RAIO-0519-65608

May 16, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 520 (eRAI No. 9642) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 520 (eRAI No. 9642)," dated March 20, 2019
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 520 (eRAI No. 9642)," dated April 15, 2019

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 9642:

- 16-61
- 16-62
- 16-63
- 16-64
- 16-65

The response to RAI Question 16-66 was previously provided in Reference 2. This letter completes all responses to eRAI 9642.

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9642

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9642

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9642

Date of RAI Issue: 03/20/2019

NRC Question No.: 16-61

Paragraph (a)(11) of 10 CFR 52.47 and paragraph (a)(30) of 10 CFR 52.79 state that a design certification (DC) applicant and a combined license (COL) applicant, respectively, are to propose technical specifications (TS) prepared in accordance with 10 CFR 50.36 and 50.36a. 10 CFR 50.36 sets forth requirements for TS to be included as part of the operating license for a nuclear power facility.

Generic TS Subsection 3.4.3 Action D states:

> D. Containment flooding initiated while RCS temperature greater than allowed by PTLR. D.1 Be in MODE 2. | Immediately AND D.2 Be in MODE 3 below the PTLR RCS temperature limit. | 36 hours AND D.3 Determine RCS is acceptable for continued operation. | Prior to entering MODE 2 from MODE 3

The staff requests additional information about Action D, regarding whether this provision is the appropriate means of addressing prevention and mitigation of the postulated inadvertent actuation of the CFDS to flood the containment vessel with RCS temperature above the RCS temperature limit in the PTLR. Condition D is an unusual Condition, which is based on an inadvertent actuation of the CFDS in MODE 1, 2, or MODE 3 above the PTLR limit on RCS temperature for containment flooding having occurred.

Revision 2 of DCA part 2, Tier 2, FSAR Section 9.3.6.2.2 describes the CFDS supply/drain isolation valve interlock with RCS hot temperature:

> *Flooding and draining an individual CNV is conducted through the same CNV penetration. The CFDS pump operation is automatically prevented if the CFDS*

> *isolation valve to more than one NPM is open, and valve operation to other NPMs is prevented once the CFDS pump aligned to an NPM and the pump is in service.* ***In addition, for the selected NPM, the CFDS module isolation valve cannot be opened and CFDS pump start is prevented if RCS wide range hot leg temperature is greater than 350°F.*** *These features, coupled with administrative controls in plant procedures, prevent inadvertent CFDS makeup to an operating NPM.*

Revision 2 of DCA part 2, Tier 2, FSAR Section 9.3.6.2.2 describes normal operation of the CES and CFDS; in a passage labled "**Containment Flooding in Preparation for Refueling**," states in part:

> When the CFDS is used to flood a CNV, **one** of the **CFDS pump**s is aligned to take suction from the reactor pool and discharge to the selected NPM. To minimize thermal stress on NPM components, flooding is initiated only after temperatures for the NPM being flooded are below a specified maximum temperature and reactor pool bulk temperature is above a specified minimum temperature. To ensure that component temperature limits are not exceeded and to prevent inadvertent flooding of an operating NPM, *the selected CFDS module isolation valve cannot be opened and CFDS pump start is prevented if the selected NuScale Power Module RCS wide range hot leg temperature is greater than 350° F.* The CFDS flow path for flooding a CNV includes connections for a temporary skidmounted heater if an off-normal condition requires flooding a CNV with elevated RPV temperatures.
>
> The CFDS alignment for flooding requires that the CFDS containment isolation valves and the CFDS module isolation valve for the NPM being flooded are open. **Both CFDS pumps** are started with the CNV at atmospheric pressure and CFDS performance is monitored by system flow rate, pressure, and temperature, and CNV level instrumentation. The CNV is flooded approximately to the elevation of the RPV pressurizer baffle plate. Automatic action shuts off **the operating CFDS pumps** and closes the NuScale Power Module CES isolation valve when the preset water level in the CNV is reached, as determined by CNV level instrumentation. At completion of flooding operation, the CFDS containment isolation valves and the CFDS module isolation valve for the NPM being flooded are closed.

Revision 2 of DCA part 2, Tier 2, FSAR Section 9.3.6.3 describes inadvertent actuation of the CFDS to flood containment event:

> Inadvertent flooding of the CNV for an NPM that is at power or not below the temperature or pressure required for flooding is prevented by system interlocks and administrative controls within plant procedures. Each NPM is isolated from the CFDS by three valves in series, the NPM isolation valve and the two CFDS containment isolation valves.

Revision 2 of DCA part 2, Tier 2, FSAR Section 15.1.6.1, Loss of Containment Vacuum/Containment Flooding - Identification of Causes and Accident Description, states in part:

> The reactor component cooling water system (RCCWS) provides heat removal to the control rod drive system. The RCCWS supplies RCCW to CNTS that then conducts RCCW to CRDS piping that passes through containment to provide this function. If **piping containing RCCW were to leak or rupture inside the CNV**, a containment flooding event would occur. Other potential containment flooding sources include: **feedwater** containing line break, **main steam** containing line break, **CVCS** fluid containing line break, **high point vent** fluid containing pipe break, and **RCCWS** fluid containing line break. The feedwater fluid containing line break event is evaluated in Section 15.2.7, the SLB event is evaluated in Section 15.1.5, and the CVCS fluid containing line break is evaluated in Section 15.6.2. **The RCCWS fluid line break is a more limiting containment flooding event than a high point vent fluid pipe because it has a temperature lower than the containment saturation temperature. If the lower temperature RCCWS fluid line ruptures, there would be no immediate boiling, preventing the high containment pressure limit from being reached.** The flooding of the CNV could cause an increase in heat transfer from the RPV to containment, cooling the RCS. As the RCS cools, reactor power increases due to the negative moderator coefficient. This unexpected rise in core power would decrease the MCHFR, and lead to an over pressurization of the RPV.

1. Regarding the above quotation from FSAR Section 9.3.6.2.2:

1.1 The first paragraph indicates one CFDS pump is aligned for flooding the CNV; the second paragraph indicates that two CFDS pumps are used, and that the pumps stop when the "specified" CNV water level is reached (based on CNV level instrumentation, which is also used for initiating ECCS - MPS Function 3.3.1.23.a). The applicant is requested to clarify whether one or two pumps are used, and state the design single pump flow rate, and the combined pump flow rate.

1.2 In the first paragraph, what are the approximate RCS hot temperature and reactor pool water temperature limits referred to in: "...temperatures for the NPM being flooded are below a specified maximum temperature and reactor pool bulk temperature is above a specified minimum temperature"?

1.3 In the italicized passage in the first paragraph, explain whether the CFDS pump is prevented from starting because of the RCS hot temperature interlock signal, or because the CFDS NPM isolation valve is closed, and its position indication is interlocked with the pump control circuit.

1.4 The applicant is requested to compare the assumed 40°F RCCW temperature and the assumed 1320 gpm break flow rate from two operating RCCWS pumps to the (? gpm) flow rate from two operating CFDS pumps into containment with a reactor pool source temperature of 110°F, and describe whether the RCCWS pipe break inside containment AOO analysis would bound the analysis of an inadvertent actuation of the CFDS to flood containment event.

1.5 The applicant is requested to explain how and why the CES isolation valve automatically closes when the preset water level in the CNV is reached while flooding the CNV.

2. The applicant is requested to (a) explain why the inadvertent actuation of the CFDS to flood containment event is not considered to be an anticipated operational occurrence (AOO) (See Section 15.1.6 "Loss of Containment Vacuum/Containment Flooding"); and (b) discuss whether operator error in aligning the CFDS to the wrong NPM combined with failure of nonsafety-related interlock on the CFDS NPM isolation valve is less likely than a pipe break inside containment in the RCCWS (AOO?), main steam system (Postulated Accident (PA)), feedwater system (PA), or CVCS (PA), which are categorized as indicated by Table 15.0-1, "Design Basis Events"?

As quoted above, Section 15.1.6.1, page 15.1-25 includes the following:

...The RCCWS fluid line break is a more limiting containment flooding event than a high point vent fluid pipe because it has a temperature lower than the containment saturation temperature. If the lower temperature RCCWS fluid line ruptures, there would be no immediate boiling, preventing the high containment pressure limit from being reached. The flooding of the CNV could cause an increase in heat transfer from the RPV to containment, cooling the RCS. As the RCS cools, reactor power increases due to the negative moderator coefficient. This unexpected rise in core power would decrease the MCHFR, and lead to an over pressurization of the RPV.

A loss of containment vacuum event is categorized as an AOO. Typically, pipe system failures are categorized as accidents, but **the containment flooding event is conservatively categorized as an AOO.**

3. Explain why there should not be an SR to verify the two CFDS containment isolation valves are closed with dc power disconnected or isolated from the solenoid, to prevent engaging the hydraulic system that opens the valves, until RCS hot temperature is less than or equal to 350°F - the PTLR limit?

4. Will the 350°F approximate upper limit for RCS hot temperature allowed by the PTLR to initiate containment flooding be affected by the change in the LCO 3.5.3 upper temperature limit of the reactor pool from 140°F to 110°F? How does this RCS hot temperature upper limit vary with the temperature of the reactor pool water source of the CFDS?

5. Should manual opening of the two CFDS CIVs be blocked unless a permissive signal exists from a new MPS Permissive Function based on wide range RCS hot temperature channels? Or justify why the existing RCS temperature nonsafety-related interlock to prevent opening the nonsafety-related (air or motor?) isolation valve in the CFDS supply line to containment (upstream of CFDS CIVs) using the manual open control function (module control system) provides adequate protection by precluding this event from occurring.

NuScale Response:

As described in FSAR section 9.3.6.1, the Containment Evacuation System (CES) and the Containment Flood and Drain System (CFDS) do not have any safety-related functions and are not required to operate during or after any design basis accident.

The systems are aligned and operated at different times during plant operations.

- The CES is in operation to maintain containment atmosphere at a vacuum during reactor operations and has no function during other times. The CES containment isolation valves are opened during operations to provide a flowpath to the vacuum pumps that maintain the containment atmosphere at a very low pressure.
- The CFDS is in use during shutdown and startup evolutions and has no function during reactor operations. The CFDS containment isolation valves are normally closed and only opened to add or remove reactor pool (ultimate heat sink - UHS) water to the containment vessel during normal shutdown and startup evolutions.

A description of the operation of the systems during different plant conditions is provided in FSAR 9.3.6.2.3.

LCO 3.4.3, Action D was provided to specify actions to be taken if inappropriate containment flooding occurred for whatever reason. It is not a means for prevention or mitigation of such an occurrence - rather it describes the actions that must be taken if the inappropriate flooding occurs.

Action D was provided based on the unique nature of the NuScale design compared to other PWR designs. The Action is somewhat analogous to Actions A, B, and C however it was recognized that those Conditions did not explicitly address conditions outside the RCS that could affect the OPERABILITY of the reactor vessel. The other Conditions and Actions in 3.4.3, and those in other PWRs are directed at controlling RCS pressure, temperature, and heatup and cooldown rates of change. However the NuScale design routinely floods the exterior of the reactor vessel during cooldown evolutions and it was determined appropriate to provide a Condition and appropriate Actions for that condition.

The following discussions are provided in response to the sub-questions in the RAI.

1.1 Use of the CFDS to flood the containment vessel as described in FSAR section 9.3.6.2.2 is a manual function that only occurs when appropriate conditions exist and is performed in accordance with approved operating procedures. Detailed operating procedures have not been prepared, however they are described FSAR section 13.5 and are required by the technical specification 5.4.1. Current plans are for only one CFDS pump to be used for flooding or draining operation with the other pump in standby. The detailed design of the non-safety related, uncredited CFDS pumps will be completed at a future time. The design, use, and flow rates of the pumps is irrelevant to the safety of the NuScale plant as supported by the discussion above and described in Section 9.3.6.1. The description of system operation in FSAR section 9.3.6.2.2 has been revised.

1.2 The temperature limits described will be specified in the PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) as required by technical specifications 5.6.4. FSAR section 5.3 describes the reactor vessel including limits and materials that will be considered in developing the specific temperature limits implemented during operations that control the use of the CFDS to flood the containment. The Ulitmate Heat Sink (UHS) minimum and maximum reactor pool temperature limits are also specified by LCO 3.5.3.

1.3 The detailed description of the non-safety interlock is provided in FSAR section 9.3.6.2.2 that states:

> The CFDS pump operation is automatically prevented if the CFDS isolation valve to more than one [NuScale Power Module (NPM)] is open, and valve operation to other NPMs is prevented once the CFDS pump aligned to an NPM and the pump is in service. In addition, for the selected NPM, the CFDS module isolation valve cannot be opened and CFDS pump start is prevented if RCS wide range hot leg temperature is greater than 350° degrees F.

Additional design details of this non-safety related function will be developed when the pump and valve control logic and equipment designs are finalized. The description provided in the FSAR will be implemented as described.

1.4 The CFDS pumps are expected to be designed to provide approximately 100 gpm per pump. Both pumps in operation would only provide about 15% of the 1320 gpm Reactor Component Cooling Water System (RCCWS) flow rate assumed in the analysis provided in FSAR section 15.1.6, loss of containment vacuum.

1.5 The valve identified as the "NuScale Power Module CES isolation valve" in the discussion is an error in section 9.3.6.2.3 of the FSAR. It should refer to the "CFDS module isolation valve." This has been corrected in the FSAR.

2.(a) Inadvertent initiation of the CFDS is not a credible initiating event during reactor operations because there is no single malfunction or operator error that could cause the event to occur.

2.(b) Pipe breaks are deterministically addressed in Chapter 15 consistent with the NuScale Design Specific Review Standard and NUREG 0800.

3. As previously noted, the inadvertent initiation of CFDS is not a credible event because there is no single malfunction or operator error that could cause the event to occur. Failure of an

RCCWS fluid line resulting in containment flooding is evaluated and conservatively categorized as an AOO as described in FSAR section 15.1.6. Plant procedures and indications in the control room provide adequate assurance that the CFDS containment isolation valves remain closed unless the system is in operation. The position of the CFDS containment isolation valves with regard to this specification does not meet any of the criteria of 10 CFR 50.36 that would result in a need for an additional surveillance requirement.

4. The limits on UHS pool temperature in LCO 3.5.3 do not affect the PTLR or the procedural limits that will be implemented to control containment vessel flooding and that ensure compliance with the PTLR limits. Whether 140°F or 110°F, the LCO 3.5.3 limits on pool temperature do not represent normal operating conditions. The LCO merely establishes limits on temperature at which action must be taken to restore the pool temperature to within limits that are credited and assumed in the safety analyses. The normal pool operating temperature remains as listed in Table 9.2.5-1 as 100°F.

The PTLR limits will be developed in accordance with the NRC-approved methodologies as required by technical specification 5.6.4. Procedures will be implemented as described in FSAR section 13.5 and in accordance with COL item 5.3-1.

5. As noted in the response to subitem 3, above, there is no regulatory basis to justify, nor precedence to include additional requirements designed to prevent a condition that is not credible.

Impact on DCA:

The FSAR has been been revised as described in the response above and as shown in the markup provided in this response.

Containment Flooding in Preparation for Refueling

The CFDS, as shown in Figure 9.3.6-2, is used to flood a CNV with reactor pool water to support NPM disassembly and refueling operations. CNV flooding is initiated as part of the NPM cooldown operation because the flooded CNV facilitates NPM cooldown by increasing heat transfer from the RPV to the reactor pool.

When the CFDS is used to flood a CNV, one of the CFDS pumps is aligned to take suction from the reactor pool and discharge to the selected NPM. To minimize thermal stress on NPM components, flooding is initiated only after temperatures for the NPM being flooded are below a specified maximum temperature and reactor pool bulk temperature is above a specified minimum temperature. To ensure that component temperature limits are not exceeded and to prevent inadvertent flooding of an operating NPM, the selected CFDS module isolation valve cannot be opened and CFDS pump start is prevented if the selected NuScale Power Module RCS wide range hot leg temperature is greater than 350~~°~~ degrees F. The CFDS flow path for flooding a CNV includes connections for a temporary skid-mounted heater if an off-normal condition requires flooding a CNV with elevated RPV temperatures.

RAI 16-61

The CFDS alignment for flooding requires that the CFDS containment isolation valves and the CFDS module isolation valve for the NPM being flooded are open. ~~Both CFDS pumps are~~The CFDS pump is started with the CNV at atmospheric pressure and CFDS performance is monitored by system flow rate, pressure, and temperature, and CNV level instrumentation. The CNV is flooded approximately to the elevation of the RPV pressurizer baffle plate. Automatic action shuts off the operating CFDS pump~~s~~ and closes the ~~NuScale Power Module CES isolation valve~~CFDS module isolation valve when the preset water level in the CNV is reached, as determined by CNV level instrumentation. At completion of flooding operation, the CFDS containment isolation valves and the CFDS module isolation valve for the NPM being flooded are closed.

Reactor Coolant System Non-Condensable Gas Removal

The CES can be used to remove dissolved gases from the RCS. This activity is performed prior to CFDS pump-down of the CNV by drawing a vacuum on the CNV with the emergency core cooling system (ECCS) valves open. The CES valve line-up for removal of dissolved gases from the RCS is similar to containment evacuation; it ensures the:

- service air system supply valve to the CES is closed
- vacuum pump bypass valve is closed
- suction and discharge valves for the vacuum pump that will be operated are open
- liquid and gaseous discharge paths from the containment evacuation condenser are established
- ECCS vent valves are open

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9642
Date of RAI Issue: 03/20/2019

NRC Question No.: 16-62

Paragraph (a)(11) of 10 CFR 52.47 and paragraph (a)(30) of 10 CFR 52.79 state that a design certification (DC) applicant and a combined license (COL) applicant, respectively, are to propose technical specifications (TS) prepared in accordance with 10 CFR 50.36 and 50.36a. 10 CFR 50.36 sets forth requirements for TS to be included as part of the operating license for a nuclear power facility.

After November 6, 2018, public meeting conference call with NuScale, the staff decided to consider a new RAI question regarding the Bases for Subsection 3.3.1 clearly stating that the Setpoint Program (SP) and the Channel Calibration surveillance requirement in Subsection 3.3.1 provide TS control of interlock and permissive settings.

1. Since the sensors and transmitters for process variables used by the RTS and ESFAS are also used to generate the interlock and permissive signals, a Channel Calibration of an MPS sensor and transmitter satisfies the calibration requirement for the shared interlock sensor and transmitter. However, it is unclear to the staff whether the settings for interlock activation and deactivation are determined using the setpoint methodology specified by the Setpoint Program (SP) and are verified to be set correctly in the SFM as a part of the Channel Calibration of each associated MPS Function. In Revision 2 of DCA part 4, the Applicable Safety Analyses, LCO, and Applicability sections of Subsection B 3.3.1 also state:

> ...The combination of the continuous self-testing features of the MPS and the CHANNEL CALIBRATION specified by SR 3.3.1.4 verify the OPERABILITY of the interlocks and permissives.

The applicant is requested to confirm that the intended meaning of this statement is that interlock settings are controlled by the SP, and are verified during Channel Calibration.

2. The staff notes that Revision 2 of DCA part 4, Subsection B 3.3.1, page B 3.3.1-19, regarding discussion of High Power Range Positive and Negative Rate – Reactor Trip and Demineralized Water System Isolation, states in part:

> ...The SFM logic unit performs calculations to determine the rate of change and compares the result to a setpoint. The trip provides protection against core damage and protects the reactor coolant pressure boundary (RCPB) during the following events:
>
> - Inadvertent decrease in boron concentration in the RCS; and
> - Control Rod Misoperation.
>
> These trips provide protection from the effects of transients that occur at power levels above the N-2H interlock. The High Positive and Negative Power Range Rate trips are automatically bypassed below the N- 2H interlock and automatically enabled above the N-2H interlock. *Actual setpoints are established in accordance with the Setpoint Program.*

2.a. The applicant is also requested to revise this discussion to make clear that the sentence in italics above means that the SP also governs the *actual settings of the interlocks and permissives*.

2.b. The staff notes that a similar sentence is provided on page B 3.3.1-19 in the discussion of High Power Range Linear Power – Reactor Trip and Demineralized Water System Isolation, but is not provided on page B 3.3.1-20 in the discussion of High Intermediate Range Log Power Rate – Reactor Trip and Demineralized Water System Isolation, nor for any other interlock enabled MPS Instrumentation Function Bases discussion in the Applicable Safety Analyses, LCO, and Applicability sections. The applicant is also requested to revise the Bases so that the relationship of the MPS instrumentation Functions, and their bypassing or enabling interlocks and permissives, to the SP controls and Channel Calibration Surveillances is clear.

3. The applicant is requested to revise SR 3.3.1.4 to explicitly require the Channel Calibration to be performed in accordance with Specification 5.5.10, Setpoint Program, as follows (mark up of Revision 2 of DCA part 4, SR 3.3.1.4):

Perform CHANNEL CALIBRATION on each required channel listed in Table 3.3.1-1 in accordance with Setpoint Program.

Since SR 3.3.1.1 (Channel Check) and SR 3.3.1.4 (Channel Calibration) apply to every MPS instrument Function listed in Table 3.3.1-1, the applicant is requested to consider whether the phrase "on each required channel listed in Table 3.3.1-1" is needed to understand which MPS instrument Functions require Channel Check, and also Channel Calibration in accordance with the Setpoint Program.

4. The applicant is requested to revise Specification 5.5.10, paragraph b, to include either the revision number or the document date of the NRC approved version of TR-0616-49121-P, "NuScale Instrument Setpoint Methodology."

NuScale Response:

1. OPERABILITY of a credited function inherently requires all aspects of the function to perform as described and credited in the FSAR and the associated Bases. This includes associated interlocks and permissives, as well as their setpoints. Verification of the OPERABILITY of the functions therefore requires the interlocks, automatic bypasses, and permissives to be surveilled in a manner that supports the OPERABILITY of the supported functions. The setpoint program (SP) is used to control interlock, automatic bypass, and permissive settings that are required to support the OPERABILITY of the associated and affected functions listed in technical specification 3.3.1-1. A sentence has been added to the subject paragraph to clearly indicate that the Setpoint Program is used to control interlock, automatic bypass, and permissive settings.

2.a The statement described in the NRC's question has been clarified to indicate that the trip, isolation, interlock, and bypass setpoints are governed by the Setpoint Program.

2.b As noted in subresponse 1 above, the Setpoint Program is used to control interlock and permissive settings. Clarification of this was added in 18 locations throughout the Applicable Safety Analyses, LCO, and Applicability section of the Bases for LCO 3.3.1.

3. The requested changes to SR 3.3.1.1 and 3.3.1.4 have been incorporated.

4. The current revision number of TR-0616-49121 was added to Technical Specification 5.5.10. Additionally, the report number, version, and title were placed in square brackets to

ensure that they reflect the most current approved version of the methodology when a COL applicant submits the technical specifications for approval.

Impact on DCA:

The Technical Specifications have been been revised as described in the response above and as shown in the markup provided in this response.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~M~~L. (continued)	~~M~~L.4 Isolate dilution source flow paths in the CVCS makeup line by use of at least one closed manual or one closed and de-activated automatic valve.~~Isolate demineralized water flow to the reactor coolant system.~~	96 hours
	AND	
	~~M~~L.5 Open pressurizer heater breakers.	96 hours
~~N. As required by Required Action C.1 and referenced in Table 3.3.1-1.~~	~~N.1 Be in MODE 2.~~	~~6 hours~~
	~~AND~~	
	~~N.2.1 Be in MODE 3 with RCS temperature below the T-2 interlock.~~	~~48 hours~~
	~~OR~~	
	~~N.2.2 Be in MODE 3 with Containment Water Level above the L-1 interlock.~~	~~48 hours~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK~~ on each required channel listed in Table 3.3.1-1~~.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	------------------------------NOTE---------------------------- Neutron detectors are excluded from the CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION in accordance with the Setpoint Program~~on each required channel listed in Table 3.3.1-1~~.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.5	------------------------------NOTE---------------------------- Not required to be met for Class 1E isolation devices that have isolated 1E circuits from non-1E power. -- Perform CHANNEL CALIBRATION on each required Class 1E isolation device.~~Verify associated Class 1E isolation devices are OPERABLE.~~	In accordance with the Surveillance Frequency Control Program.

5.5 Programs and Manuals

5.5.9 Containment Leakage Rate Testing Program (continued)

c. Containment leakage rate acceptance criterion is < 0.60 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and Type C tests.

d. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

e. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.10 Setpoint Program (SP)

a. The Setpoint Program (SP) implements the regulatory requirement of 10 CFR 50.36(c)(1)(ii)(A) that technical specifications will include items in the category of limiting safety system settings (LSSS), which are settings for automatic protective devices related to those variables having significant safety functions.

b. The Limiting Trip Setpoint (LTSP), Nominal Trip Setpoint (NTSP), As-Found Tolerance (AFT), and As-Left Tolerance (ALT) for each Technical Specification required automatic protection instrumentation function shall be calculated in conformance with [TR-0616-49121-P, Revision 1, "NuScale Instrument Setpoint Methodology."]

c. For each Technical Specification required automatic protection instrumentation function, performance of a CHANNEL CALIBRATION surveillance "in accordance with the Setpoint Program (SP)" shall include the following:

1. The as-found value of the instrument channel trip setting shall be compared with the previously recorded as-left value.

   i. If all as-found measured trip setpoint values during calibration and surveillance testing are inside the two-sided limits of Nominal Trip Setpoint (NTSP) plus or minus the Performance and Test Acceptance Criteria Band (PTAC), then the channel is fully OPERABLE, no additional actions are required.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

RTS and ESFAS Operating Bypass Interlocks and Permissives

Reactor protection permissives and interlocks are provided to ensure reactor trips and ESF actuations are in the correct configuration for the current unit status (Ref. 4). This is to ensure that the protection system functions are not bypassed during unit conditions under which the safety analysis assumes the functions are OPERABLE. Therefore, the permissive and interlock functions do not need to be OPERABLE when the associated reactor trip and ESF functions are outside the applicable MODES. Proper operation of these permissive and interlocks supports OPERABILITY of the associated reactor trip and ESF functions and/or the requirement for actuation logic OPERABILITY. The permissives and interlocks must be in the required state, as appropriate, to support OPERABILITY of the associated functions. The permissives and interlocks associated with each MPS Instrumentation Function channel, each Reactor Trip System (RTS) Logic and Actuation Function division, and each Engineered Safety Features Actuation System (ESFAS) Logic and Actuation Function division, respectively, must be OPERABLE for the associated Function channel or Function division to be OPERABLE. The combination of the continuous self-testing features of the MPS and the CHANNEL CALIBRATION specified by SR 3.3.1.4 verify the OPERABILITY of the interlocks and permissives. Specification 5.5.10, Setpoint Program is used to control interlock and permissive setpoints. The permissives and interlocks are:

Intermediate Range Log Power Permissive, N-1

The Intermediate Range Log Power, N-1 permissive is established when the Intermediate Range Log Power channel increases to approximately one decade above the channel lower range limit. The N-1 permissive performs the following:

1. On increasing power, the N-1 permissive allows the manual block of the following:

   - High Source Range Count Rate Reactor Trip and Demineralized Water System Isolation actuation; and

   - High Source Range Log Power Rate Reactor Trip and Demineralized Water System Isolation actuation.

   This prevents the premature block of the High Source Range Count Rate and High Source Range Log Power Rate trips and allows the operator to ensure that the Intermediate Range channel is OPERABLE as power increases prior to leaving the source range.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Four channels of High Power Range Linear Power are required to be OPERABLE in MODE 1 and in MODES 2 and 3 with the RTBs closed and the CRDMs capable of withdrawing any CRA. In MODES 2 and 3, with no capability of withdrawing any CRA, the reactor will remain subcritical. In MODES 4 and 5 the reactor is subcritical with the CRDMs and CVCS incapable of affecting the reactivity in the unit. Four channels are provided to permit one channel to be in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The High Power Range Linear Power trip logic functions include a permissive, N-2L, that allows the operator to manually bypass the lower Power Range Neutron Flux High trip when power is increased above the N-2L permissive. The Power Range High Linear Power trip setpoint is automatically reset to the lower setpoint when power is reduced below the N-2L permissive. Actual, interlock and permissive setpoints are established in accordance with the Setpoint Program.

b. High Power Range Positive and Negative Rate – Reactor Trip and Demineralized Water System Isolation

The Power Range Rate is measured using the power range neutron monitors that measure neutron flux for the High Linear Power trip. The Power Range Rate function measures the rate-of-change in neutron flux received at the detectors. The SFM logic unit performs calculations to determine the rate of change and compares the result to a setpoint. The trip provides protection against core damage and protects the reactor coolant pressure boundary (RCPB) during the following events:

- Inadvertent decrease in boron concentration in the RCS; and
- Control Rod Misoperation.

These trips provide protection from the effects of transients that occur at power levels above the N-2H interlock. The High Positive and Negative Power Range Rate trips are automatically bypassed below the N-2H interlock and automatically enabled above the N-2H interlock. Actual trip, isolation, interlock, and permissive setpoints are established and governed by the Setpoint Program.~~setpoints are established in accordance with the Setpoint Program.~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Four channels of Power Range Rate are required to be OPERABLE in MODE 1 with reactor power above the N-2H interlock to limit the rate of change of the reactor power as measured by the excore neutron detectors. In MODE 1 with reactor power below the N-2L interlock, and MODES 2 and 3, the High Source and Intermediate Range Log Power Rate trips provide protection from transients that result in high rates of change in reactor power. In MODES 4 and 5 the reactor is subcritical with the CRDMs and CVCS incapable of affecting the reactivity in the unit. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

c. High Intermediate Range Log Power Rate – Reactor Trip and Demineralized Water System Isolation

The Neutron Monitoring System (NMS) provides an intermediate range doubling time signal which is used by the SFM to determine the rate of change and compares the result to a setpoint. The High Intermediate Range Log Power Rate trip provides protection against core damage and protects the reactor coolant pressure boundary (RCPB) during an inadvertent decrease in boron concentration in the RCS that is postulated to occur at low power.

The High Intermediate Range Log Power Rate trip is only necessary for events that are postulated to occur from a subcritical condition or during the approach to critical operations and at low-power levels. It is not required to be OPERABLE at power levels above the N-2L interlock. The High Intermediate Range Log Power Rate trip is automatically bypassed when above the N-2L interlock and automatically enabled below the N-2L interlock. Interlock and permissive setpoints are governed by the Setpoint Program.

Four channels of High Intermediate Range Log Power Rate are required to be OPERABLE in MODE 1 with reactor power below the N-2L interlock and in MODES 2 and 3 when capable of CRA withdrawal because the events that it is design to protect against occur at low power levels. This will limit the rate of change of the reactor power as measured by the excore neutron detectors. At power levels above the N-2L interlock, the High Power Rate trip provides protection from events that result in high rates of change in reactor power. In MODES 2 and 3, with no capability of withdrawing any CRA, the reactor will remain subcritical.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODES 4 and 5 the reactor is subcritical with the CRDMs and CVCS incapable of affecting the reactivity in the unit.

Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

d. High Source Range Count Rate – Reactor Trip and Demineralized Water System Isolation

The NMS provides a source range log power signal which is used by the SFM to determine a source range count rate and compares the result to a setpoint. The High Source Range Count Rate trip provides protection against core damage and protects the reactor coolant pressure boundary (RCPB) during the following events:

- Inadvertent decrease in boron concentration in the RCS; and
- Uncontrolled CRA withdrawal from a subcritical or low power.

Four channels of High Source Range Count Rate are required to be OPERABLE in MODE 1 with power less than approximately one decade above the Intermediate Range channel lower limit and in MODES 2 and 3 when capable of CRA withdrawal. In MODE 1 with power approximately one decade above the Intermediate Range channel lower limit, the Intermediate Range Log Power Rate trips and the Power Range High Linear Power trip provide protection from transients that result in high rates of change in reactor power. In MODES 2 and 3, with no capability of withdrawing any CRA, the reactor will remain subcritical. In MODES 4 and 5 the reactor is subcritical with the CRDMs and CVCS incapable of affecting the reactivity in the unit. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The High Source Range Count Rate trip can be manually bypassed when the intermediate range flux increases to approximately one decade above the channel lower limit (above the N-1 permissive) and is automatically enabled when the intermediate range flux decreases below the N-1 permissive. Interlock and permissive setpoints are governed by the Setpoint Program.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

e. High Source Range Log Power Rate – Reactor Trip and Demineralized Water System Isolation

The NMS provides a source range doubling time signal which is used by the SFM to determine a source range log power rate and compares the result to a setpoint. The High Source Range Log Power Rate trip provides protection against core damage and protects the reactor coolant pressure boundary (RCPB) during the following events:

- Inadvertent decrease in boron concentration in the RCS; and
- Uncontrolled CRA withdrawal from a subcritical or low power.

Four channels of Source Range Log Power Rate are required to be OPERABLE in MODE 1 with power less than approximately one decade above the Intermediate Range channel lower limit and in MODES 2 and 3 when capable of CRA withdrawal. In MODE 1 with power approximately one decade above the Intermediate Range channel lower limit, the Intermediate Range Log Power Rate trips and the Power Range High Linear Power trip provide protection from transients that result in high rates of change in reactor power. In MODES 2 and 3, with no capability of withdrawing any CRA, the reactor will remain subcritical. In MODES 4 and 5 the reactor is subcritical with the CRDMs and CVCS incapable of affecting the reactivity in the unit. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The High Source Range Log Power Rate trip can be manually bypassed above the N-1 permissive and is automatically enabled when the intermediate range flux decreases below the N-1 permissive. Interlock and permissive setpoints are governed by the Setpoint Program.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

f. High Subcritical Multiplication – Demineralized Water System Isolation

The NMS provides a source range log power signal which is used by the SFM to determine a subcritical multiplication rate and compares the result to a setpoint. The High Subcritical

Multiplication trip provides protection against core damage and protects the reactor coolant pressure boundary (RCPB) during the following events:

- Inadvertent decrease in boron concentration in the RCS; and
- Uncontrolled CRA withdrawal from a subcritical or low power.

Four channels of Subcritical Multiplication are required to be OPERABLE in MODE 1 with power less than approximately one decade above the Intermediate Range channel lower limit and in MODES 2 and 3 when capable of CRA withdrawal. In MODE 1 with power approximately one decade above the Intermediate Range channel lower limit, the Intermediate Range Log Power Rate trips and the Power Range High Linear Power trip provide protection from transients that result in high rates of change in reactor power. In MODES 2 and 3, with no capability of withdrawing any CRA, the reactor will remain subcritical. In MODES 4 and 5 the reactor is subcritical with the CRDMs and CVCS incapable of affecting the reactivity in the unit. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single failure will disable this trip Function.

The High Subcritical Multiplication trip is automatically bypassed above the N-1 interlock and is automatically enabled when the intermediate range flux decreases below the N-1 interlock. Interlock and permissive setpoints are governed by the Setpoint Program.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The Reactor Trip and ESFAS actuation of the ~~DHRS,~~ DWSI~~, CVCS isolation, and pressurizer heater breaker trip~~ by the Low Pressurizer Pressure trip function is automatically bypassed when the RCS temperature is below the T-4 interlock, and is automatically enabled when RCS temperature is above the T-4 interlock. Interlock and permissive setpoints are governed by the Setpoint Program.

c. Low Low Pressurizer Pressure – Reactor Trip, Demineralized Water System Isolation, ~~Decay Heat Removal System Actuation,~~ CVCS Isolation and Secondary System Isolation~~, Pressurizer Heater Breaker Trip~~

The Low Low Pressurizer Pressure trip is designed to protect against RCS line breaks outside of containment and protect the RCS subcooled margin against flow instability events.

The RTS and ESFAS Low Low Pressurizer Pressure setpoint is approximately 1600 psia. Actual setpoints are established in accordance with the Setpoint Program. Four Low Low Pressurizer Pressure reactor trip and DWSI channels are required to be OPERABLE when operating in MODE 1 and in MODES 2 when capable of CRA withdrawal. In MODE 2 with no capability of withdrawing any CRA, and in MODES 3, 4, and 5 the function is fulfilled because the CRAs are inserted. Four Low Low Pressurizer Pressure ~~DHRS,~~ CVCSI ~~and Pressurizer Heater Trip~~ channels are required to be OPERABLE when operating in MODES 1 and 2. In MODES 3, 4, and 5 the reactor is subcritical. Four Low Low Pressurizer Pressure Secondary System Isolation signals are required when operating in MODE 1. In MODES 2, 3, 4, and 5 the reactor is subcritical. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~when RCS temperature is above the T-2 interlock and containment water level is below the L-1 interlock. In MODE 3 with RCS temperature below the T-2 interlock or containment water level above the L-1 interlock with PASSIVE COOLING in operation, sufficient cooling for decay heat loads is met. In MODES 4 and 5 the reactor is subcritical and passively cooled.~~

The Low Low Pressurizer Level ~~DHRS~~CIS, SSI, and CVCS Isolation trip channels are automatically bypassed when the RCS temperature is below the T-2 interlock or containment water level is above the L-1 interlock. The Low Low Pressurizer Level ~~DHRS~~CIS, SSI, and CVCS Isolation trip channels are automatically enabled when RCS temperature is above the T-2 interlock and containment water level is below the L-1 interlock. Interlock and permissive setpoints are governed by the Setpoint Program.

~~d. Low RPV Riser Level – Emergency Core Cooling System Actuation~~

~~The Low RPV Riser Level trip signal provides protection for low water level above the core in LOCA events.~~

~~Four Low RPV Riser Level trip channels are required to be OPERABLE when operating in MODES 1, 2 and 3. In MODES 4 and 5 the function is fulfilled. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.~~

4. RCS Hot Temperature

Narrow Range RCS Hot Temperature is measured by three resistance temperature detectors (RTDs) per separation group (a total of 12 RTDs), located in the RCS flow near the top of the reactor vessel downcomer.

a. High Narrow Range RCS Hot Temperature – Reactor Trip, Decay Heat Removal System Actuation, Pressurizer Heater Trip, and Demineralized Water System Isolation, Secondary System Isolation

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The Low Main Steam Pressure trip causes the reactor trip breakers to open and the ~~DHRS,~~ DWSI, and SSI~~Pressurizer Heater Trip~~ to actuate.

Four Low Main Steam Pressure reactor trip, DWSI, ~~DHRS, and Pressurizer Heater~~and SSI Trip channels measuring pressure on each steam line are required to be OPERABLE when operating in MODES 1 with power range linear power above N-2H. In MODE 1 below N-2H and in MODE 2 the unit is protected by the Low Low Main Steam Pressure function. In MODES 3, 4, and 5 the reactor is subcritical. Interlock and permissive setpoints are governed by the Setpoint Program.

Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

c. Low Low Main Steam Pressure – Reactor Trip, Demineralized Water System Isolation, and Secondary System Isolation~~Decay Heat Removal System Actuation, and Pressurizer Heater Breaker Trip~~

The Low Low Main Steam Pressure trip provides protection for:

- Increase in steam flow;
- Inadvertent opening of the turbine bypass system;
- Loss of feedwater flow;
- Steam system piping failures inside and outside the containment vessel; and
- Feedwater system pipe breaks inside and outside the containment vessel.

The Low Low Main Steam Pressure trip causes the reactor trip breakers to open and the ~~DHRS,~~ DWSI~~,~~ and SSI ~~Pressurizer Heater Breaker Trip~~ to actuate.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Four Low Low Main Steam Pressure reactor trip, ~~and~~ DWSI, and SSI channels measuring pressure on each steam line are required to be OPERABLE when operating in MODE 1 and MODE 2 when capable of CRA withdrawal. In MODE 2 with no capability of withdrawing any CRA and in MODES 3, 4, and 5 the reactor is subcritical.

Four Low Low Main Steam Pressure ~~DHRS~~ SSI Trip channels are required to be OPERABLE when operating in MODES 1 and 2. Protection from low main steam pressure is not required in MODES 3, 4, and 5.

~~Four Low Low Main Steam Pressure Pressurizer Heater Breaker Trip channels are required to be OPERABLE in MODE 1 and MODE 2 when pressurizer heater breakers are closed. In MODE 2 with pressurizer heater breakers open and in MODES 3, 4, and 5 the function is fulfilled.~~

The Low Low Main Steam Pressure SSI~~DHRS~~ channels are automatically bypassed when water level is above the L-1 interlock~~ and the RTBs are open~~. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. Interlock and permissive setpoints are governed by the Setpoint Program.

7. Steam Superheat

Steam Superheat is determined by MPS SFM processing of main steam temperature and pressure data. Steam pressure sensors are shared between the High and Low Main Steam Pressure trips and are used as input to the High and Low Steam Superheat trips. Four steam temperature sensors are located on each steam pipe upstream of the MSIVs. Each channel of superheat receives two steam generator pressure inputs and two steam temperature inputs (one pressure and one temperature signal from each steam line). The degree of superheat is found by determining the saturation temperature (T_{SAT}) at the measured main steam pressure (P_{STM}), and subtracting this value from the measured main steam temperature (T_{STM}). The main steam saturation temperature is found via a simple steam table lookup function using the measured steam pressure value.

$$T_{SH} = T_{STM} - T_{SAT}(P_{STM})$$

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The Low Steam Superheat SSI~~DHRS and Pressurizer Heater Trip~~ actuation is automatically bypassed when containment water level is above the L-1 interlock or 1 FWIV is closed with power less than the N-2H setpoint. Interlock and permissive setpoints are governed by the Setpoint Program.

8. Containment Pressure

Narrow Range Containment pressure is measured by four sensors (one per separation group) located near the top of the containment vessel.

a. High Narrow Range Containment Pressure – Reactor Trip, Demineralized Water System Isolation, Containment Isolation, Secondary System Isolation,~~Decay Heat Removal System Actuation, Pressurizer Heater Trip,~~ and CVCS Isolation

The High Containment Pressure trip provides protection for:

- System malfunctions that increase the RCS inventory;
- Inadvertent operation of the ECCS;
- Loss of containment vacuum;
- Steam system piping failures inside and outside the containment vessel;
- Feedwater system pipe breaks inside and outside the containment vessel; and
- Loss-of-coolant accidents from a spectrum of postulated piping breaks inside the containment vessel.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The High Narrow Range Containment Pressure trip causes the reactor trip breakers to open, the containment to be isolated, the ~~DHRS and Pressurizer Heater Trip~~SSI to be actuated, and the DWS and CVCS to be isolated.

Four High Narrow Range Containment Pressure reactor trip and DWSI channels are required to be OPERABLE when operating in MODE 1 and MODES 2 and 3 when capable of CRA withdrawal.

Four High Narrow Range Containment Pressure SSI~~DHRS~~ channels are required to be OPERABLE in MODES 1 and 2, and MODE 3 without PASSIVE COOLING in operation. In MODE 3 with PASSIVE COOLING in operation, sufficient cooling for decay heat loads is met. In MODES 4 and 5 the reactor is subcritical and passively cooled.

~~Four Pressurizer Heater Trip channels are required to be OPERABLE when operating in MODE 1 and in MODES 2 and 3 with the pressurizer heater trip breakers closed. In MODES 2 and 3 with the pressurizer heater trip breakers open and in MODES 4 and 5 this function is fulfilled.~~

Four High Containment Pressure CVCSI and CIS channels are required to be OPERABLE when operating in MODES 1 and 2, and MODE 3 with RCS temperature above the T-3 interlock. In MODE 3 with RCS temperature is below the T-3 interlock, and in MODES 4 and 5 the containment pressure is allowed to exceed this setpoint and is expected, isolation is not required.

Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function.

The High Containment Pressure Containment Isolation, SSI, ~~DHRS, Pressurizer Heater Trip,~~ and CVCSI actuations are automatically bypassed when RCS temperature is below the T-3 interlock. The High Containment Pressure ~~DHRS and Pressurizer Heater Trip actuation~~SSI is also automatically bypassed when containment water level is above the L-1 interlock. Interlock and permissive setpoints are governed by the Setpoint Program.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Containment Water Level

The High Containment Water Level trip signal causes ECCS actuation. Four ECCS High Containment Water Level trip channels are required to be OPERABLE when operating in MODES 1 and 2, and MODE 3 with RCS hot temperature above T-3 or PZR level below L-2. In MODE 3 with RCS hot temperature below T-3 and PZR level above L-2, and MODES 4 and 5 the function is fulfilled. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The high containment water level ECCS actuation is automatically bypassed when RCS temperature is below the T-3 interlock and PZR level is above L-2, and automatically enabled when RCS temperature is above the T-3 interlock or PZR level is below L-2. Interlock and permissive setpoints are governed by the Setpoint Program. ~~Containment Water Level is measured by 4 sensors (one per separation group) located in the containment vessel. The level is measured by a radar instrument which will run the entire distance of the measurement, from the containment head to an elevation below 45 ft.~~

~~a. High Containment Water Level – Emergency Core Cooling System Actuation~~

~~The High Containment Water Level trip provides protection for LOCA events.~~

~~The High Containment Water Level trip signal causes ECCS actuation. Four ECCS High Containment Water Level trip channels are required to be OPERABLE when operating in MODES 1 and 2, and MODE 3 without PASSIVE COOLING in operation. In MODE 3 with PASSIVE COOLING in operation, and MODES 4 and 5 the function is fulfilled. Four channels are provided to permit one channel in trip or bypass indefinitely and still ensure no single random failure will disable this trip Function. The high containment water level ECCS actuation is automatically bypassed when RCS temperature is below the T 3 interlock and the RTBs open, and automatically enabled when RCS temperature is above the T 3 interlock.~~

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Wide Range RCS Pressure and Wide Range RCS Cold Temperature

Wide range RCS pressure is measured to determine the RCS pressure, as represented by the steam space near the top of the reactor vessel. The MPS is supplied signals from four sensors (one for each separation group) that measure pressure from about 0 to 2500 psia.

Wide range RCS cold temperature is measured to determine a representative minimum temperature in the RCS as measured at four locations in the lower downcomer region of the reactor vessel. The MPS is supplied signals from four sensors (one for each separation group) that measure temperature from about 40 to 700°F.

a. High RCS Pressure – Low Temperature Overpressure Protection (LTOP)

The High RCS Pressure – Low Temperature trip provides protection for low temperature overpressure events.

The High RCS Pressure – Low Temperature trip signal causes the reactor vessel vent valves to open.

Four High RCS Pressure – Low Temperature trip channels are required to be OPERABLE when operating in MODE 3 with wide range RCS cold temperature below the LTOP enable temperature specified in the PTLR (T-1 Interlock) and more than one reactor vent valve closed. In MODES 1 and 2 the reactor vessel is at a higher temperature and overpressure protection is provided by the safety valves and the DHRS. In MODE 3 with two RVVs open, and MODES 4 and 5 the reactor vessel is protected from overpressure by the openings that exist between the reactor vessel and the containment or the conduction of heat between the reactor vessel and the refueling pool. The LTOP function is automatically bypassed when wide range RCS cold temperature is above the T-1 interlock and automatically enabled when wide range RCS cold temperature is below the T-1 interlock. Interlock and permissive setpoints are governed by the Setpoint Program.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9642
Date of RAI Issue: 03/20/2019

NRC Question No.: 16-63

Paragraph (a)(11) of 10 CFR 52.47 and paragraph (a)(30) of 10 CFR 52.79 state that a design certification (DC) applicant and a combined license (COL) applicant, respectively, are to propose technical specifications (TS) prepared in accordance with 10 CFR 50.36 and 50.36a. 10 CFR 50.36 sets forth requirements for TS to be included as part of the operating license for a nuclear power facility.

It is customary and prudent for a design certification application to provide generic TS Bases along with the generic TS, which are required to be provided in a design certification application by 10 CFR 52.47 and 10 CFR 50.36(a)(2). The Bases should be consistent with the proposed design.

In the LCO 3.0.4 Bases, the applicant is requested to consider the following staff suggested NuScale design-specific paragraph change, as follows (see Rev 2 of DCA Part 4, page B 3.0-7:

> The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, ~~and~~ MODE 2 to MODE 3 and not PASSIVELY COOLED, and not PASSIVELY COOLED to PASSIVELY COOLED.

In the SR 3.0.4 Bases, the applicant is requested to consider the following staff suggested NuScale design-specific paragraph change, as follows (see Rev 2 of DCA Part 4, page B 3.0-21:

> The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, ~~and~~ MODE 2 to MODE 3 <u>and not PASSIVELY COOLED, and not PASSIVELY COOLED to</u> <u>PASSIVELY COOLED</u>.

The staff identified these apparent oversights while verifying the changes in response to RAI 157-9033, Question 16-15 had been incorporated in
Subsection B 3.0 of Revision 2 of DCA Part 4.

NuScale Response:

The examples provided in the Bases were developed to parallel the corresponding examples in standard technical specifications (STS). The NuScale and STS bases provide introductory statements that explicitly includes consideration of 'other specified conditions."

> The provisions of LCO 3.0.4 shall not prevent changes in MODES **or other specified conditions** in the Applicability that are required to comply with ACTIONS... (Emphasis added.)

The Bases of LCO 3.0.4 have been modified to address the condition identified by the staff.

Impact on DCA:

The Technical Specifications have been been revised as described in the response above and as shown in the markup provided in this response.

BASES

LCO 3.0.4 (continued)

provide for continued operation for an unlimited period of time and a risk assessment has not been performed. This allowance may apply to all the ACTIONS or to a specific Required Action of a Specification. The risk assessments performed to justify the use of LCO 3.0.4.b usually only consider systems and components. For this reason, LCO 3.0.4.c is typically applied to Specifications which describe values and parameters (e.g., RCS Specific Activity) and may be applied to other Specifications based on NRC unit-specific approval.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, ~~and~~ MODE 2 to MODE 3, and MODE 3 not PASSIVELY COOLED to MODE 3 PASSIVELY COOLED.

Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, utilizing LCO 3.0.4 is not a violation of SR 3.0.1 or SR 3.0.4 for any Surveillances that have not been performed on inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

BASES

SR 3.0.4 (continued)

removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes. SR 3.0.4 does not restrict changing MODES or other specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the LCO not met has been delayed in accordance with SR 3.0.3.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown. In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, ~~and~~ MODE 2 to MODE 3, and MODE 3 not PASSIVELY COOLED to MODE 3 PASSIVELY COOLED.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into a MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO’s Applicability, would have its Frequency specified such that it is not “due” until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note, as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs’, annotation is found in Section 1.4, “Frequency.”

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9642
Date of RAI Issue: 03/20/2019

NRC Question No.: 16-64

Paragraph (a)(11) of 10 CFR 52.47 and paragraph (a)(30) of 10 CFR 52.79 state that a design certification (DC) applicant and a combined license (COL) applicant, respectively, are to propose technical specifications (TS) prepared in accordance with 10 CFR 50.36 and 50.36a. 10 CFR 50.36 sets forth requirements for TS to be included as part of the operating license for a nuclear power facility.

Parts of this question are a followup of RAI 197-9051, Question 16-28.

The staff requests the applicant to address the following concerns not already addressed by the planned supplemental response to RAI 197-9051, Question 16-28:

(1) To ensure the SP will govern all Channel Calibration SRs, each Channel Calibration Surveillance statement needs to append the phrase “in accordance with the Setpoint Program.”

(2) LCO 3.3.4 needs to specify a Channel Calibration for the Class 1E isolation devices associated with the manual RTS and ESF actuation Functions.

(3) The applicant needs to provide additional justification for why the surveillance column Notes for SR 3.3.1.5, SR 3.3.2.3, and SR 3.3.3.3 are needed. Specifically, address the expected operational restrictions or burdens that would be avoided by invoking the Note. Also, explain how the action requirements would be applied if an associated Class 1E isolation device is known to be unable to open on an OC or UV condition for an MPS Function, an RTS Function, an ESFAS Function, and a manual Function.

(4) The applicant needs to address the expected operational restrictions that would be avoided by invoking the exception to meeting the automatic actuation verification Survellance for each valve and trip breaker specified by the SRs quoted below in the background discussion.

(5) In STS, since an Actions table Note is usually used to specify an allowance to open (or close) a valve (or circuit breaker), which is closed (or open) to comply with a Required Action, provided the valve is operated using administrative controls (which are usually defined and described in the Bases discussion of the Note), the applicant needs to explain the need for specifying such an exception in a Surveillance statement, such as proposed in SR 3.1.9.2, SR 3.4.6.3, SR 3.6.2.2, SR 3.6.2.3, and SR 3.6.2.4; or in a surveillance column Note, such as proposed in SR 3.3.3.2, SR 3.3.3.4, and SR 3.4.6.2.

(6) The applicant needs to resolve the apparent error noted below in the background discussion about listing LCO 3.5.2, LCO 3.7.1, and LCO 3.7.2 in the discussion of TSTF-541 in Table C-1 of RCDR Revision 1.

Background Discussion:

The TSTF submitted unapproved traveler TSTF-541, Revision 0, “Add Exceptions to Surveillance Requirements When the Safety Function is Being Performed,” for NRC review on September 10, 2013 (ML13253A390). The traveler’s proposed changes would provide exceptions to certain SRs for ventilation system dampers and cooling water system valves that are in certain positions. For example, the traveler proposes to revise the quoted SRs of the following W-STS Subsections, by adding text denoted by *italics*:

- W-STS Subsection 3.6.11, “Iodine Cleanup System (ICS)”
    - SR 3.6.11.3 *[-----------------------------NOTE-----------------------------*
      Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
      ---]

      Verify each ICS train actuates on an actual or simulated actuation signal.

- W-STS Subsection 3.7.10, “Control Room Emergency Filtration System (CREFS)”
    - SR 3.7.10.3 *[-----------------------------NOTE-----------------------------*
      Not required to be met for dampers and valves locked,

sealed or otherwise secured in the actuated position.
---]

Verify each CREFS train actuates on an actual or simulated actuation signal.

- W-STS Subsection 3.7.12, "ECCS Penetration Room Exhaust Air Cleanup System (PREACS)"
  - SR 3.7.12.3 *[------------------------------NOTE------------------------------*
    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

    Verify each ECCS PREACS train actuates on an actual or simulated actuation signal.

- W-STS Subsection 3.7.13, "Fuel Building Air Cleanup System (FBACS)"
  - SR 3.7.13.3 *[------------------------------NOTE------------------------------*
    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

    Verify each FBACS train actuates on an actual or simulated actuation signal.

- W-STS Subsection 3.7.14, "Penetration Room Exhaust Air Cleanup System (PREACS)"
  - SR 3.7.14.3 *[------------------------------NOTE------------------------------*
    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

    Verify each PREACS train actuates on an actual or simulated actuation signal.

In addition to CE-STS Subsections equivalent to these W-STS Subsections, the traveler also proposes to revise the quoted SR of the following CE-STS Subsection:

- CE-STS Subsection 3.7.10, “Essential Chilled Water (ECW)”
  - SR 3.7.10.2 *[-----------------------------NOTE-----------------------------*
    Not required to be met for valves locked, sealed or otherwise secured in the actuated position.
    --]

    Verify the proper actuation of each ECW System component on an actual or simulated actuation signal.

Similar changes are proposed for the B&W-STS, GE-BWR4-STS, and GE-BWR6-STS.

In a letter dated February 25, 2016 (ML16012A427), for the second time, the staff asked the TSTF for additional information about TSTF-541, Revision 0. This letter contained 15 information requests from three technical branches and 5 information requests from the technical specifications branch. As of April 2018, with the response to the letter still pending, the TSTF was planning to submit a revision to the traveler to clarify its scope and intent.

The entry for this traveler in Table C-1, “TSTF traveler evaluation,” of DCDR, Revision 0, indicates its adaptation to the NuScale GTS would affect Subsection 3.6.2, “Containment Isolation Valves,” and states the following:

> The passive NuScale design includes a limited number of valves with potential for the addressed condition to exist. Exceptions consistent with the traveler were incorporated into the surveillance requirements of [GTS Subsection] 3.6.2.

The staff compared the changes proposed in TSTF-541 with SR 3.6.2.3 of Revision 0 of DCA Part 4, which states the following (emphasis added):

> SR 3.6.2.3 Verify each automatic containment isolation valve *that is not locked, sealed, or otherwise secured in position*, actuates to the isolation position on an actual or simulated actuation signal.

This surveillance statement appears to include the intent of the bracketed surveillance column Notes proposed by the traveler for similar automatic valve actuation SRs in STS, as quoted above.

Compared to these Notes, the GTS surveillance statement uses the phrase "locked, sealed, or otherwise secured in position" instead of "locked, sealed, or otherwise secured in *the actuated* position."

In Revisions 1 and 2 of DCA Part 4, SR 3.6.2.3 is numbered SR 3.6.2.4, and includes an additional phrase, indicated by underline below, that provides an exception to when the Surveillance must be met:

SR 3.6.2.4	Verify each automatic containment isolation valve *that is not locked, sealed, or otherwise secured in position,* actuates to the isolation position on an actual or simulated actuation signal <u>except for valves that are open under administrative controls</u>.

In Revision 0 of DCA Part 4, the Bases for SR 3.6.2.3 contain the first sentence of the passage quoted below. This statement also appears similar to the intent of the traveler's bracketed surveillance column Note (emphasis added). In Revisions 1 and 2 of DCA Part 4, the Bases for SR 3.6.2.4 (as renumbered beginning in DCA Revision 1) also includes the second sentence of this passage, regarding the added exception, that states the following:

> The Surveillance is not required for valves that are locked, sealed, or otherwise secured in the *required* position *under administrative controls*. An exception to the SR is also provided for valves that are open under administrative control.

Compared to the traveler's surveillance column bracketed Note, the first of the above GTS Bases sentences uses the phrases (1) "not required for valves" instead of "not required <u>to be met</u> for <u>automatic</u> valves"; and (2) "required position" instead of "<u>actuated</u> position." The first sentence also includes the phrase "under administrative controls," which inappropriately specified an implied exception to meeting the valve position verification requirement. The addition of an explicit exception to meeting the valve position verification requirement in Revision 1 of the SR and the associated Bases statement, corrected this inappropriate use of the Bases to modify the applicability of a Surveillance. Since the phrasing of the surveillance statement and associated Bases differ only in presentation from the traveler's proposed bracketed surveillance column Note, the staff concludes that the exception to meeting SR 3.6.2.4 is editorially consistent with the intent of TSTF-541, Revision 0. However, since this traveler does not address exceptions to automatic containment isolation valve testing, the staff concludes that the proposed exception to meeting SR 3.6.2.4 is not supported by the intended scope of the traveler.

In Revision 1 of DCA Part 4, the applicant included the following SRs for verifying [automatic] valve actuation on an "actual or simulated [actuation] signal," to the position stated in the Surveillance, which is also quoted below, on a Frequency of "In accordance with the Surveillance Frequency Control Program." (Note that the SR enumeration is that of Revision 2 of DCA Part 4. Also, listed here are the Surveillance base Frequencies, which are given in the latest revision of DCA Part 2, Table 16.1-1.)

SR 3.1.9.2	CVCS demineralized water isolation valves	isolation position	24 months
SR 3.4.6.3	CVCS automatic [isolation] valves	isolation position	24 months
SR 3.4.10.1	LTOP RVVs	open position	24 months
SR 3.5.1.1	ECCS RVVs and RRVs	open position	24 months
SR 3.6.2.4	automatic containment isolation valves	isolation position	24 months

In each of these SRs, (1) the surveillance statement includes either the phrase, "except for valves that are open under administrative controls," or the phrase, "that is not locked, sealed, or otherwise secured in [the isolated] position," or (2) the SR includes a surveillance column Note or surveillance table Note that states, "Not required to be met for valves that are open."

Because the staff has not approved TSTF-541, Revision 0, and considering the above observations, the staff could not determine whether the proposed application of the intent of the traveler is needed for the NuScale GTS.

The staff issued RAI 197-9051 (ML17237C008), Question 16-28, with the above observations and to request that the applicant revise the evaluation of TSTF-541 in RCDR Table C-1 to indicate that it is not applicable to automatic valve actuation surveillances in the GTS; (i.e., withdraw allowances based on the intent of the traveler for valves that are not within the scope of the traveler).

In its response (ML17291A299) to Question 16-28, the applicant stated the following (emphasis added):

> NuScale has adopted and incorporated the conceptual basis of TSTF traveler 541 into the proposed GTS *independent of the outcome of the TSTF-NRC traveler review and approval activities*.

The conceptual basis and underlying issue is that as described in Section SR 3.0, if a surveillance requirement cannot be met, then the associated LCO is not being met and the associated Condition must be entered. However in some circumstances *the*

OPERABILITY of the equipment that cannot meet the surveillance requirement *is being met* because the safety function has been met. *A commonly used example is a valve that is in the position to perform its safety function, and is not assumed to move following actuation.*

The NuScale safety systems include various valves and breakers that align to a particular position to perform their specified safety function. In each case, the valve or breaker has a single credited actuation position - the design does not include any valve or breakers that must move to alternative positions after they have performed their required safety function.

The staff observes that the above response appears consistent with the rationale of a Reviewer's Note that the traveler proposed to add to the Bases for each affected STS SR. This Reviewer's Note states the following:

----------------------------------REVIEWER'S NOTE----------------------------------
Adoption of the Note excluding valves that are locked, sealed, or otherwise secured in the actuated position requires confirmation by the licensee that movement of the valves following an accident is not assumed in the safety analysis.

The traveler also proposed to insert the following passage in the Bases of each affected SR:

[The SR is modified by a Note excluding valves that are locked, sealed, or otherwise secured in the actuated position. It is not necessary to test valves that are locked, sealed, or otherwise secured in the actuated position because the affected valves were verified to be in the actuated position assumed in the accident analysis prior to being locked, sealed, or otherwise secured, and because movement following an accident is not assumed in the accident analysis.]

The above quoted response to Question 16-28 also appears consistent with this passage. However, the staff observes that the applicant's proposed exceptions to meeting selected SRs, which verify automatic valve and breaker actuation, are not always specified by a surveillance column Note. Several of the proposed exceptions are specified by inserting exception language in the surveillance statement instead of in a Note.

The response to Question 16-28 proposed to broadly apply the traveler's "conceptual basis" to selected SRs for (1) CVCS demineralized water isolation valves – when closed; (2) Class 1E isolation devices for MPS instrumentation Function channels – when opened; (3) Class 1E isolation devices for RTS and ESFAS actuation logic divisions – when opened; (4) reactor trip breakers – when opened; (5) pressurizer heater trip breakers – when opened; (6) CVCS isolation valves – when closed; (7) ECCS valves (RRVs and RVVs) – when opened; (8) LTOP RVVs – when opened; (9) containment isolation valves – when closed; (10) decay heat removal actuation valves – when opened; (11) main steam isolation valves and main steam isolation bypass valves – when closed; and (12) feedwater isolation valves and feedwater regulation valves – when closed.

In Technical Report (TR)-1116-52011-NP, "Technical Specifications Regulatory Conformance and Development," Revision 1 (RCDR) (ML18304Annn), the applicant revised Table C-1 to address draft Revision 1 of TSTF-541, dated May 29, 2018; as of February 14, 2019, formal submission of this revised traveler for staff review was pending. The revised Table C-1 states the following:

> Although not directly applicable, the Intent of the traveler was adopted in the NuScale GTS. NuScale safety-related reactor trip system and ECCS components are credited with a single safety-related position, each of which is achieved by the component being de-energized.
>
> The implementation of this traveler is under additional review and consideration as requested by the NRC staff at the time this technical report was developed. See RAI [197-9051, Question] 16-28.

In RCDR Revision 1, Table C-1 listed the affected GTS Specifications influenced by TSTF-541, draft Revision 1. These Specifications, along with the affected SRs, are listed below. The staff notes that Specification 3.3.1, "MPS Instrumentation," should also be included in the list because the supplemental response (ML18355Annn) to RAI 196-9050, Question 16-17, revised Subsection 3.3.1 of Revision 2 of DCA Part 4 so that SR 3.3.1.5 specifies performing a Channel

Calibration of Class 1E isolation devices for MPS instrumentation components. The supplemental response to Question 16-17 similarly revised SR 3.3.2.3 and SR 3.3.3.3; the revised SRs are quoted below.

In the below quotations of the affected SRs, the Frequencies are abbreviated: “IAW SFCP” stands for “In accordance with the Surveillance Frequency Control Program”; and “IAW ISTP” stands for “In accordance with the INSERVICE TESTING PROGRAM.” Staff suggested clarification edits are indicated by shaded mark up.

3.1.9, Boron Dilution Control

SR 3.1.9.2 Verify each automatic CVCS demineralized water isolation valve that is not...secured in the isolated position, actuates to the isolated position on an actual or simulated actuation signal *except for valves that are open under administrative controls*. | IAW SFCP

3.3.1, MPS Instrumentation

SR 3.3.1.5 -----------------------------NOTE----------------------------------
Not required to be met for Class 1E isolation devices that have isolated 1E circuits from non-1E power.

Perform CHANNEL CALIBRATION on each ~~required~~ Class 1E isolation device. | IAW SFCP

3.3.2, Reactor Trip System Logic and Actuation

SR 3.3.2.1 -----------------------------NOTE----------------------------------
Not required to be met for reactor trip breakers that are open.

Perform ACTUATION LOGIC TEST. | IAW SFCP

SR 3.3.2.2 -----------------------------NOTE----------------------------------
Not required to be met for reactor trip breakers (RTBs) that are open.

Verify ~~required~~ response time is within limits. | IAW SFCP

SR 3.3.2.3 -----------------------------NOTE----------------------------------
Not required to be met for Class 1E isolation devices that have isolated 1E circuits from non-1E power.

Perform CHANNEL CALIBRATION on each ~~required~~ Class 1E isolation device. | IAW SFCP

SR 3.3.2.4 -----------------------------NOTE----------------------------------
Not required to be met for reactor trip breakers that are open.

Verify each RTB actuates to the open position on an actual or simulated actuation signal. | IAW SFCP

3.3.3, Engineered Safety Feature Actuation System Logic and Actuation

SR 3.3.3.2 -----------------------------NOTE----------------------------------
Not required to be met for pressurizer heater trip breakers that are open or closed under ~~manual~~ ~~control~~ administrative controls.

Verify ~~required~~ pressurizer heater trip breaker response time is within limits. | IAW SFCP

SR 3.3.3.3 ----------------------------NOTE----------------------------------
Not required to be met for Class 1E isolation devices that have isolated 1E circuits from non-1E power.

Perform CHANNEL CALIBRATION on each ~~required~~ Class 1E isolation device. | IAW SFCP

SR 3.3.3.4 ----------------------------NOTE----------------------------------
Not required to be met for pressurizer heater trip breakers that are open or ~~breakers~~ closed under administrative controls.

Verify each pressurizer heater trip breaker ~~(PHTB)~~ actuates to the open position on an actual or simulated actuation signal. | IAW SFCP

3.4.6, Chemical and Volume Control System Isolation Valves

SR 3.4.6.2 ----------------------------NOTE----------------------------------
Not required to be met for valves that are closed or open under administrative controls.

Verify the ~~required~~ isolation time of each automatic power operated CVCS valve is within limits. | IAW ISTP

SR 3.4.6.3 Verify each automatic CVCS valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal *except for valves that are open under administrative controls*. | IAW SFCP

3.4.10, Temperature Overpressure Protection Valves

In Revision 2 of DCA Part 4, the Subsection 3.4.10 SR table Note (“Not required to be met for valves that are open.”) provides an explicit exception to meeting the SRs for the RVVs, and the RVV inadvertent actuation block, when an RVV is open. In its response (ML18346Annn) to RAI 506-9614 (ML18278Annn), Question 16-53, Sub-question B, regarding the LTOP function of the RVVs and LCO 3.4.10, the applicant stated:

> An RVV that is not closed has completed its safety function and providing a vent path and no further action or actuation is required. Therefore any RVV not closed is outside the scope of required components in this LCO.
>
> An editorial correction has been made by removal of the Note at the Surveillance Requirements table. *The Note was removed as unnecessary because the LCO only applies to closed reactor vent valves.*

This exception provided by the Note, however, is also implied by LCO 3.4.10, which states, “Each *closed* reactor vent valve (RVV) shall be OPERABLE.”

The staff concludes that the SRs only support the operability of the LTOP automatic open function of three closed RVVs.

3.5.1, Emergency Core Cooling System

SR 3.5.1.1 -----------------------------NOTE----------------------------------
Not required to be met for valves that are open.

Verify each RVV and RRV actuates to the open position on an actual or simulated actuation signal. | IAW SFCP

SR 3.5.1.2 -----------------------------NOTE----------------------------------

Not required to be met for valves that are open.

Verify the open actuation time of each RVV and RRV is within limits. | IAW ISTP

Subsection 3.5.1 specifies SR 3.5.1.3 ("Verify the inadvertent actuation block function of each RVV and RRV is OPERABLE. | IAW SFCP") and SR 3.5.1.4 ("Verify the inadvertent actuation block setpoint is within limits for each RVV and RRV. | IAW ISTP"). However, there is no explicit exception to meeting these SRs "for valves that are open."

3.5.2, Decay Heat Removal System

Subsection 3.5.2 specifies SR 3.5.2.3 ("Verify that each DHRS actuation valve actuates to the open position on an actual or simulated actuation signal. | IAW SFCP") and SR 3.5.2.4 ("Verify the open actuation time of each DHRS actuation valve is within limits. | IAW ISTP"). However, there is no explicit exception to meeting these SRs "for DHRS actuation valves that are open" specified in Revision 2 of DCA Part 4. Therefore, it appears that including LCO 3.5.2 in the list of affected LCOs in Revision 1 of DCDR Table C-1 is an error.

3.6.2, Containment Isolation Valves

SR 3.6.2.2 Verify each containment isolation manual valve...that is...required to be closed...is closed, *except for containment isolation valves that are open under administrative controls.* | IAW SFCP

SR 3.6.2.3 Verify the isolation time of each automatic containment isolation valve is within limits *except for valves that are open under administrative controls*. | IAW ISTP

SR 3.6.2.4 Verify each automatic containment isolation valve... actuates to the isolation position on an actual or simulated actuation signal *except for valves that are open under administrative controls*. | IAW SFCP

3.7.1, Main Steam Isolation Valves

Subsection 3.7.1 specifies SR 3.7.1.2 ("Verify isolation time of each MSIV and MSIV bypass valve is within limits on an actual or simulated actuation signal. | IAW ISTP"). However, there is no explicit exception to meeting this SR "for isolation valves that are closed" specified in Revision 2 of DCA Part 4. Therefore, it appears that including LCO 3.7.1 in the list of affected LCOs in Revision 1 of DCDR Table C-1 is an error.

3.7.2, Feedwater Isolation

Subsection 3.7.2 specifies SR 3.7.2.2 ("Verify the closure time of each FWIV and FWRV is within limits on an actual or simulated actuation signal. | IAW ISTP"). However, there is no explicit exception to meeting this SR "for isolation and regulation valves that are closed" specified in Revision 2 of DCA Part 4. Therefore, it appears that including LCO 3.7.2 in the list of affected LCOs in Revision 1 of DCDR Table C-1 is an error.

NuScale Response:

(1) This issue has been addressed in the response to RAI 16-62.

(2) As described in FSAR Section 7.0.4.1.4 and shown on FSAR figure 7.0-11a and 7.0-11b, the power supply to logic devices that implement the manual Reactor Trip System (RTS) and Engineered Safety Features (ESF) actuation functions are those that supply, and are tested in the corresponding calibrations required as SR 3.3.2.3 and SR 3.3.3.3 respectively. There are no additional Class 1E isolation devices associated with the RTS and ESF actuation functions that require testing. Therefore no separate Class 1E isolation device CHANNEL CALIBRATIONS are required.

(3) The basis for the Notes in the Surveillance column of SR 3.3.1.5, SR 3.3.2.3, and SR 3.3.3.3 is that when a Class 1E circuit supplying the associated portion of the module protection system (MPS) is isolated from the non-Class 1E circuit, the isolation device has no further credited function. This Note provides operational flexibility to address unidentified conditions

that may exist in future operations while ensuring the credited safety function has been accomplished. 'Operational restrictions' and 'burdens' are not the basis for the Notes.

(4) NuScale is not implementing TSTF-541, and was not a party to its development. The 'Background Discussion' section of the TSTF as quoted below is unrelated to the NuScale design, operations, or proposed specifications. The TS are not submitted for consideration based on deviation or changes from existing STS because no STS exist that are consistent with the NuScale design. The content of the TS is based on the requirements of 10 CFR 50.36 as described in FSAR Chapter 16.

The TS were developed and submitted based on the unique NuScale design and operational paradigm. Contents, applicability, and implementation are based on those factors. As previously stated in the responses to RAI 16-28, the exceptions to the automatic actuation verification are based only on the NuScale design and operations - specifically that each credited device has only a single credited position or function. When a credited device is actuated, it has completed and satisfied its credited safety function. This design is unlike similarly-named functions in legacy designs.

Performing a surveillance on a credited NuScale component that has actuated to its safety function would require the component to be placed in its unactuated configuration, thereby removing it from credited alignment. While unlikely, if the component failed to actuate, the safety function would be jeopardized *merely to perform a test to verify the capability for action that has already occurred,* thereby creating a small, but real possibility for new initiating event and failure. Not performing a surveillance test on a component that is already in its only actuated position and has performed it safety function ensures that no additional potential initiating failure can occur. In effect, requiring a surveillance test on these components increases the likelihood that the associated safety function may not be met.

SR 3.0.4 and LCO 3.0.4 adequately ensure that surveillance requirements will be met when the Applicable MODE or other specified condition require it.

(5) As previously noted, the NuScale design and operations are not consistent with any STS. The NuScale specifications are written based on the NuScale design and operations.

The allowances in the listed SRs permit surveillance requirements to be deferred when the safety functions have been satisfied, but continued operations or other preferred operational methods using normal means requires alignment of the components to their non-safety actuated position, contrary to LCO 3.0.5. This is a reasonable allowance with the use of administrative controls as widely used across the commercial nuclear industry. Explanation of circumstances or examples of specific postulated circumstances that are addressed by the allowances in the identified surveillance requirements are provided below.

SR 3.1.9.2

Performing a surveillance on a credited NuScale component that has actuated to its safety function would require the component to be placed in its unactuated configuration, thereby removing it from credited alignment. While unlikely, if the component failed to actuate, the safety function would be jeopardized merely to perform a test to verify the capability actuate that has already occurred, thereby creating a new initiating failure. Not performing a surveillance test on a component that is already in its only actuated position and has performed it safety function ensures that no additional potential initiating failure can occur. In effect, requiring a surveillance test on these components increases the likelihood that the associated safety function may not be met.

SR 3.3.3.2 and SR 3.3.3.4

The pressurizer heater breaker safety-related function is to isolate the heaters from their power source to ensure the integrity of the reactor coolant pressure boundary if the heaters are uncovered. The MPS provides a trip function on lowering pressurizer level that removes power to the heaters prior to pressurizer level reaching the top of the pressurizer heaters. The safety-related function of pressurizer heater circuit breakers is not to ensure an electrical power supply to the pressurizer heaters, as described in FSAR Section 5.4.5.2.

The allowances in SR 3.3.3.2 and SR 3.3.3.4 permit breaker operation without satisfying the surveillance requirements while under administrative control. Although pressurizer heater function is not credited or required to respond to design basis accidents, they provide important operational RCS pressure control capability during normal operations and shutdown. If the allowance was not provided, the operation of the breakers during continued operations while performing repairs, or during a normal shutdown evolution would violate the requirements of LCO 3.0.5 which indicates that equipment removed from service may be returned to service *solely* to perform testing required to demonstrate OPERABILITY.

SR 3.4.6.2

An allowance is provided to permit opening the valves under administrative controls. The valves are credited to respond to certain design basis accidents by closing. However opening the valves provides important operational capability such as boron injection, dilution, and RCS inventory control during normal operations and shutdown that require the valves to be opened. If the allowance was not provided, the operation of the valves during continued operations such as while performing repairs, or during a normal shutdown evolution, would violate the requirements of LCO 3.0.5 which indicates that equipment removed from service may be returned to service *solely* to perform testing required to demonstrate OPERABILITY.

SR 3.4.6.3

Performing a surveillance on a credited NuScale component that has actuated to its safety function would require the component to be placed in its unactuated configuration, thereby removing it from credited alignment. While unlikely, if the component failed to actuate, the safety function would be unnecessarily defeated *merely to perform a test* thereby creating a new initiating failure. Not performing a surveillance test on a component that is already in its only actuated position and has performed it safety function ensures that no additional potential initiating failure can occur. In effect, requiring a surveillance test on these components increases the likelihood that the associated safety function may not be met.

SR 3.6.2.2, SR 3.6.2.3, and SR 3.6.2.4

Performing a surveillance on a credited NuScale component that has actuated to its safety function would require the component to be placed in its unactuated configuration, thereby removing it from credited alignment. While unlikely, if the component failed to actuate, the safety function would be unnecessarily defeated *merely to perform a test* thereby creating a new initiating failure. Not performing a surveillance test on a component that is already in its only actuated position and has performed it safety function ensures that no additional potential initiating failure can occur. In effect, requiring a surveillance test on these components increases the likelihood that the associated safety function may not be met.

An allowance is provided for each of these surveillances to permit opening the valves under administrative controls. The valve functions are credited to respond to design basis accidents by closing. However opening the valves provides important operational capability such as boron injection, dilution, and RCS inventory control during normal operations and shutdown that require the valves to be opened. If the allowance was not provided, the operation of the valves during continued operations such as while performing repairs, or during a normal shutdown evolution, would violate the requirements of LCO 3.0.5 which indicates that equipment removed from service may be returned to service *solely* to perform testing required to demonstrate OPERABILITY. The SRs were modified to more consistently describe these allowances.

(6) Reference to LCO 3.5.2, 3.7.1, and 3.7.2 will be removed from the NuScale Specifications Affected column of TR-1116-52011, Technical Specifications Regulatory Conformance and Development Report, Table C-1, traveler 541 entry in the next revision.

As previously noted, the NuScale design and operations are not consistent with any STS. The NuScale specifications are written based on the NuScale design and operations.

NuScale is not implementing TSTF-541, and was not a party to its development. NuScale is not proposing to use TSTF-541 to support the content or adequacy of the NuScale TS. This and all TSTF travelers are unrelated to the NuScale design, operations, or proposed specifications.

Conceptually, and as previously discussed with the staff during multiple public meetings, the staff should not apply, request information, or evaluate the adequacy or safety of the NuScale design based on documents that are not related to the NuScale design or operations.

Furthermore, the TS are not submitted for consideration based on deviation or change from existing STS because no STS exist that are consistent with the NuScale design. The content of the TS is based on the requirements of 10 CFR 50.36 as described in FSAR Chapter 16.

The allowances in the listed SRs permit surveillance requirements to be deferred when the safety functions have been satisfied, but continued operations or other preferred operational methods using normal means requires alignment of the components to their non-safety actuated position, contrary to LCO 3.0.5. This is a reasonable allowance with the use of administrative controls as widely used across the commercial nuclear industry. Explanation of circumstances or examples of specific postulated circumstances that are addressed by the allowances in the identified surveillance requirements are provided below.

The specific staff concern described in this section is unclear and appears to contradict previous requests for additional information. In each case, the RAI has failed to request information related to the safety of the proposed allowance; rather they have focused on comparison to unrelated STS or TSTF travelers. For example, the phrase “is not supported by the intended scope of the traveler” is inappropriate and unrelated to the proposed specifications.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9642
Date of RAI Issue: 03/20/2019

NRC Question No.: 16-65

Paragraph (a)(11) of 10 CFR 52.47 and paragraph (a)(30) of 10 CFR 52.79 state that a design certification (DC) applicant and a combined license (COL) applicant, respectively, are to propose technical specifications (TS) prepared in accordance with 10 CFR 50.36 and 50.36a. 10 CFR 50.36 sets forth requirements for TS to be included as part of the operating license for a nuclear power facility.

The following observations are a followup to the response to RAI 506-9614, Question 16-50.

In Revision 2 of DCA Part 4, the applicant revised Section 1.1 by omitting the RTS and ESF response time definitions and defined terms; the applicant also revised the Section 3.3 response time Surveillances and associated Bases, which are quoted below. In these quotations, underlined and lined-through text indicate staff recommended additional editorial corrections to the Surveillance statements and associated Bases. Following the quoted material for each SR, the staff has provided its observations about shaded text. The applicant is requested to address each of the observations below.:

SR 3.3.1.3 Verify channel ~~required~~ response time is within limits. | 24 months

The Bases for SR 3.3.1.3 state:

> This SR ~~3.3.1.3~~ verifies that the individual channel ~~actuation~~ response times are less than or equal to the maximum values assumed in the accident analysis. The channel ~~actuation~~ response time is the time from when the process variable exceeds its setpoint until the output from the channel analog logic reaches the input of the MPS

digital logic. Response time testing criteria are included in FSAR Chapter 7.

Channel response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the channel response time is verified. Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications. ...

Observations on SR 3.3.1.3 and associated Bases:

1. The word “required” is unnecessary in the Surveillance statement.

2. In the Bases phrase, “channel actuation response time,” the word “actuation” is unnecessary and inconsistent with SR 3.3.1.3, which uses the phrase “channel response time.”

3. The phrase “accident analysis” is used in the Bases for SR 3.3.1.3, but the phrase “safety analysis” is used in the corresponding similar sentences in the Bases for SR 3.3.2.2 and SR 3.3.3.2. This appears to be inconsistent.

4. In the Bases, consider modifying the reference “FSAR Chapter 7” to say “FSAR Section 7.2 (Ref. 1).”

5. The “channel response time” verified by SR 3.3.1.3 appears to span the channel’s process sensor to the channel’s output from the analog to digital converter, and excludes the comparison of the digital signal with the channel trip setpoint in the SFM. SER Section 7.2 gives the staff’s evaluation of the “digital response time” verification testing.

6. When “channel response time” is meant, the Bases should use the full phrase for clarity, not just “response time,” which is more general. Consider discussing the overlappingcomponent response times in an MPS instrument channel (e.g., “sensor response time” is already called out).

7. Regarding allocated MPS instrument channel component response times, the last

sentence of the definitions of the W-AP1000-STS defined terms RTS Response Time and ESF Response Time states:

> In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been peviously reviewed and approved by the NRC.

Unless the staff has previously reviewed and approved the components and methodology for response time verification [by allocation] as a part of the NuScale DCA review, as documented in SER Chapter 7, the above quoted SR 3.3.1.3 Bases statement, “Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications.” may need to be designated as a COL action item.

SR 3.3.2.2 Verify ~~required~~ response time is within limits. | 24 months

The Bases for SR 3.3.2.2 state:

> This SR ensures that the response time<u>s</u> of the <u>two</u> RTS divisions are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the process variable exceeds the trip setpoint value at the sensor to the time at which the [reactor trip breakers (RTBs)] open. Total response time may be verified by any series of sequential, overlapping, or total channel measurements.
>
> ... The maximum digital time response is described in the FSAR. This SR encompasses the response time of the RTS division from the output of the equipment interface modules until the RTBs are open. ...

Observations on SR 3.3.2.2 and associated Bases:

1. The word “required” is unnecessary in the Surveillance statement.

2. The phrase “accident analysis” is used in the Bases for SR 3.3.1.3, but the phrase “safety analysis” is used in the corresponding similar sentences in the Bases for SR 3.3.2.2 and SR 3.3.3.2. This appears to be inconsistent.

3. In the Bases, consider modifying the reference to “FSAR” to say “FSAR Section 7.2 (Ref. 1).”

4. Consider whether it would be more accurate to say “total division measurements” in place of “total channel measurements.”

5. The “RTS division response time” verified by SR 3.3.2.2, appears to span the analog output of the RTS EIM to the division’s two RTBs, and excludes verification of the “digital time response,” which appears to span the components from receipt of the digital process signal, to the setpoint comparison in the SFM, through the SVM, and through the priority logic of the RTS EIM. SER Section 7.2 gives the staff’s evaluation of the “digital response time” verification testing.

6. Consider discussing in the Bases the overlapping digital component response times in an RTS division and how “maximum digital time response” is verified.

SR 3.3.3.2 Verify ~~required~~ pressurizer heater breaker response time is within limits. | 24 months

The Bases for SR 3.3.3.2 state:

> This SR ensures that the pressurizer heater breaker opening response times are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the process variable exceeds the trip setpoint value at the sensor to the time at which the ESF

component actuates. Total response time may be verified by any series of sequential, overlapping, or total channel measurements.

Response times of the sensors are tested in accordance with LCO 3.3.1 , "MPS Instrumentation." The maximum digital time response is described in the FSAR. This SR encompasses the response time of the ESFAS from the output of the equipment interface modules to the loss of voltage at the output of the pressurizer heater breaker. The response time of valves actuated by the ESFAS are verified in accordance with the IST program, and LCO 3.4.6, "Chemical and Volume Control System Isolation Valves," LCO 3.4.10, "LTOP Valves," LCO 3.5.1, "ECCS," LCO 3.5.2, "DHRS," LCO 3.6.2, "Containment Isolation Valves," LCO 3.7.1, "MSIVs," and LCO 3.7.2, "Feedwater Isolation."

Observations on SR 3.3.3.2 and associated Bases:

1. The word "required" is unnecessary in the Surveillance statement.

2. The phrase "accident analysis" is used in the Bases for SR 3.3.1.3, but the phrase "safety analysis" is used in the corresponding similar sentences in the Bases for SR 3.3.2.2 and SR 3.3.3.2. This appears to be inconsistent.

3. In the Bases, consider modifying the reference to "FSAR" to say "FSAR Section 7.2 (Ref. 1)."

4. Consider whether it would be more accurate to say "total division measurements" in place of "total channel measurements."

5. The "ESFAS division response time" verified by SR 3.3.3.2, appears to span the analog output of the pressurizer heater breaker EIM to the division's two pressurizer heater breakers, and excludes verification of the "digital time response," which appears to span the components from receipt of the digital process signal, to the setpoint comparison in the SFM, through the SVM, and through the priority logic of the pressurizer heater breaker EIM. Also excluded is the digital portion of the ESFAS division for the other ESF Logic and Actuation functions. See SER Section 7.2 for the staff's evaluation of the "digital response time" verification testing.

6. Consider discussing in the Bases the overlapping digital component response times in an ESFAS division and how “maximum digital time response” is verified.

7. Consider clarifying in the Bases for the following SRs for Inservice Testing Program ESFAS valve actuations (The Frequency of “In accordance with the Inservice Testing Program” is taken to mean 24 months for these SRs.) that the valve "isolation (or 'closure') time" or "open actuation time" (the time to stroke closed or stroke open, respectively) is included in the ESF Function's overall response time. Also, the word "required" is not needed because it is redundant to "within limits.":

SR 3.4.6.2	Verify the ~~required~~ isolation time of each automatic power operated CVCS valve is within limits.
SR 3.4.10.2	Verify the open actuation time of each RVV is within limits.
SR 3.5.1.2	Verify the open actuation time of each RVV and RRV is within limits.
SR 3.5.2.4	Verify the open actuation time of each DHRS actuation valve is within limits.
SR 3.6.2.3	Verify the isolation time of each automatic containment isolation valve is within limits except for valves that are open under administrative controls.

Note that the staff is tracking the exception to SR 3.6.2.3 as an open item under RAI 197-9051 (ML17237C008), Question 16-28, which is described in SER Section 16.4.8.5, “Proposed exceptions to meeting certain surveillances for isolation valves and circuit breakers.”

SR 3.7.1.2	Verify isolation time of each MSIV and MSIV bypass valve is within limits on an actual or simulated actuation signal.
SR 3.7.2.2	Verify the closure time of each FWIV and FWRV is within limits on an actual or simulated actuation signal.

BACKGROUND DISCUSSION

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

In Revision 1 of DCA Part 4, GTS Section 1.1, “Definitions,” included the W-STS definition of ESF RESPONSE TIME with changes related to the NuScale design’s lack of ESF pumps and Class 1E diesel generators, as indicated in the following mark up of the W-STS definition:

> The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions~~, pump discharge pressures reach their required values, etc.~~). ~~Times shall include diesel generator starting and sequence loading delays, where applicable.~~ The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

Because these changes resulted in an ESF RESPONSE TIME definition appropriate for the NuScale design, the staff considered the changes acceptable. However, in Revision 1 of DCA Part 4, Section 3.3, “Instrumentation,” did not use the ESF RESPONSE TIME defined term, but did use the RTS RESPONSE TIME defined term, even though Section 1.1 did not include its definition. Section 3.3 stated the response time Surveillances as follows:

SR 3.3.1.3 Verify channel RESPONSE TIME is within limits. | 24 months

SR 3.3.2.2 Verify RTS RESPONSE TIME is within limits. | 24 months

SR 3.3.3.2 Verify required RESPONSE TIME is within limits. | 24 months

In RAI 506-9614 (ML18289A751), Question 16-50, the staff requested that the applicant provide justification for not including response time defined terms and their definitions in GTS Section 1.1, and in response time SRs in Section 3.3. In its response (ML18347A619) to RAI 506-9614, Question 16-50, the applicant explained in detail the reasons the STS response time definitions are not suitable for the NuScale instrumentation design, and how the response time for the digital signal processing is “verified during factory acceptance testing of the MPS as described in associated inspections, tests, analyses, and acceptance criteria listed in [Revision 2 of DCA Part 2,] Tier 1, Table 2.5-7 of the FSAR.” The response also stated:

> The self-testing features of the design will notify operators of failures that could impact system function, however degradation of the system response time cannot occur. An OPERABLE MPS has a defined digital response time that does not change and does not require further verification.

Pending completion of its review of the applicant’s response, the staff is tracking the omission of the response time definitions and the adequacy of the proposed response time verification Surveillances as an open item under RAI 506-9614, Question 16-50. The staff is tracking the completion of the disposition of the above observations under this RAI question.

NuScale Response:

The word "required" was removed from the locations identified. The word "required" was also removed from SR 3.3.2.3 and SR 3.3.3.3.

The word "actuation" was removed from the Bases of SR 3.3.1.3.

The phrase "accident analysis" in the Bases of SR 3.3.1.3 was changed to "safety analyses" for consistency with the Bases of SR 3.3.2.2 and SR 3.3.3.2. Other locations in the Bases for LCO 3.3.1 were also modified.

The references to FSAR Chapter 7 were not changed to refer specifically to FSAR Section 7.2. System response time is also discussed in FSAR Section 7.1.

The channel response time verified by SR 3.3.1.3 spans from the process sensor to the output of the analog to digital converter. If a channel's design performs a digital comparison to evaluate a trip setpoint then that evaluation is within the scope of the digital response time described in FSAR Section 7.1.4 and Section 7.0 of TR-1015-18653-P-A. If a channel includes analog processing or comparison before the analog to digital converter, then the analog processing and comparison would be within the scope of the channel response time. Practically, there may be no difference if procedures use a channel response time measured from the sensor conservatively through a digital comparison device - however the channel response time is as described from the sensor to the output of the channel's analog to digital converter.

The phrase "response time" is used in different contexts. Editorial clarifications were made to the Bases of SR 3.3.1.3 to differentiate between channel response times, actuation response times, and response times in general.

The allowance to allocate sensor response times has been placed in square brackets and a new COL action item, COL Item 16.1-3, created to describe the inclusion of the phrase and its use requires prior NRC approval. Corresponding changes to FSAR Table 1.8-2 were made.

The Bases of SR 3.3.2.2 and SR 3.3.3.2 were modified to refer to "total division measurements."

The Bases for the testing of inservice test program ESFAS-actuated valves were previously modified to include a description of the test so that "[i]solation time is measured from output of the module protection system interface module until the valves are isolated." This statement, in combination with the Bases of SR 3.3.3.2 describes how the ESF function's overall response time is measured.

The Bases Control Program required by specification 5.5.7, including the requirement for evaluation of any change against the criteria in 10 CFR 50.59 is adequate to assure that response time testing will measure the full response times described in the Bases of SR 3.3.3.2. Any proposed change that reduced the scope of testing would not support the assumptions of the safety analyses and therefore require NRC approval before implementation.

Impact on DCA:

The Technical Specifications have been been revised as described in the response above and as shown in the markup provided in this response.

RAI 01-61, RAI 02.04.13-1, RAI 03.04.01-4, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.03-1, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI 03.05.03-4, RAI 03.06.02-6, RAI 03.06.02-15, RAI 03.06.03-11, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-4S3, RAI 03.07.02-6S1, RAI 03.07.02-6S2, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.07.02-15S5, RAI 03.07.02-16S1, RAI 03.07.02-23S1, RAI 03.07.02-26, RAI 03.08.04-1S1, RAI 03.08.04-3S2, RAI 03.08.04-23S1, RAI 03.08.04-23S2, RAI 03.08.04-23S3, RAI 03.08.05-14S1, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.02-67, RAI 03.09.02-69, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-16S1, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 03.11-14S1, RAI 03.11-18, RAI 03.13-3, RAI 04.02-1S2, RAI 05.02.03-19, RAI 05.02.05-8, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 05.04.02.01-19, RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-19, RAI 06.02.06-22, RAI 06.02.06-23, RAI 06.04-1, RAI 09.01.01-20, RAI 09.01.01-20S1, RAI 09.01.02-4, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-3, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8, RAI 10.02-1, RAI 10.02-2, RAI 10.02-3, RAI 10.02.03-1, RAI 10.02.03-2, RAI 10.03.06-1, RAI 10.03.06-5, RAI 10.04.06-1, RAI 10.04.06-2, RAI 10.04.06-3, RAI 10.04.10-2, RAI 11.01-2, RAI 11.01-2S2, RAI 12.03-55S1, RAI 12.03-63, RAI 13.01.01-1, RAI 13.01.01-1S1, RAI 13.02.02-1, RAI 13.03-4, RAI 13.05.02.01-2, RAI 13.05.02.01-2S1, RAI 13.05.02.01-3, RAI 13.05.02.01-3S1, RAI 13.05.02.01-4, RAI 13.05.02.01-4S1, RAI 14.02-7, RAI 16-65, RAI 19-31, RAI 19-31S1, RAI 19-38, RAI 20.01-13

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL applicant that references the NuScale Power Plant design certification will list additional site-specific piping and instrumentation diagrams and legends as applicable.	1.7
COL Item 1.8-1:	A COL applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the design parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	A COL applicant that references the NuScale Power Plant design certification will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, except Section 2.4.8 and Section 2.4.10.	2.4

Table 1.8-2: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 14.2-2:	~~-~~A COL applicant that references the NuScale Power Plant design certification is responsible for the development of the Startup Administration Manual that will contain the administrative procedures and requirements that control the activities associated with the Initial Test Program. The COL applicant will provide a milestone for completing the Startup Administrative Manual and making it available for NRC inspection.	14.2
COL Item 14.2-3:	A COL applicant that references the NuScale Power Plant design certification will identify the specific operator training to be conducted during low-power testing related to the resolution of TMI Action Plan Item I.G.1, as described in NUREG-0660, NUREG-0694, and NUREG-0737.	14.2
COL Item 14.2-4:	A COL applicant that references the NuScale Power Plant design certification will provide a schedule for the Initial Test Program.	14.2
COL Item 14.2-5:	A COL applicant that references the NuScale Power Plant design certification will provide a test abstract for the potable water system pre-operational testing.	14.2
COL Item 14.2-6:	A COL applicant that references the NuScale Power Plant design certification will provide a test abstract for the seismic monitoring system pre-operational testing.	14.2
COL Item 14.2-7:	A COL applicant that references the NuScale Power Plant design certification will select the plant configuration to perform the Island Mode Test (number of NuScale Power Modules in service).	14.2
COL Item 14.3-1:	A COL applicant that references the NuScale Power Plant design certification will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for emergency planning.	14.3
COL Item 14.3-2:	A COL applicant that references the NuScale Power Plant design certification will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for structures, systems, and components within their scope.	14.3
COL Item 16.1-1:	A COL applicant that references the NuScale Power Plant design certification will provide the final plant-specific information identified by [ ] in the generic Technical Specifications and generic Technical Specification Bases.	16.1
COL Item 16.1-2	A COL applicant that references the NuScale Power Plant design certification will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.	16.1
COL Item 16.1-3	A COL applicant that references the NuScale Power Plant design certification, and uses allocations for sensor response times based on records of tests, vendor test data, or vendor engineering specifications as described in the Bases for Surveillance Requirement 3.3.1.3, will do so for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.	16.1
COL Item 17.4-1:	A COL applicant that references the NuScale Power Plant design certification will describe the ~~r~~Reliability ~~a~~Assurance ~~p~~Program conducted during the operations phases of the plant's life.	17.4
COL Item 17.4-2:	A COL applicant that references the NuScale Power Plant design certification will identify site-specific structures, systems, and components within the scope of the Reliability Assurance Program.	17.4
COL Item 17.4-3:	A COL applicant that references the NuScale Power Plant design certification will identify the quality assurance controls for the Reliability Assurance Program structures, systems, and components during site-specific design, procurement, fabrication, construction, and preoperational testing activities.	17.4
COL Item 17.5-1:	A COL applicant that references the NuScale Power Plant design certification will describe the ~~q~~Quality ~~a~~Assurance ~~p~~Program applicable to site-specific design activities and to the construction and operations phases.	17.5
COL Item 17.6-1:	A COL applicant that references the NuScale Power Plant design certification will describe the program for monitoring the effectiveness of maintenance required by 10 CFR 50.65.	17.6
COL Item 18.5-1:	A COL applicant that references the NuScale Power Plant design certification will address the staffing and qualifications of non-licensed operators.	18.5

Table 16.1-1 provides the initial surveillance test frequencies to be incorporated into the Surveillance Frequency Control Program (SFCP) required by NuScale GTS 5.5.11. The table identifies each GTS surveillance test requirement that references the SFCP, the base testing frequency for evaluation of future changes to the surveillance test frequency, and the basis for that test frequency.

Incorporation of Technical Specification Task Force Change Travelers

Technical Specification Task Force (TSTF) travelers issued since publication of Revision 4 of the ISTS were reviewed in the development of the NuScale GTS. Travelers were incorporated into the NuScale GTS or utilized as a basis for similar NuScale situations as described in the conformance report (Reference 16.1-1). The TSTF travelers considered in development of the NuScale GTS are listed in that report.

The GTS are intended to be used as a guide in the development of the plant-specific technical specifications. Preliminary information has been provided in single brackets [ ]. Combined license applicants referencing the NuScale Power Plant are required to provide the final plant-specific information.

COL Item 16.1-1: A COL applicant that references the NuScale Power Plant design certification will provide the final plant-specific information identified by [ ] in the generic Technical Specifications and generic Technical Specification Bases.

RAI 03.06.03-11

COL Item 16.1-2: A COL applicant that references the NuScale Power Plant design certification will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.

RAI 16-65

COL Item 16.1-3: A COL applicant that references the NuScale Power Plant design certification, and uses allocations for sensor response times based on records of tests, vendor test data, or vendor engineering specifications as described in the Bases for Surveillance Requirement 3.3.1.3, will do so for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

16.1.2 References

16.1-1 NuScale Power, LLC, "Technical Specifications Regulatory Conformance and Development Technical Report," TR-1116-52011, Rev. 0.

16.1-2 Nuclear Energy Institute, "Risk-Informed Technical Specifications Initiative 5b-Risk-Informed Method for Control of Surveillance Frequencies," NEI 04-10, Revision 1, Washington, DC, April 2007.

16.1-3 Nuclear Energy Institute, Risk-Informed Technical Specifications Initiative 4b-Risk-Managed Technical Specifications (RMTS) Guidelines," NEI 06-09, Rev. 0-A, Washington, DC, November 2006.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------------------NOTES--------------------------- 1. Adjust Neutron Monitoring System (NMS) nuclear instrument channel when absolute difference is > 1% RTP. 2. Not required to be performed until 12 hours after reaching 15% RTP. 3. If the calorimetric heat balance is < 50% RTP, and if NMS nuclear instrumentation channel indicated power is: a. lower than the calorimetric measurement by > 1%, then adjust the NMS nuclear instrumentation channel upward to match the calorimetric measurement. b. higher than the calorimetric measurement, then no adjustment is required. --- Compare results of calorimetric heat balance to NMS nuclear instrument channel output.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.3	-------------------------------NOTE----------------------------- Neutron detectors are excluded from response time testing. --- Verify channel ~~required~~ response time is within limits.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	-----------------------------NOTE----------------------------- Not required to be met for reactor trip breakers that are open. --- Perform ACTUATION LOGIC TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.2	-----------------------------NOTE----------------------------- Not required to be met for reactor trip breakers that are open. --- Verify ~~required~~ response time is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.3	-----------------------------NOTE----------------------------- Not required to be met for Class 1E isolation devices that have isolated 1E circuits from non-1E power. --- Perform CHANNEL CALIBRATION on each Class 1E isolation device.~~Verify associated Class 1E isolation devices are OPERABLE.~~	In accordance with the Surveillance Frequency Control Program
SR 3.3.2.4	-----------------------------NOTE----------------------------- Not required to be met for reactor trip breakers that are open. --- Verify each RTB actuates to the open position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.3	------------------------------NOTE------------------------------ Not required to be met for Class 1E isolation devices that have isolated 1E circuits from non-1E power. --- Perform CHANNEL CALIBRATION on each Class 1E isolation device~~Verify associated Class 1E isolation devices are OPERABLE~~.	In accordance with the Surveillance Frequency Control Program
SR 3.3.3.4	------------------------------NOTE------------------------------ Not required to be met for pressurizer heater ~~trip~~ breakers that are open or breakers closed under administrative controls. --- Verify each pressurizer heater ~~trip~~ breaker ~~(PHTB)~~ actuates to the open position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required valves accumulator pressures are within limits.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.6.2	---------------------------------NOTE------------------------------ Not required to be met for valves that are closed or open under administrative controls. --- Verify the ~~required~~ isolation time of each automatic power operated CVCS valve is within limits.	 In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.6.3	Verify each automatic CVCS valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal except for valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program

BASES

BACKGROUND (continued)

- digital logic circuits; and
- communication engines.

The signal conditioning input sub-modules of the SFM are comprised of an analog circuit and a digital circuit. The analog circuit converts analog voltages or currents into a digital representation. The digital representation of the process sensor output is communicated from the signal conditioning input sub-module to the digital logic circuits that form the trip or actuation determination block.

An SFM trip or actuation determination block accepts input from up to four signal conditioning input sub-modules. The output of each of the signal conditioning input sub-modules is sent to three redundant core logic signal paths in the programmable portion of the SFM that form the trip determination block.

The core logic functions in each of the three redundant signal paths independently:

- performs the safety function algorithm;
- compares the safety function algorithm output to a setpoint and makes a reactor trip and ESF actuation determination; and
- generates permissives and control interlocks.

The information provided via the signal conditioning input sub-modules to the core logic is also provided to the module control system (MCS), the safety display and indication (SDI) system, and the maintenance workstation (MWS) via the monitoring and indication bus communication module (MIB-CM).

The trip and actuation setpoints used in the SFM core logic function are based on the analytical limits derived from safety~~accident~~ analysis (Ref. 9~~5~~). The calculation of the LTSP specified in the Setpoint Program (SP) is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those MPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the LTSP specified in the SP is conservative with respect to the analytical limits. The nominal trip setpoint (NTSP) is the LTSP with margin added and is always equal to or more conservative than the LTSP. A detailed description of the methodology

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Each of the analyzed accidents and transients which require a reactor trip or engineered safety feature can be detected by one or more MPS Functions. The MPS Functions that are credited to mitigate specific design basis events are described in the FSAR Chapter 15 (Ref. 9). Setpoints are specified in the [owner-controlled requirements manual].

Each MPS setpoint is chosen to be consistent with the function of the respective trip. The basis for each setpoint falls into one of three general categories:

- To ensure that the SLs are not exceeded during AOOs;
- To actuate the RTS and ESFAS during accidents; and
- To prevent material damage to major components (equipment protection).

The MPS maintains the SLs during AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed.

The Module Protection System instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Permissive and interlock setpoints automatically provide, or allow manual or automatic blocking of trips during unit evolutions. They are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the initial conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple conservative initial conditions for the safety~~accident~~ analysis, they are generally considered as nominal values without regard to measurement accuracy.

Operating bypasses are addressed in the footnotes to Table 3.3.1-1. They are not otherwise addressed as specific Table entries.

The automatic bypass removal features must function as a backup to manual actions for all safety related trips to ensure the trip Functions are not operationally bypassed when the safety analysis assumes the Functions are OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2

A periodic calibration (heat balance) is performed when THERMAL POWER is above 15%. The Linear Power Level signal and the nuclear instrumentation system addressable constant multipliers are adjusted to make the nuclear power calculations agree with the calorimetric calculation if the absolute difference is ≥ 1%. The value of 1% is adequate because this value is assumed in the safety analysis. These checks (and, if necessary, the adjustment of the nuclear power signal) are adequate to ensure that the accuracy is maintained within the analyzed error margins. The power level must be above 15% RTP to obtain accurate data. At lower power levels, the accuracy of calorimetric data is questionable.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

The Surveillance is modified by three Notes. The first Note indicates that the neutron monitoring system nuclear instrument channel must must be calibrated when the absolute difference is > 1% when compared to the calorimetric heat balance. The second Note indicates that this Surveillance need only be performed within 12 hours after reaching 15% RTP. The 12 hours after reaching 15% RTP is required for unit stabilization, data taking, and flow verification. The secondary calorimetric is inaccurate at lower power levels. A third Note is provided that permits operation below 15% RTP without adjusting the instrument channel as long as the indicated nuclear instrument power is conservatively higher than the calorimetric heat balance results. This third Note is an exception to the first Note and only applies when below 15% RTP.

SR 3.3.1.3

This SR 3.3.1.3 verifies that the individual channel ~~actuation~~ response times are less than or equal to the maximum values assumed in the safety~~accident~~ analysis. The channel ~~actuation~~ response time is the time from when the process variable exceeds its setpoint until the output from the channel analog logic reaches the input of the MPS digital logic. Response time testing criteria are included in FSAR Chapter 7.

Response time may be verified by any series of sequential, overlapping or total channel measurements, including allocated sensor response time, such that the response time is verified. [Allocations for sensor response times may be obtained from records of test results, vendor test data, or vendor engineering specifications.] The actuation response time testing of the RTS and ESFAS divisions are tested in accordance with LCO 3.3.2 and 3.3.3.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3 is modified by a Note indicating that neutron detectors are excluded from channel response time testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.4

This SR is modified by a Note that indicates that neutron detectors are excluded from CHANNEL CALIBRATION.

The Surveillance verifies that the channel responds to a measured process variable within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift between successive calibrations to ensure that the channel remains operational between successive tests. The test is performed in accordance with the SP. If all as-found measured values during calibration and surveillance testing are inside the as-left tolerance band, then the channel is fully operable, no additional actions are required.

If all as-found measured values during calibration testing and surveillance testing are within the as-found tolerance band but outside the as-left tolerance band, then the instrumentation channel is fully operable, however, calibration is required to restore the channel within the as-left tolerance band.

If any as-found measured value is outside the as-found tolerance band, then the channel is inoperable, and corrective action is required. The unit must enter the Condition for the particular MPS Functions affected. The channel as-found condition will be entered into the Corrective Action Program for further evaluation and to determine the required maintenance to return the channel to OPERABLE.

Interlocks and permissives are required to support the Function's OPERABILITY and are addressed by this CHANNEL CALIBRATION. This is accomplished by ensuring the channels are calibrated properly in accordance with the SP. If the interlock or permissive is not functioning as designed, the condition is entered into the Corrective Action Program and appropriate OPERABILITY evaluations are performed for the affected Function(s). The affected Function's OPERABILITY can be met if the interlock is manually enforced to properly enable the affected Function.

BASES

REFERENCES

1. Regulatory Guide 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation.”

2. 10 CFR 50, Appendix A, GDC 21.

3. 10 CFR 50.34.

4. FSAR, Chapter 7, "Instrumentation and Controls."

5. FSAR, Chapter 14, "Initial Test Program and ITAAC."

6. 10 CFR 50.49.

7. TR-0606-49121, Rev. 0, “NuScale Instrument Setpoint Methodology.”

8. IEEE Standard 603-1991.

9. FSAR, Chapter 15, "Transient and Accident Analyses."

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

This SR ensures that the response time of the RTS divisions are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the process variable exceeds the trip setpoint value at the sensor to the time at which the RTBs open. Total response time may be verified by any series of sequential, overlapping, or total division~~channel~~ measurements.

Response times of the sensors are tested in accordance with LCO 3.3.1. The maximum digital time response is described in the FSAR. This SR encompasses the response time of the RTS division from the output of the equipment interface modules until the RTBs are open.

A note provides an allowance for the SR so that it does not need to be met for reactor trip breakers that are open. This allowance permits continued operation when a trip breaker may not be able to satisfy the requirements of the SR but is already open. When a reactor trip breaker is open it has performed its safety function.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2

This SR ensures that the pressurizer heater breaker opening response times are verified to be less than or equal to the maximum values assumed in the safety analysis. Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the process variable exceeds the trip setpoint value at the sensor to the time at which ESF component actuates. Total response time may be verified by any series of sequential, overlapping, or total division~~channel~~ measurements.

Response times of the sensors are tested in accordance with LCO 3.3.1. The maximum digital time response is described in the FSAR. This SR encompasses the response time of the ESFAS from the output of the equipment interface modules to the loss of voltage at the output of the pressurizer heater breaker. The response time of valves actuated by the ESFAS are verified in accordance with the IST program, and LCO 3.4.6, "Chemical and Volume Control System Isolation Valves," LCO 3.4.10, "LTOP Valves," LCO 3.5.1, "ECCS," LCO 3.5.2, "DHRS," LCO 3.6.2, "Containment Isolation Valves," LCO 3.7.1, "MSIVs," and LCO 3.7.2, "Feedwater Isolation."

A note provides an allowance for the SR so that it does not need to be met for pressurizer heater breakers that are open in their actuated position. This allowance permits continued operation when a pressurizer heater ~~trip~~ breaker is open because it has performed its safety function. The note also allows intermittent closure of the breakers under ~~manual~~ administrative control when the SR is not met because the slowly occurring nature of the phenomena the automatic heater ~~trip~~ breakers mitigate.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.