May 31, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report Section 3.2, "Classification of Structures, Systems, and Components," Section 3.9, "Mechanical Systems and Components," and Section 17.4, "Reliability Assurance Program," Related to the Decay Heat Removal System and Emergency Core Cooling System Actuation Logic

REFERENCES:

1. Letter from NuScale Power, LLC to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application, Revision 2," dated October 30, 2018 (ML18311A006)
2. Letter from NuScale Power, LLC to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report Related to the Decay Heat Removal System and Emergency Core Cooling System Actuation Logic, " dated April 15, 2019 (ML19105B292)
3. Letter from NuScale Power, LLC to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report Section 6.2 "Containment Systems", and Technical Report TR-0516-49084 "Containment Response Analysis Methodology" Related to the Decay Heat Removal System and Emergency Core Cooling System Actuation Logic, " dated May 8, 2019 (ML19129A278)

During a public teleconference with members of the NRC staff on February 19, 2019, NuScale Power, LLC (NuScale) discussed updates to Final Safety Analysis Report (FSAR) related to the decay heat removal system and emergency core cooling system actuation logic. The changes affect FSAR Section 3.2, "Classification of Structures, Systems, and Components," Section 3.9, "Mechanical Systems and Components," and Section 17.4, "Reliability Assurance Program." The Enclosure to this letter provides a mark-up of the FSAR pages incorporating revisions to these chapters, in redline/strikeout format. NuScale will include this change as part of a future revision to the NuScale Design Certification Application.

This letter makes no regulatory commitments or revisions to any existing regulatory commitments.

If you have any questions, please feel free to contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

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Enclosure: Changes to NuScale Final Safety Analysis Report Section 3.2, “Classification of Structures, Systems, and Components,” Section 3.9, “Mechanical Systems and Components,” and Section 17.4 “Reliability Assurance Program”

Enclosure:

Changes to NuScale Final Safety Analysis Report Section 3.2, “Classification of Structures, Systems and Components,” Section 3.9, “Mechanical Systems and Components,” and Section 17.4 “Reliability Assurance Program”

RAI 03.02.01-2, RAI 03.02.01-3, RAI 03.02.02-2, RAI 03.02.02-6, RAI 03.08.02-14, RAI 03.08.04-1S1, RAI 03.09.02-64, RAI 05.04.02.01-6, RAI 06.02.04-2, RAI 09.01.03-1, RAI 09.02.02-1, RAI 09.02.04-1, RAI 09.02.04-1S1, RAI 09.02.05-1, RAI 09.02.06-1, RAI 09.02.07-4, RAI 09.02.07-5, RAI 09.02.09-2, RAI 09.03.04-5, RAI 09.04.02-1, RAI 09.04.02-1S1, RAI 10.04.07-2, RAI 11.02-1, RAI 12.02-32, RAI 15-17, RAI 15-17S1, RAI 19-14

Table 3.2-1: Classification of Structures, Systems, and Components

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
CNTS, Containment System							
All components (except as listed below)	RXB	A1	N/A	Q	None	B	I
• CVC Injection Check Valve • CVC Discharge Excess Flow Check Valve • CVC PZR Spray Check Valve	RXB	B2	None	AQ-S	None	C	I
• CVC Injection & Discharge Nozzles • CVC PZR Spray Nozzle • CVC PZR Spray CIV • CVC RPV High Point Degasification Nozzle • CVC RPV High Point Degasification CIV • RVV & RRV Trip/Reset # 1 & 2 Nozzles • RVV Trip 1 & 2/Reset #3 Nozzles • CVC Injection & Discharge CIVs	RXB	A1	N/A	Q	None	A	I
• NPM Lifting Lugs • Top Support Structure • Top Support Structure Diagonal Lifting Braces	RXB	B1	None	AQ-S	• ~~ANSI/ANS 57.1-1992~~ • ANSI N14.6~~ASME NOG-1~~ • NUREG-0612~~0554~~	N/A	I
• CNV Fasteners • Hydraulic skid • CNV Seismic Shear Lug • CNV CRDM Support Frame • Containment Pressure Transducer (Narrow Range) • Containment Water Level Sensors (Radar Transceiver) • SG 1 & 2 Steam Temperature Sensors (RTD)	RXB	A1	N/A	Q	None	N/A	I
CNTS CFDS Piping in containment	RXB	B2	None	AQ-S	None	B	II
Piping from (CES, CFDS, FWS, MSS, and RCCWS) CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	D	I
CVCS Piping from CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	C	I
CIV Close and Open Position Sensors: • CES, Inboard and Outboard • CFDS, Inboard and Outboard • CVCS, Inboard and Outboard PZR Spray Line • CVCS, Inboard and Outboard RCS Discharge • CVCS, Inboard and Outboard RCS Injection • CVCS, Inboard and Outboard RPV High-Point Degasification • ~~FWS, Supply to SGs and DHR HXs FWIV~~ • RCCWS, Inboard and Outboard Return and Supply • SGS, Steam Supply CIV/MSIVs and CIV/MSIV Bypasses	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
CIV Close and Open Position Indication • FWS, Supply to SGs and DHR HXs FWIV	RXB	A1	None	Q	None	N/A	I
Containment Pressure Transducer (Wide Range)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Containment Air Temperature (RTDs) • FW Temperature Transducers	RXB	B2	None	AQ-S	None	N/A	II
SGS, Steam Generator System							
• SG tubes • ~~Feedwater plenums~~Integral steam plenums • ~~Steam plenums~~Feedwater plenums	RXB	A1	N/A	Q	None	A	I
• SG tube supports • Upper and lower SG supports	RXB	A1	N/A	Q	None	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

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• Steam piping inside containment • Feedwater piping inside containment • Feedwater supply nozzles • Main steam supply nozzles • Thermal relief valves • Feedwater plenum access port covers • Steam plenum access port covers	RXB	A2	N/A	Q	None	B	I
Flow restrictors	RXB	A2	N/A	Q	None	N/A	I
RXC, Reactor Core System							
Fuel assembly (RXF)	RXB	A1	N/A	Q	None	N/A	I
Fuel Assembly Guide Tube	RXB	A2	N/A	Q	None	N/A	I
Incore Instrument Tube	RXB	B2	None	AQ-S	None	N/A	I
CRDS, Control Rod Drive System							
• Control Rod Drive Shafts • Control Rod Drive Latch Mechanism	RXB	A1	N/A	Q	None	N/A	I
CRDM Pressure Boundary (Latch Housing, Rod Travel Housing, Rod Travel Housing Plug)	RXB	A2	N/A	Q	None	A	I
CRDS Cooling Water Piping and Pressure Relief Valve	RXB	B2	None	AQ-S	None	B	II
Rod Position Indication (RPI) Coils	RXB	B2	None	AQ-S	None	N/A	I
• Control Rod Drive Coils • CRDM power cables from EDN breaker to MPS breaker • CRDM power cables from MPS breaker to CRDM Cabinets	RXB	B2	None	AQ-S	None	N/A	II
• CRDM Control Cabinet • CRDM Power & Rod Position Indication Cables • Rod Position Indication Cabinets (Train A/B)	RXB	B2	None	AQ	None	N/A	III
CRA, Control Rod Assembly							
All components	RXB	A2	N/A	Q	None	N/A	I
NSA, Neutron Source Assembly							
All components	RXB	B2	None	AQ-S	None	N/A	I
RCS, Reactor Coolant System							
All components (except as listed below)	RXB	A1	N/A	Q	None	A	I
• Reactor vessel internals (upper riser assembly (Note 7), lower riser assembly, core support assembly, flow diverter, and pressurizer spray nozzles)	RXB	A1	None	Q	None	N/A	I
• Reactor vessel internals upper riser bellows-lateral seismic restraining structure	RXB	A1	N/A	Q	None	N/A	I
• Reactor vessel internals upper riser bellows-vertical expansion structure • Narrow Range Pressurizer Pressure Elements • PZR/RPV Level Elements • Narrow Range RCS Hot Leg Temperature Elements • Wide Range RCS Hot Leg Temperature Elements • RCS Flow Transmitters (Ultrasonic)	RXB	B2	N/A	AQ-S	ASME B&PVC Section III Division 1 NG guidance	N/A	II
• Wide Range RCS Pressure Elements • Wide Range RCS Cold Leg Temperature Elements	RXB	A2	N/A	Q	None	N/A	I
Reactor Safety Valve Position Indicator	RXB	B2	None	AQ-S	Environmental Qualification Power from EDS	N/A	I
• PZR Control Cabinet • PZR Vapor Temperature Element • PZR heater power cabling from MPS breaker to PZR heaters • Pressurizer Liquid Temperature Element • Narrow Range RCS Cold Leg Temperature Element	RXB	B2	None	AQ-S	None	N/A	II
PZR heater power cabling from ELV breaker to MPS breaker	RXB	B2	None	None	None	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

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• Containment sampling system sample panel	RXB	B2	None	AQ	• ANSI N13.1 • RG 1.7 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III
• Primary sampling system sample cooler cooling water chillers	RXB	B2	None	AQ	Quality Group D	D	III
• Combined polisher effluents sample line isolation valve • Condensate polisher sample line isolation valves • Condensate pump discharge sample line isolation valve • Condenser hotwell sample line isolation valve • Feedwater sample line isolation valves • Main Steam bypass sample line isolation valves • Main steam sample line isolation valves	TGB	B2	None	None	None	D	III
MSS, Main Steam System							
• Start-up Isolation Valves • RXB Steam Traps	RXB	B2	None	AQ-S	None	D	I
• Secondary Main Steam Isolation Valves (Note 6) • Secondary Main Steam Isolation Bypass Valves (Note 6)	RXB	B2	None	AQ-S	• Technical Specification Surveillance for operability and in-service testing. • Valve Leak Detection	D	I
• Secondary Main Steam Isolation Bypass Valve Close and Open Position Indicators • Secondary Main Steam Isolation Valve Close and Open Position Indicators	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Auxiliary Steam Supply Valve • Auxiliary Steam Warm-up Valve • Main Steam Safety Valves • Main Steam Vent Valve • N2 Injection Isolation Valves • Steam Sample Panel Isolation Valve • TGB Steam Traps	TGB TGB Yard TGB RXB TGB TGB	B2	None	None	None	D	III
• Main Steam Flow Transmitters • Main Steam Radiation Monitors	RXB, TGB	B2	None	AQ	• IEEE 497-2002 with CORR 1 • ANSI N42.18-2004 (Radiation Monitors)	N/A	III
• Main Steam Pressure Transmitters • Main Steam Temperature Elements	RXB, TGB	B2	None	AQ	None	N/A	III
All other components	RXB, TGB	B2	None	None	None	N/A	III
FWS, Condensate and Feedwater System							
All components (except as listed below)	TGB, RXB	B2	None	None	None	N/A	III
Feedwater Regulating Valve A/B (Note 6)	RXB	B2	None	AQ-S	Technical Specification Surveillance for operability and in-service testing.	D	I
Feedwater Supply Check Valve (Note 6)	RXB	B2	None	AQ-S	Inservice Testing	D	I
Feedwater Regulating Valve Accumulators	RXB	B2	None	AQ	Technical Specification Surveillance for operability and in-service testing.	D	III
Feedwater Regulating Valve A/B Limit Switch	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Condensate Storage Tank (located adjacent to TCB) • Condensate Storage Tank Makeup Level Control Valve	Yard	B2	None	None	None	D	III
Steam Generator Differential Pressure Transmitter	RXB	B2	None	AQ	None	N/A	III

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

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Division I and Division II: • CTB Communication Module • Enable Nonsafety Control Switch • Hard-Wired Module • Scheduling and Bypass Modules • Safety Function Modules for CRV Post-filter Radiation Sensor • Safety Function Module for CRV Post-filter Radiation Sensor Trip/Bypass Switches	CRB	B2	None	AQ-S	RG 1.78	N/A	I
Division I and Division II: • CRV Outside Air Isolation Damper Equipment Interface Module • Manual Outside Air Isolation Actuation Switch • Safety Function Module for CRV Toxic Gas Sensor • Safety Function Module for CRV Toxic Gas Sensor Trip/Bypass Switch	CRB	B2	None	AQ-S	RG 1.78	N/A	I
Division I and Division II Maintenance Workstations	CRB	B2	None	AQ-S	None	N/A	II
RMS, Radiation Monitoring System							
RM system that monitors PAM B & C variables	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Radiation monitors that monitors Type E variables	RXB, TGB	B2	None	AQ	IEEE 497-2002 with CORR 1	N/A	III
Area airborne radiation monitors that monitors Type E Variable	CRB, RXB	B2	None	AQ	• IEEE 497-2002 with CORR 1 • ANSI/HPS N13.1-2011	N/A	III
Area airborne radiation monitors in: • Annex Building • Radioactive Waste Building • Reactor Building	ANB, RWB, RXB	B2	None	AQ	ANSI/HPS N13.1-2011	N/A	III
Radiation monitors in: • Annex Building • Control Building • Radioactive Waste Building • Reactor Building • Turbine Buildings	ANB, CRB, RWB, RXB, TGB	B2	None	AQ	None	N/A	III
RXB, Reactor Building							
Reactor Building (includes interior walls and floor forming UHS pool)	Yard	A1	N/A	Q	None	N/A	I
RBC, Reactor Building Cranes							
Reactor Building Crane	RXB	B1	None	AQ-S	ASME NOG-1	N/A	I
Module Lifting Adapter	RXB	B1	None	AQ-S	ANSI N14.6	N/A	N/A~~I~~
Traveling Jib Crane	RXB	B2	None	N/A	None	N/A	II
Wet Hoist	RXB	B2	None	AQ	ASME NOG-1	N/A	N/A~~III~~
RBCM, Reactor Building Components							
Over-Pressurization Vents (OPV)	RXB	A2	None	Q	None	C(d)	I
• UHS Pool Liner and Dry Dock Liner Dry Dock Gate support stainless steel plates at plate-to-liner weld locations	RXB	B2	None	AQ-S	ANSI/ANS 57.2-1983 with additions, clarifications, and exceptions of RG 1.13	N/A	I
Bioshield	RXB	B2	None	AQ-S	EQ requirements to GDC 4 and 23	N/A	II
Reactor Building Equipment Door	RXB	B2	None	AQ-S	None	N/A	II
Dry Dock Gate	RXB	B2	None	AQ-S	None	N/A	II
• Dry Dock Gate Closure instrumentation • Reactor Building Equipment Door Condition Instrumentation	RXB	B2	None	None	None	N/A	III
[[TGB, Turbine Generator Building]]							
Turbine Generator Building	Yard	B2	None	None	None	N/A	III
[[TBC, Turbine Building Cranes]]							
Turbine Building Cranes	TGB	B2	None	None	None	N/A	III
RWB, Radioactive Waste Building							
Radioactive Waste Building	Yard	B2	None	AQ	None	RW-IIa	II, RW-IIa

postulated combinations of normal operating conditions, AOOs, postulated pipe breaks, and seismic events in compliance with the requirements of GDC 14 and 15.

- 10 CFR 50, Appendix B, Section III, as it relates to quality of design control. Section 17.5 satisfies the requirements of 10 CFR 50, Appendix B, to ensure that SSC are designed, procured, fabricated, inspected, erected, and tested to standards commensurate with their contribution to plant safety.
- 10 CFR 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics. This requirement is met by including design-transient seismic events as part of the design basis for withstanding the effects of natural phenomena.

3.9.1.1 Design Transients

The design transients define thermal-hydraulic conditions (i.e., pressure, temperature, and flow) for the NPM. Bounding thermal-hydraulic design transients are defined for components of the RCPB. The number of cycles for each design transient is based on a plant life of 60 years. The transients are defined for safety-related equipment design purposes and are intended to provide a bounding representation of the NPM operation.

The following operating condition categories, as defined in the ASME BPVC, Section III (Reference 3.9-1), apply to Class 1, 2, and ~~2~~3 components, the containment vessel (CNV), supports, reactor vessel internals (RVI), piping, and valves inside and outside of containment up to the outermost containment isolation valve:

- **ASME Service Level A**

  Service Level A includes conditions associated with events that are planned to occur due to routine operation of the plant. Examples include startup, power maneuvers, and shutdown.

- **ASME Service Level B**

  Service Level B includes conditions associated with transients that occur often enough that the operability of the plant is not affected. These transients will not result in damage requiring repairs.

- **ASME Service Level C**

  Service Level C events may result in permanent deformation and repairs may be required to correct large deformations in areas of structural discontinuity.

- **ASME Service Level D**

  Service Level D events may result in gross deformation and dimensional instability. Repair or replacement of components may be necessary to correct mechanical damage.

- **Test Conditions**

These conditions include pressure tests required by ASME BPVC, Section III (Reference 3.9-1), and other tests required by the design specifications.

Table 3.9-1, Summary of Design Transients, lists the design transients by ASME service level and includes the number of events over the design life of the plant for each transient. Load combinations and their acceptance criteria are given in Section 3.9.3 for mechanical components and associated supports and in Section 3.12 for piping systems.

The Service Level A and B transients are representative of events that are expected to occur during plant operation. These transients are severe or frequent enough to be evaluated for component cyclic behavior and equipment fatigue life, and the analyzed conditions are based on a conservative estimate of the frequency and magnitude of temperature and pressure changes. When used as a basis for component fatigue evaluation, the bounding transients provide confidence that the component is appropriate for its application over the design life of the plant. Service Level C and D conditions are not typically included in fatigue evaluations in accordance with the ASME BPVC, Section III (Reference 3.9-1). For select component and transient combinations, Service Level C events are evaluated against Level B stress limits. This selection is made either because the event contains significant stress cycles or the transient is considered a normal design operation for that component. The following sections describe the assumptions used in thermal-hydraulic analysis for each Service Level.

3.9.1.1.1 Service Level A Conditions

Service Level A Transient 1 - Reactor Heatup to Hot Shutdown

This transient covers the heatup and pressurization from transition mode to hot shutdown. The event begins with a depressurized reactor vessel filled with water. The CNV also is filled initially with water up to the elevation of the pressurizer baffle plate.

The CNV is pressurized to at-or-above the minimum pressure required to begin the containment drain process. The RCS is pressurized equivalently by adding nitrogen gas to the pressurizer. Once pressurizer heaters are actuated to increase RCS pressure, the nitrogen is removed through the reactor pressure vessel (RPV) high point degasification line and is replaced with steam. A single SGS train is used in heatup while the other is isolated to ensure passive cooling capabilities. The RCS temperature changes are limited to 100 degrees F/hr and 200 degrees F/hr in the pressurizer. Subcooling between the pressurizer and RCS hot leg is limited to less than 250 degrees F. After the RCS has reached the hot shutdown temperature and normal operating pressure of 1850 psia, a system leakage test is performed per the requirements of ASME BPVC Section XI (Reference 3.9-2).

Service Level A Transient 2 - Reactor Cooldown from Hot Shutdown

This transient encompasses the cooling from hot shutdown to transition mode and is generally the reverse of the reactor heatup to hot shutdown. The temperature of the RCS is continually reduced by controlling the feedwater flow rate with one SGS train isolated to ensure passive cooling capabilities. The steam and feedwater flow rates are controlled to keep the cooling rate below the maximum of 100 degrees F/hr (200 degrees F/hr in the pressurizer). The RCS temperature changes also are limited to maintain subcooling between the pressurizer and RCS hot leg less than 250 degrees Fahrenheit. The chemical and volume control system (CVCS) is used to increase the boron concentration to shutdown levels and to add makeup to compensate for coolant shrinkage. The containment flooding and drain system is used to add pool water to containment to continue cooling the CNV and RPV. Once the pressurizer steam bubble is collapsed, nitrogen gas is added to the pressurizer to control primary pressure.

Service Level A Transient 3 - Power Ascent from Hot Shutdown

This transient covers the power ascent from hot zero power conditions in hot shutdown mode to 15 percent of full power at which point the control systems are placed in automatic mode. Automatic mode is expected to cover power levels above 15 percent of full power. Throughout this transient, the steam and feedwater flow rates through the unisolated SGS train are controlled to match the demanded load ramp, which is specified to be limited to 0.5 percent of full power per minute. The feedwater temperature remains at the condenser hot well temperature as the feedwater heaters are unavailable.

Service Level A Transient 4 - Power Descent to Hot Shutdown

RAI 03.09.01-3S1

This transient covers the reactor conditions that span from 15 percent of full power to hot zero power conditions in hot shutdown mode. The lower limit of the power range where the reactor is under automatic control occurs at 15 percent of full power. Since the turbine is offline, steam from the unisolated SGS train produced by cooling the RCS is diverted through the turbine bypass valve. Reduced flow through this SGS train may result in flow oscillations. The maximum allowed ramp decrease in power is a rate of 0.5 percent of full power per minute. The reactor is tripped after the turbine is tripped, at which point the cooldown rate is controlled to below 100 degrees F/hr (200 degrees F/hr in the pressurizer). Feedwater heating is not available as the turbine is offline and, therefore, the feedwater temperature is equal to the condenser hot well temperature.

Service Level A Transient 5 - Load Following

The reactor could be required to provide load following capabilities to match the electrical demand of the grid over a 24-hour period. The load begins at full power and ramps down to 20 percent of full power over two hours. The load then remains constant for up to ten hours before ramping back to full power over two hours. The load remains constant at full power for the remainder of a 24-hour cycle.

Service Level A Transient 6 - Load Regulation

Load regulation refers to fluctuations in load due to the plant participating in some form of grid frequency control. The frequency control transient is defined as a 10 percent of full power increase or decrease in load at 2 percent of full power per minute. This load regulation is a plant-wide capacity, thus the change in plant load is the total power change for all operating modules. Load regulation is provided while at a steady power level or while performing the ramp power changes required for load following. Reactor power will lag behind the step change in load demand.

Service Level A Transient 7 - Steady State Fluctuations

While operating at a steady load, there may be small fluctuations in RCS temperature and pressure. These fluctuations could be due to minor control system malfunctions, instrument drifts, small power variations, or other unplanned variations. The full-power, normal operating bands for RCS average temperature and pressurizer pressure are expected to be ±0.5 degrees Fahrenheit and ±5 psia. There may also be small flow oscillations on the secondary side caused by similar small parameter variations. The maximum level of normal oscillations is limited to 10 percent about the mean flow rate measured at the SGS tube inlet (20 percent peak-to-peak).

Service Level A Transient 8 - Load Ramp Increase

When the reactor is in automatic mode, the reactor will be capable of providing a load increase at a rate of 5 percent of full power per minute over the power range of automatic control, 15 to 100 percent of full power. A rate of 5 percent of full power per minute is an upper bound on the load increase rate for power maneuvers and is consistent with other pressurized water reactor designs. Throughout this transient, the pressure, average RCS temperature, and pressurizer level are under automatic control.

Service Level A Transient 9 - Load Ramp Decrease

When the reactor is in automatic mode, the reactor will be capable of providing a load decrease at a rate of 5 percent of full power per minute over the power range of automatic control, 15 to 100 percent of full power. A rate of 5 percent of full power per minute is an upper bound on the load decrease rate for power maneuvers and is consistent with other pressurized water reactor designs. Automatic control mode is initiated at 15 percent of full power. Feedwater temperature will decrease due to less feedwater heating as the power level decreases.

Service Level A Transient 10 - Step Load Increase

When the reactor is in automatic mode, the nuclear steam supply system components are designed to withstand the cycles associated with a 10 percent of full power step load increase. This transient could occur due to a disruption in the electrical grid. As load is increased, the imbalance between load and core power causes the RCS temperature and pressure to decrease. The pressurizer heaters will respond to a pressure decrease by increasing proportional heater output and energizing the backup heaters.

Service Level A Transient 11 - Step Load Decrease

Nuclear steam supply system components must also be capable of withstanding the cycles associated with a 10 percent of full power step load decrease. This transient could occur due to a disruption in the electrical grid. As load is decreased, the imbalance between load and core power causes the temperature and pressure to increase. The pressurizer will respond to any large pressure increase by reducing heater output and initiating normal spray flow.

Service Level A Transient 12 - Large Step Load Decrease

This transient occurs when there is a large decrease in the demanded load by the grid from full power down to 20 percent of full power. When the load decreases, steam pressure increases and steam flow rate decreases. RCS temperature and pressure increase due to the decrease in secondary heat removal. Some steam will likely need to bypass the turbine to prevent a reactor trip on high pressurizer pressure or level. The bypass load is ramped down at about 5 percent power per minute to give the reactor time to reduce power. There are two control signals to detect a large step load decrease and to regulate the bypass steam flow. An error signal between the demanded and actual turbine load will determine the need for steam bypass, and a signal will provide the expected bypass valve position as a function of the demanded load.

Service Level A Transient 13 - Refueling

During refueling, the containment vessel flange and reactor vessel flange are opened and the upper portion of the NuScale Power Module is lifted away from the lower portion, exposing the reactor core for refueling. This operation takes place in the refueling pool. There is a negligible thermal cycle on the RPV as the flanges are unbolted and cold pool water mixes in. For module handling operations to begin, the RCS must be in transition mode~~, below 200 degrees Fahrenheit. The maximum temperature change would then be 200 degrees Fahrenheit minus the minimum pool water temperature~~. There will be negligible thermal cycles for cold unbolting and re-bolting of the RPV flanges for reactor startup, because the RPV temperature will be near equilibrium with the surrounding reactor pool water following the duration of a refueling outage. There will also be stress cycles introduced by the module handling operations, such as module lifting, bolting, placing, and removing the NPM in module handling tools. ~~and unbolting.~~

Service Level A Transient 14 - Reactor Coolant System Makeup

The RCS makeup transient consists of the normal replenishment of RCS fluid due to minor leakage or for boron concentration adjustment by the CVCS makeup pumps. The CVCS continuously circulates coolant through the demineralizers and filters and back to the RCS. Makeup flow is required to maintain the pressurizer level, change the boron concentration, or adjust the RCS chemistry.

This transient begins when the CVCS makeup pumps are energized to add makeup coolant. The makeup coolant can be demineralized or borated water. CVCS flow is pumped to the RCS through the RCS injection line and to the pressurizer through into the containment vessel does not condense and collect at the bottom. If there is leakage, then the containment evacuation system is expected to run, continuously or intermittently, until the leak is fixed during the next reactor shutdown.

Service Level A Transient 18 - Containment Flooding and Drain

The containment flooding and drain system connects to a CNV nozzle with piping extending from the top head of the CNV to the bottom for each module. The piping is used to add and remove water to and from the CNV. This transient is split into two events: containment flooding operations after shutdown and containment drain operations prior to startup. After shutdown, the containment flooding and drain containment isolation valves are opened and the pump transfers water from the reactor pool to the CNV. Prior to startup of the module, the containment flooding and drain system containment isolation valves are opened and containment is pressurized through the containment evacuation system penetration to the minimum pressure required or to provide adequate net positive suction head to the pump, which helps drain the CNV of water.

3.9.1.1.2 Service Level B Conditions

Service Level B Transient 1 - Decrease in Feedwater Temperature

A decrease in feedwater temperature could occur due to many different malfunctions in the secondary side system. However, the bounding malfunction is the loss of feedwater heating. Such a failure at full power drops the feedwater temperature significantly, which quickly reduces the RCS temperature and adds reactivity due to the negative moderator temperature coefficient. ~~The colder coolant and reactivity insertion cause core power to increase and can result in a reactor trip on high power.~~ The secondary control system compensates for the lower feedwater temperature by adjusting the feedwater flow rate to reach the load demand setpoint. ~~If a trip setpoint is not reached, r~~Reactivity feedback will allow the reactor power to re-adjust to match the demanded load.

Service Level B Transient 2 - Increase in Secondary Flow

An equipment or control system malfunction could cause an increase in secondary flow. A malfunction could be on the steam side, such as opening the turbine throttle valve, or on the feedwater side, such as opening the feedwater regulating valve or increasing the feedwater pump speed. Any of these malfunctions leads to an increase in feedwater flow rate, but the feedwater pressure could increase or decrease. One of the control valves opening leads to a feedwater pressure decrease while an increase in feedwater pump speed increases the feedwater pressure.

The bounding cases are the following: the complete opening of either the feedwater regulating valve, turbine throttle valve, or turbine bypass valve or the feedwater pump speed increasing to 100 percent. The RCS responds to an increase in secondary flow rate with a decrease in temperature and pressure. Reactivity feedback then causes an increase in reactor power.

There will also be a control system response for the secondary side. The steam superheat will fall below the setpoint and the actual SG load will be larger than the setpoint. The feedwater regulating valve and the turbine throttle valve will both close to try to match the superheat and load setpoints. ~~Depending on the magnitude of the feedwater flow or steam flow increase and what caused the malfunction, this transient may~~This transient leads to a turbine trip due to low superheat or a higher load than demanded. A reactor trip will also occur on low pressurizer level, low steam pressure, or high reactor power, and if the change in steam pressure causes the main steam and feedwater isolation valves to close, then the DHRS will be actuated.

Service Level B Transient 3 - Turbine Trip without Bypass

The turbine trip transient may be caused by any of several equipment or control system malfunctions. This transient covers the scenario where the turbine trip leads to a reactor trip. Once the turbine trips, the turbine stop valve shuts, stopping all steam flow and increasing steam pressure. The turbine bypass is postulated to be unavailable.

The RCS pressure and temperature increase due to the loss of heat removal and the pressurizer level rises due to the expanding RCS fluid. The reactor will trip and actuate both trains of the decay heat removal system (DHRS) to remove decay heat and cool the RCS. The reactor safety valves (RSVs) do not open.

Service Level B Transient 4 - Turbine Trip with Bypass

The turbine trip transient may be caused by any of several equipment or control system malfunctions. This transient covers the scenario when the turbine trips and the turbine bypass flow is available. After switching to bypass flow, the feedwater temperature decreases due to the feedwater heaters being offline. Reactor power stabilizes at its original level. The reactor does not trip. Reactor power is then decreased at a rate consistent with the Service Level A Transient 9-Load Ramp Decrease. Feedwater heating is not available throughout the power decrease.

Service Level B Transient 5 - Loss of Normal AC Power

A loss of normal AC power consists of a loss of AC power with no credit taken for the backup power supply system. Under these circumstances the reactor trips, the containment isolation valves fail closed, and the DHR actuation valves fail open. The module reaches a safe shutdown state by dissipating the heat through the DHR condensers. Batteries supply power to the five emergency core-cooling-system (ECCS) valves (three reactor vent valves (RVVs) and two reactor recirculation valves (RRVs)) that hold the valves closed. Once battery power is supplied to the ECCS valves a 24 hour timer begins. After 24 hours, battery power is removed and the RVVs and RRVs fail open. Actuation of the ECCS establishes a two-phase, natural circulation loop. Steam generated in the RPV exits through the RVVs and condenses on the walls of the CNV. The condensed water returns to the RPV through the RRVs. Coincident losses of the DC power systems, EDS or EDNS, as well as delays in MPS actuations, are considered to determine bounding pressure and temperature responses for mechanical design.

Service Level B Transient 6 - Inadvertent Main Steam Isolation Valve Closure

An inadvertent closure of an MSIV will cause a sudden decrease in the secondary-side flow for the affected SG and an increase in flow in the other SG. The closed MSIV causes the SG pressure to increase. The reactor trips on either high-steam pressure or high-pressurizer pressure.

The RSVs do not lift. Both trains of the DHRS are actuated. The DHRS removes heat through the two SGs and rejects the heat to the reactor pool. The components of the DHRS are sized to remove decay heat and cool the RCS.

Service Level B Transient 7 - Inadvertent Operation of the Decay Heat Removal System

The inadvertent operation of the DHRS could occur in two ways. The first is the inadvertent opening of one of the DHRS actuation valves. Opening an actuation valve allows flow between the DHRS condenser and the steam line as the steam and feedwater pressures equalize. The initial pressure equalization in the secondary side causes a disruption in the primary temperature. Both DHRS trains actuate and the reactor trips. The second way to inadvertent DHRS actuation is by the module protection system (MPS) sending a signal to actuate the DHRS by closing the MSIVs and feedwater isolation valves and opening the DHRS actuation valves on both trains of the DHRS. This results in the full-power operation of both trains of the DHRS. The DHRS actuation signal causes a reactor trip. The RSVs do not lift for either occurrence.

Service Level B Transient 8 - Reactor Trip from Full Power

A reactor trip from full power could be caused by multiple spurious sensor signals to the module protection system (MPS), or a spurious trip signal from the MPS, or miscellaneous failures that cause a reactor trip setpoint to be reached and are not already included in other transients. Once the trip begins, the control rods drop into the core to take the core subcritical. This reduces the core thermal power to decay heat and causes the hot- and cold-RCS temperatures to converge close to the average RCS temperature. Cooling is then initiated by one of two methods, either normal feedwater or actuating the DHRS. If the DHRS is actuated, then a containment isolation signal may also be generated. When circulating feedwater through the SGs, the steam produced is directed through the turbine bypass valve to the condenser. The steam and feedwater flow rates are controlled to keep the cooling rate below the maximum of 100 degrees F/hr (200 degrees F/hr in the pressurizer). This transient ends once the reactor reaches approximately steady hot shutdown conditions. Any cooldown from there is accounted for in the cycles of the cooldown from hot shutdown. If the DHRS is actuated for a more severe failure, heat is removed through the DHRS condenser to the pool.

Service Level B Transient 9 - Control Rod Misoperation

This transient includes misoperations of the control rod assemblies (CRAs), such as the drop of a single CRA, the drop of a bank of CRAs, withdrawal of a single CRA, or withdrawal of a CRA bank. The CRA adds significant negative reactivity to the core that quickly reduces reactor power. Such a reduction in power leads to a decrease in RCS temperature and pressure. The decreasing temperature leads to a reactivity insertion due to the negative moderator temperature coefficient. The reactor ~~will likely~~ trips on low pressure or pressurizer level. ~~However, if the rod worth is low enough, the reactor may reach a new steady state at the original power level but with a lower average RCS temperature.~~ Removal of decay heat is by feedwater flow.

Service Level B Transient 10 - Inadvertent Pressurizer Spray

The inadvertent pressurizer spray transient entails, either through equipment failure or operator error, actuation of continuous pressurizer spray. With the spray control valve fully open, spray flow at the maximum design flow and the minimum expected temperature is provided to the pressurizer. The pressurizer heaters energize to counteract the decrease in pressurizer pressure. A reactor trip on low pressurizer pressure will occur. The low pressurizer pressure also triggers containment isolation and actuates both trains of the DHRS. Once the reactor trips, it will take the operators some time to identify the failure that caused depressurization. Removal of decay heat is by the DHRS.

Service Level B Transient 11 - Cold Overpressure Protection

When the RPV is at low temperatures, the metal is more prone to brittle failure. To prevent this type of failure, lower maximum pressure limits are implemented when the RPV is at low temperature. Cold overpressurization could be caused by equipment malfunctions or operator error that cause excessive heat or inventory to be added to the RCS. The RVVs are providing protection against low-temperature overpressurization.

If the RCS is at or below the low-temperature overpressure protection enable temperature and the RCS pressure is at or above the low temperature overpressure protection pressure setpoint, the RVVs will open to relieve the pressure by blowing down to the CNV. Interlocks in the control system will prevent this action when the reactor coolant is above the low temperature overpressure protection enable temperature. When the RVVs open, all of the components within the RPV experience a rapid decrease in fluid pressure. The CNV pressure will increase as it receives coolant from the RPV and once the RCS pressure and CNV pressure reach equilibrium, the RRVs will be opened.

Service Level B Transient 12 - CVCS Malfunctions

This transient includes malfunctions of the CVCS that can cause an increase in RCS inventory or addition of cooler water to the RCS. An increase in RCS inventory could result from a spurious makeup pump operation, excessive charging, or a failure in the letdown line to compensate for the increase in inventory. These events could cause the pressurization of the RCS and a CVCS isolation or reactor trip will likely occur. If there is a malfunction of the pressurizer spray and the RCS pressure is high enough to reach the RSV setpoint, the RSVs will lift to release pressure. Another CVCS malfunction transient is possible if recirculation flow is stopped due to the malfunction of the CVCS recirculation pumps. A full or partial valve closure in the letdown line is also specified, which limits the amount of letdown flow. This would allow colder makeup water to be pumped to the RCS using the makeup pumps, with limited heat addition through the regenerative heat exchanger. Depending on the reactor power level and primary flow rate, the addition of colder makeup water could affect the reactivity, which ~~could possibly~~ results in a reactor trip on high reactor power.

3.9.1.1.3 Service Level C Conditions

Service Level C Transient 1 - Spurious ECCS Valve Actuation

The ECCS consists of three RVVs and two RRVs. In the event of an inadvertent actuation of an RVV or RRV, the inadvertent actuation block feature provides mechanical pressure-locking to prevent opening of the valve when the RCS and CNV are at normal operating pressure. The inadvertent actuation block is treated as a passive device and is not considered for single active failure. The inadvertent opening of a single ECCS valve is an event analyzed in 15.6.6. The opening of a single RVV or RRV could be caused by a malfunction of passive equipment, ~~such as failure of the inadvertent actuation block,~~ if an ECCS signal is present or if DC power is lost. This event causes a decrease of RCS inventory due to the blowdown of RCS fluid to the CNV. The bounding operating condition for the opening of an RVV or RRV is full power operation. When the ECCS valve opens, a reactor trip signal is generated on either high containment pressure or low pressurizer pressure. The high containment pressure signal would also cause a containment isolation and DHRS actuation. The open ECCS valve allows reactor coolant to blow down into the CNV. As the hot steam contacts the CNV walls, it condenses to liquid and accumulates in the bottom of the CNV. The CNV wall is cooled by convection to the surrounding reactor pool. The remaining ECCS valves open when either the liquid accumulating in the CNV reaches the high CNV water level setpoint ~~or the RCS liquid level decreases to the low RCS level setpoint~~. This configuration establishes a two-phase, natural recirculation loop that provides cooling for the RCS through the RVVs and keeps the core covered by returning liquid to the RPV through the RRVs.

Simultaneous failures that result in the inadvertent opening of multiple ECCS valves when the RCS is at normal operating pressure is beyond design basis with respect to identifying initiating events.

Service Level C Transient 2 - Inadvertent Opening of a Reactor Safety Valve

The inadvertent opening of one of the RSVs causes the RCS to quickly depressurize as the primary coolant blows down to the CNV. The reactor will trip likely due to high containment pressure or low pressurizer pressure. The high containment pressure causes a containment isolation and DHRS actuation signal. The hot vapor entering the CNV will condense on the walls and fall to the bottom of the CNV. When the high CNV liquid level setpoint is reached~~either the low RCS or high CNV liquid level setpoints are reached~~, all five ECCS valves will open. The open valves establish the ECCS two-phase, natural recirculation loop. Decay heat is removed by the vapor moving through the RVVs to the CNV and the core is kept covered by the liquid returning to the RPV through the RRVs. Removal of decay heat is expected through the containment wall and peak pressure in the CNV is kept below design pressure.

Service Level C Transient 3 - CVCS Pipe Break

The CVCS Pipe Break is characterized by a rupture of a pipe penetrating the RCPB. The break could occur inside or outside of containment. A break inside containment maximizes the dynamic response of the RPV and RVI and captures a pressure and thermal cycle for the CNV and components inside containment. A break outside of containment could cause stresses on the components just outside of containment. In this transient, the RCS depressurizes through the break and the level in the pressurizer decreases. The reactor trips due to either low pressurizer pressure or level or high containment pressure, and the DHRS is actuated. The ECCS ~~may eventually actuate based on either a low RCS liquid level or~~actuates on a high water level in containment. Removal of decay heat is expected through the containment wall and peak pressure in the CNV is kept below design pressure.

Service Level C Transient 4 - Steam Generator Tube Failure

The steam generator tube failure (SGTF) transient is bounded by the double-ended failure of a SG tube. The term “failure” is used here to include both a tube collapsing due to higher external pressure and a tube bursting due to higher inner pressure. Multiple simultaneous SGTFs are considered beyond design basis. In this transient, the RCS blows down into the SG. A reactor trip would occur quickly due to high steam pressure, low pressurizer pressure, or low pressurizer level. Both trains of the DHRS will be actuated to remove the decay heat as normal cooldown using feedwater flow is not possible with SGTF. A SGTF incapacitates one train of the DHRS, but cooldown is still accomplished with the other train. Components within the RPV will experience a decrease in pressure when the SG tube fails and the RCS blows down to the SG. Once the MSIVs and feedwater isolation valves close and the DHRS actuates, the pressure decrease will slow to be only a function of the RCS cooldown rate. The cooldown rate is determined by the performance of the single DHRS train.

3.9.1.1.4 Service Level D Conditions

Service Level D Transient 1 - Steam Piping Failures

A main steam line break~~ could cover a wide range of break types. A rupture~~ will cause an increase in steam flow rate and will reduce the SG inventory. A break inside containment is not postulated to occur because of leak before break detection on these lines. A break outside of containment could cause stresses on the components just outside of containment. RCS temperature and pressure briefly decrease due to the excess heat removal provided by the steam line blowdown. A break will quickly cause a reactor trip on low steam pressure or high containment pressure. Once the reactor is tripped, both trains of the DHRS will be activated. ~~One train of the DHRS will be ineffective due to the break. A single train of~~If the break compromises the water inventory inside one DHRS train, the remaining train of the DHRS will be capable of removing the decay heat from the reactor. The RSVs do not lift and there is no ECCS actuation. Removal of decay heat is by the DHRS and peak pressure in the CNV is kept below design pressure.

Service Level D Transient 2 - Feedwater Piping Failures

A feedwater line break could cover a wide range of break types. Due to the interaction of the DHRS and feedwater system, the spectrum of feedwater piping breaks includes breaks in the DHRS. A feedwater piping break inside containment is not postulated to occur because of leak-before-break detection on these lines, but a break in the DHRS condensate line inside containment is postulated. A break outside of containment could cause stresses on the nearby components. RCS temperature and pressure briefly decrease due to the excess cooling provided by the feedwater line blowdown. Once the quick blowdown phase is over, the transient results in heating and pressurization of the RCS. A break will quickly cause a reactor trip on low steam pressure or high containment pressure. Once the reactor is tripped, both trains of the DHRS will be activated. ~~One train of the DHRS will be ineffective due to the line break. A single train of~~If the break compromises the water inventory inside one DHRS train, the remaining train of the DHRS will be capable of removing the decay heat from the reactor. The RSVs do not lift and there is no ECCS actuation. Peak pressure in the CNV is kept below design pressure.

Service Level D Transient 3 - Control Rod Assembly Ejection

This transient covers a spectrum of possible control rod ejection scenarios in order to find the most limiting case. Scenarios must be considered at different power levels, fuel burnups, and rod configurations. The most reactive control rod for a given scenario is postulated to be ejected from the core. Removing the control rod causes a local reactivity insertion that leads to a pressure increase. Once the rod is ejected, there will be a delay before the module protection system trips the reactor. The trip could be caused by high reactor power or high-rate power change.

Service Level D Transient 4 - Combustible Gas Detonation

This transient covers the CNV and components necessary to maintain safe shutdown during a hydrogen event. This includes maintaining continuity of ECCS operation and maintaining containment pressure integrity; DHRS operation is not required during these events beyond maintaining containment integrity (as applicable). The CNV and components must withstand the environmental conditions created by the burning of hydrogen within the first 72 hours of any design basis event and maintain containment structural integrity and safe shutdown capability.

A typical design basis event where combustible gas control is relevant is any event that results in ECCS actuation. Initiating events that result in ECCS operation include ~~LOCAs~~CVCS pipe breaks, spurious valve openings, and a loss of all DC power. Regardless of the initiating event, the outcome is similar: the ECCS successfully actuates and maintains RPV liquid level above the top of the core. Because heat removal from the containment is very effective, temperatures will usually decrease rapidly. Subatmospheric pressure in the containment is expected within a few hours after event initiation.

Continued operation and long term cooling by the ECCS will result in a stable condition. Aside from temperature gradually approaching the reactor pool

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Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
		Containment System (CNTS)	
• Supports reactor building by providing a barrier to contain mass, energy, and fission product release from a degradation of the reactor coolant pressure boundary (RCPB) • Supports reactor building by providing a barrier to contain mass, energy, and fission product release by closure of the containment isolation valves (CIVs) upon containment isolation signal • Supports emergency core cooling system (ECCS) operations by providing a sealed containment and thermal conduction for the condensation of steam that provides makeup water to the reactor coolant system (RCS) • Supports control rod drive system (CRDS) by providing structural support for the control rod drive mechanisms • Supports RCS by providing structural support for the reactor pressure vessel (RPV) • Supports RCS by transferring core heat from reactor coolant in containment to the ultimate heat sink (UHS) • Supports ECCS by providing structural support of the trip and reset valves for the ECCS reactor vent and recirculation valves • Supports neutron monitoring system (NMS) by providing structural support for the ex-core detectors • Supports ECCS by providing electrical penetration assemblies for reactor instrumentation cables through containment vessel (CNV) • Supports RCS by closing the CIVs for pressurizer spray, chemical and volume control system (CVCS) makeup, CVCS letdown, and RPV high point degas when actuated by module protection system (MPS) for RCS Isolation • Supports MPS by providing MPS actuation instrument information signals through CNV	A1	All CNTS SSC with the exception of the following: • CIV close and open position sensors: - Containment evacuation system (CES), inboard and outboard - Containment flooding and drain system (CFDS), inboard and outboard - Chemical and volume control system (CVCS) inboard and outboard pressurizer spray line - CVCS, inboard and outboard RCS discharge - CVCS, inboard and outboard RCS injection - CVCS, inboard and outboard reactor pressure vessel (RPV) high-point degasification - ~~Feedwater system (FWS), supply to steam generators and decay heat removal (DHR) heat exchangers feedwater isolation valves~~ - Reactor component cooling water system (RCCWS), inboard and outboard return and supply - Steam generator system (SGS), steam supply CIV/ main steam isolation valves (MSIVs) and CIV/MSIV bypasses • CVCS Discharge Excess Flow Check Valve • CVCS Injection Check Valve • CVCS PZR Spray Check Valve • CFDS piping inside containment • Containment air temperature detectors (RTDs) • Piping from systems (CES, CFDS, CVCS, FWS, MSS, RCCWS) CIVs to disconnect flange (outside containment) • Containment pressure transducers (wide range) • Feedwater temperature transducers (RTDs) • Hydraulic skid for valve reset	Determination by probabilistic risk assessment (PRA) and concurrence by the expert panel as being needed for maintaining containment and RCPB integrity, removing fuel assembly heat, reactivity control, and emergency response

Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
• Supports reactor building crane (RBC) by providing lifting attachment points that the RBC can connect to, so that the module can be lifted	B1	• NuScale Power Module lifting lugs and top auxiliary mechanical access structure diagonal lifting braces • Top auxiliary mechanical access structure	Determination by PRA and concurrence by the expert panel as being needed for maintaining containment integrity
Steam Generator System (SGS)			
• Supports RCS by supplying part of the RCPB	A1	• Steam generator tubes • Steam generator tube supports • Feedwater plenums • Integral ~~S~~steam plenums	Determination by PRA and concurrence by the expert panel as being needed for maintaining RCPB integrity
Reactor Core System (RXC)			
• Supports control rod assembly (CRA) by providing control rod guide tubes to receive and align the CRA • Supports RCS by containing fission products and transuranics within the fuel rods to minimize contamination of the reactor coolant • Supports RCS by maintaining a coolable geometry	A1	• Fuel assembly	Determination by PRA and concurrence by the expert panel as being needed for reactivity control, radioactivity control, and removing fuel assembly heat
Control Rod Drive System (CRDS)			
• Supports CRA by releasing control rod during a reactor trip	A1	• Control rod drive shafts • Control rod drive latch mechanism	Determination by PRA and concurrence by the expert panel as being needed for reactivity control

Table 17.4-1: D-RAP SSC Functions, Categorization, and Categorization Basis (Continued)

System Function	Function Category (A1 & B1)	SSC Required to Perform System Function	Basis for Function Categorization
• Supports CNT by removing electrical power to the trip solenoids of the following CIVs on a containment system isolation actuation signal: - RCS injection CIVs - RCS discharge CIVs - Pressurizer spray CIVs - RPV high point degasification CIVs - Feedwater isolation valves - Main steam isolation valves - Main steam bypass isolation valves - Containment evacuation isolation valves - Reactor component cooling water inlet and outlet CIVs - Containment flooding and drain CIVs • Supports CNT by removing electrical power to the trip solenoids of the following CIVs on a CVCS isolation actuation signal: - RCS injection CIVs - RCS discharge CIVs - Pressurizer spray CIVs - RPV high point degasification CIVs • Supports CVCS by removing electrical power to the trip solenoids of the demineralized water system isolation valves on a DWS isolation actuation signal • Supports CRDS~~non-safety DC electrical and AC distribution system~~ by removing electrical power to the CRDS for a reactor trip • Supports CNT by providing power to sensors • Supports DHR by providing power to main steam pressure sensors • Supports RCS by providing power to sensors • Supports CNT by removing electrical power to the trip solenoids of the following valves on a secondary system actuation signal. - main steam isolation valves - main steam isolation bypass valves - feedwater isolation valves • Supports CNT by providing power to position sensors on the feedwater isolation valves.		• 24-hour timers for post-accident monitoring-only mode • Division I and Division II: - ESFAS - equipment interface module for loss of AC voltage function - ESFAS monitoring and indication bus - communication module - MPS gateway - Reactor trip system monitoring and indication bus - communication module • Separation Groups A, B, C, and D: - Monitoring and indication bus - communication module • Separation Groups B and C - safety function modules for post-accident monitoring (PAM) indication functions • Separation Group A - safety function module: - Feedwater indication and control - Leak detection into containment • Separation Groups B and C - safety function module for PAM indication functions • Separation Group D - safety function module: - Leak detection into containment • Division I and II maintenance workstations	