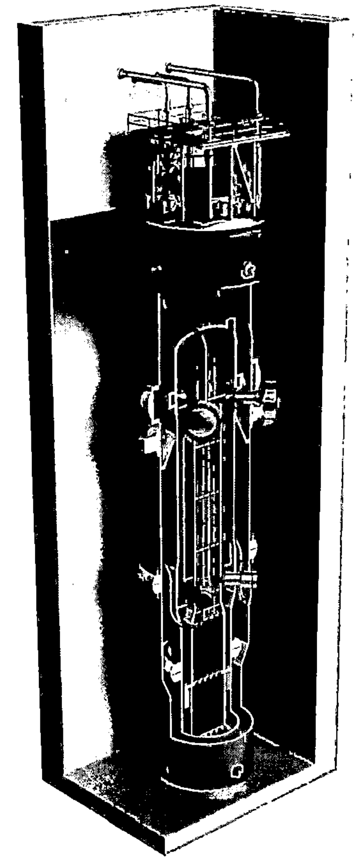

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Additional Information

MODE Definitions

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	INDICATED REACTOR COOLANT TEMPERATURES (°F)
1	Operations	≥ 0.99	All ≥ 420
2	Hot Shutdown	< 0.99	Any ≥ 420
3	Safe Shutdown [(a)]	< 0.99	All < 420
4	Transition [(b)(c)]	< 0.95	N/A
5	Refueling [(d)]	N/A	N/A

(a) Any CRA capable of withdrawal, any CVCS or CFDS connection to the module not isolated.

(b) All CRAs incapable of withdrawal, CVCS and CFDS connections to the module isolated, and all reactor vent valves electrically isolated.

(c) All reactor vessel flange bolts fully tensioned.

(d) One or more reactor vessel flange bolts less than fully tensioned.

SG Modeling

- NRELAP5 validation and NPM sensitivity calculations:
  - SIET-TF1 secondary side heat transfer and pressure drop
  - SIET-TF2 primary and secondary side heat transfer and pressure drop
  - NPM sensitivity calculations for steam generator modeling
    - Axial nodalization, heat transfer variation
- Key conclusions for NPM non-LOCA analysis:
  - Steam generator heat transfer variation affects steady-state conditions:
    - Steam generator initial secondary side inventory, steam temperature
    - RCS initial flow and temperature conditions due to influence of secondary side conditions on natural circulation driving head
  - Steam generator heat transfer impact on initial conditions affects which process condition is first reached that actuates reactor trip and/or other engineered safety systems
    - For events analyzing a spectrum of change, changes in steam generator secondary initial conditions will tend to shift the magnitude of the limiting change but not otherwise change the type of event progression.
      Example: the limiting temperature decrease for the decrease in FW temperature event analysis
  - Steam generator heat transfer does not directly affect margin to MCHFR
    - RCS steady-state flow rate is biased low for MCHFR cases
    - After reactor trip, power decrease much faster than flow rate decrease, even considering variation in steam generator heat transfer

Analysis Results – FSAR Rev. 2; Rev 4

15.1 Increase in heat removal by secondary system

Sec.	Event (1) (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		MCHFR (> limit: 1.284)	
15.1.1	Decrease in feedwater temperature	1959	2005	1432	1541	1.921	1.847
15.1.2	Increase in feedwater flow	1936	2002	1424	1491	1.944	1.854
15.1.3	Increase in steam flow	2018	1981	1208	804	1.957	1.881
15.1.4	Inadvertent opening of steam generator relief or safety valve	NA	NA	NA	NA	NA	NA
15.1.5	Steam piping failures	2156	2081	1346	1495	1.861	1.866
15.1.6	Loss of containment vacuum/containment flooding (1)	1992	1937	1342	1426	2.761	2.66

(1) NuScale unique event

Significant margin to acceptance criteria for all events

Analysis Results – FSAR Rev. 2; Rev 4

15.2 Decrease in heat removal by secondary system

Sec.	Event (1) (Acceptance criteria)	Peak RCS Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		Peak SG Pressure (< 110% P_{design}: 2310 psia) (< 120% P_{design}: 2520 psia)		MCHFR (> limit: 1.284)	
15.2.1	Loss of external load	2158	2161	1474	1545	2.579	2.441
15.2.2	Turbine trip	2158	2161	1474	1545	2.579	2.441
15.2.3	Loss of condenser vacuum	2158	2161	1474	1545	2.579	2.441
15.2.4	Closure of main steam isolation valve	2160	2161	1481	1512	2.567	2.670
15.2.6	Loss of non-emergency AC to station auxiliaries	2162	2160	1361	1415	2.569	2.539
15.2.7	Loss of normal feedwater flow	**2165**	**2171**	1434	1528	2.569	2.426
15.2.8	Feedwater system pipe breaks	**2164**	**2164**	1328	1389	2.607	2.496
15.2.9	Inadvertent operation of the decay heat removal system (1)	2163	2161	**1582**	**1592**	2.489	2.67

(1) NuScale unique event

Significant margin to acceptance criteria for all events

Analysis Results – FSAR Rev. 2; Rev 4

15.4 Reactivity and Power Distribution Anomalies – focus on SAFDLs

Sec.	Event [(1)] (Acceptance criteria)	MCHFR (> limit: 1.284)		Fuel centerline (< Tmelt)		LHR (< 21.22 kW/ft)	
15.4.1	Uncontrolled control rod assembly withdrawal from subcritical or low power	>10	>10	890.8 F	1051.8F	NA	NA
15.4.2	Uncontrollled control rod assembly withdrawal at power	1.624	1.499	NA	NA	8.97 kW/ft	9.16 kW/ft
15.4.3	Control rod misalignment	2.509	1.437	NA	NA	7.10 kW/ft	8.39
15.4.3	Control rod withdrawal	**1.624**	**1.375**	NA	NA	7.84 kW/ft	8.29
15.4.3	Control rod drop	1.641	1.432	NA	NA	8.42 kW/ft	6.71
15.4.6	Inadvertent decrease in boron concentration in RCS	NA	NA	NA	NA	NA	NA
15.4.7	Inadvertent loading and operation of a fuel assembly in improper position	1.916	1.437	NA	NA	7.87 kW/ft	8.39
15.4.8	Spectrum of rod ejection accidents	2.477	1.838	2162 F	2345 F	NA	NA

Control rod withdrawal has limiting MCHFR for reactivity events

Analysis Results – FSAR Rev. 2; Rev 4

15.5 Increase in reactor coolant inventory

Sec.	Event [1] (Acceptance criteria)	Peak RCS Pressure ($\leq$ 110% P_{design}: 2310 psia)		Peak SG Pressure ($\leq$ 110% P_{design}: 2310 psia)		MCHFR ($\geq$ limit: 1.284)	
15.5.1	Chemical and volume control system malfunction	2130	2160	1418	1430	2.379	2.702

Significant margin to acceptance criteria

Analysis Results – FSAR Rev. 2; Rev 4

15.6 Decrease in reactor coolant inventory

<table>
<tr><th>Sec.</th><th>Event[1]
(Acceptance criteria)</th><th colspan="2">Peak RCS Pressure
(< 110% P_{design}: 2310 psia)
(< 120% P_{design}: 2520 psia)</th><th colspan="2">Peak SG Pressure
(< 110% P_{design}: 2310 psia)
(< 120% P_{design}: 2520 psia)</th><th colspan="2">MCHFR</th><th colspan="2">Additional</th></tr>
<tr><td>15.6.1</td><td>Inadvertent opening of reactor safety valve</td><td colspan="2">NA</td><td colspan="2">NA</td><td colspan="2">NA</td><td colspan="2">NA</td></tr>
<tr><td>15.6.2</td><td>Failure of small lines carrying primary coolant outside containment</td><td>2047</td><td>2067</td><td>1368</td><td>1473</td><td colspan="2">NA</td><td colspan="2">Note 2</td></tr>
<tr><td>15.6.3</td><td>Steam generator tube failure</td><td>2073</td><td>2158</td><td>1806</td><td>1871</td><td colspan="2">NA</td><td colspan="2">Note 2</td></tr>
<tr><td rowspan="2">15.6.5</td><td rowspan="2">Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary</td><td colspan="2" rowspan="2">NA</td><td colspan="2" rowspan="2">NA</td><td>1.796</td><td>1.74</td><td>1.5 ft</td><td>1.5 ft</td></tr>
<tr><td colspan="2">Acceptance criteria: > 1.29</td><td colspan="2">Minimum level above top of core</td></tr>
<tr><td rowspan="2">15.6.6</td><td rowspan="2">Inadvertent operation of emergency core cooling system[1]</td><td colspan="2" rowspan="2">NA</td><td colspan="2" rowspan="2">NA</td><td>Result: 1.41</td><td>Result: 1.32</td><td colspan="2" rowspan="2">NA</td></tr>
<tr><td colspan="2">Acceptance criteria: > 1.13</td></tr>
</table>

(1) NuScale unique event

(2) Mass release and iodine spiking time provided as input to radiological analyses

SG tube failure maximum secondary pressure remains below design pressure

Valve opening and LOCA events demonstrate margin to acceptance criteria