RAIO-1117-57093

November 13, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 228 (eRAI No. 9034) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 228 (eRAI No. 9034)," dated September 14, 2017

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. 9034:

- 16-30
- 16-31
- 16-32
- 16-33
- 16-34
- 16-35
- 16-36

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 Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9034

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9034

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

eRAI No.: 9034
Date of RAI Issue: 09/14/2017

NRC Question No.: 16-30

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 model standard technical specifications (STS) in the following documents provide NRC guidance on format and content of TS as acceptable means to meet 10 CFR 50.36 requirements. These documents may be accessed using the Agencywide Documents Access and Management Systems (ADAMS) by their accession numbers.

- NUREG-1431, "STS Westinghouse Plants," Revision 4 (ADAMS Accession Nos. ML12100A222 and ML12100A228)
- NUREG-1432, "STS Combustion Engineering Plants," Revision 4 (ADAMS Accession Nos. ML12102A165 and ML12102A169)
- NUREG-2194, "STS Westinghouse Advanced Passive 1000 (AP1000) Plants," Revision 0 (ADAMS Accession No. ML16111A132)

The NRC staff needs to evaluate technical differences in the proposed generic TS (GTS) from applicable provisions in these documents, which are referenced by the DC applicant in Design Control Document (DCD) Tier 2, Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the GTS to ensure adequate protection of public health and safety, and the completeness and accuracy of the GTS Bases.

Acronyms used in this comment are as follows:

LCO	Limiting Condition for Operation
SDM	SHUTDOWN MARGIN
COLR	CORE OPERATING LIMITS REPORT
CRAs	control rod assemblies
CVCS	Chemical and Volume Control System
CFDS	Containment Flood and Drain System

SFCP Surveillance Frequency Control Program
SR Surveillance Requirement

The following background information is provided to support the staff's questions, which are stated afterwards.

The LCO and Applicability statements of proposed Subsection 3.1.1 are:

LCO 3.1.1	SDM shall be within the limits specified in the COLR.
APPLICABILITY:	MODE 1 with k_{eff} < 1.0, MODES 2, 3, and 4.

The definition of MODE 4 (Transition) in Table 1.1-1 requires (with staff recommended edits in blue) that keff be < 0.95, and by

Footnote (b), that

- All control rod assemblies (CRAs) are incapable of withdrawal,
- The CVCS and CFDS connections to the MODULE are isolated, and
- One or more reactor vent valves are de-energized; and by

Footnote (c), that

- All reactor vessel flange bolts are fully tensioned.

SR 3.1.1.1 states (with staff recommended edits in blue):

SR 3.1.1.1	------------------------------NOTE------------------------------- Not required to be performed in MODE 4. --- Verify SDM to be within the limits specified in the COLR.	[24 hours OR In accordance with the Surveillance Frequency Control Program]

The SR section of the Bases for Subsection 3.1.1, concerning SR 3.1.1.1, states,

> In MODE 1 with k_{eff} < 1.0, and in MODES 2, 3, and 4, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:
>
> a. RCS boron concentration;

b. CRA position;
c. RCS average temperature;
d. Fuel burnup based on gross thermal energy generation;
e. Xenon concentration;
f. Samarium concentration; and
g. Isothermal Temperature Coefficient (ITC).

The SR section of the Bases for Subsection 3.1.1 also states,

> SR 3.1.1.1 is modified by a Note that allows entry into MODE 4 prior to performing the SR.

GTS Section 1.4 provides no example that matches the Note for SR 3.1.1.1. The most similar example is Example 1.4-3, which has a Note that modifies the 7 day Frequency of peformance by stating:

> -----------------------------NOTE------------------------------
> Not required to be performed until 12 hours after
> ≥ 25% RTP.
> ---

This example states, in part

> ... Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance....

End of Background Information

a. The staff is unable to determine whether SR 3.1.1.1 can be performed in MODE 4 (following entry from MODE 3) if the performance requires measurements of the boron concentration and temperature of the reactor coolant in the reactor vessel, since connections to the plant sampling system (PSS) presumably would be isolated.

   1. If such measurements in MODE 4 <u>are not necessary</u> to perform the SDM calculation, then the calculation would need to rely on such data obtained in MODE 3, and also on an assurance that, after entry into MODE 4 until entry into MODE 5, core reactivity changes (due to changes in reactor coolant temperature, and the Xenon and Samarium distributions in the core) would not violate the MODE 4 criterion that k_{eff} be maintained < 0.95. The applicant is requested to explain how the MODE 3 boron concentration is adjusted to provide such assurance.

   2. If such measurements in MODE 4 <u>are necessary</u> to perform the SDM calculation, the applicant is requested to describe how such measurements would be obtained.

b. The proposed Note seems to indicate that the SDM calculation is not performed in MODE 4. However, the quoted statement from the Bases seems to indicate that the SDM calculation is performed in MODE 4. The applicant is requested to revise the presentation of SR 3.1.1.1 Note and Frequency, and the content of the Bases to be mutually consistent, and also consistent with the intended restrictions and allowances for performing the SDM calculation in MODES 3 and 4.

c. The proposed Note also applies to the performance of SR 3.1.1.1 while in MODE 5 before entry into MODE 4, in accordance with SR 3.0.4; and after entry into MODE 4.

- In the first case, SR 3.5.3.3 ("Verify Ultimate Heat Sink bulk average boron concentration is within limits.") ensures that the keff < 0.95 criterion of MODE 4 is satisfied because LCO 3.5.3.c states that the Ultimate Heat Sink "bulk average boron concentration shall be maintained within the limit specified in the COLR"; which is presumably more than sufficient to ensure the reactor is > 5 percent shutdown. Also, Specification 4.3.1.b indicates that water in the spent fuel pool (and by inference, the Ultimate Heat Sink) has a "minimum soluble boron concentration of 800 ppm."

- In the second case, it would appear that the above discussion in Sub-questions a.1 and a.2 would apply.

The applicant is requested to revise the presentation of SR 3.1.1.1 Note and Frequency, and the content of the Bases to be mutually consistent, and also consistent with the intended restrictions and allowances for performing the SDM calculation in MODE 5 with a full core in the reactor vessel and after entry into MODE 4.

d. The applicant is requested to explicitly state the base Frequency for SR 3.1.1.1, using the bracketed presentation indicated in the above markup. Note that the stated 24 hours may not be an appropriate Frequency for verifying SDM is within limits for NuScale in MODES 2, 3, and 4. The applicant is requested to provide a bracketed justification for the base Frequency in the Bases for SR 3.1.1.1. For example, the Bases for the 24 hour Frequency of SR 3.1.1.1 of the CE STS, states:

[The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

OR

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

----------------------------------REVIEWER'S NOTE----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

e. The SR 3.1.1.1 Frequency states: "In accordance with the Surveillance Frequency Control Program." The applicant is requested to state the Frequency as "[24 hours OR In accordance with the Surveillance Frequency Control Program]" as indicated above.

   The applicant is referred to a memorandum dated May 20, 2010 (ADAMS Accession No. ML101390330), from Robert B. Elliott, Chief, Technical Specifications Branch, NRR, to branch chiefs in the NRR Division of Operating Reactor Licensing, "Notification of Issue with NRC Approved TSTF-425, Revision 3, 'Relocate Surveillance Frequencies to Licensee Control-RITSTF Initiative 5b.'" In this memorandum, the staff stated that a licensee requesting an amendment to the operating license to incorporate TSTF-425, Revision 3, into the facility technical specifications must include in the license amendment request the following statement "regarding SF [Surveillance Frequency] Bases relocated to the Surveillance Frequency Control Program (SFCP)" [without changing the SF]:

   > *The existing Bases information describing the basis for the Surveillance Frequency will be relocated to the licensee-controlled Surveillance Frequency Control Program.*

   For most GTS SRs, the NuScale DCA includes neither the base SFs nor the base SF Bases.

   1. Including the above reviewer's note in the SR section of the Bases for each affected GTS SR is recommended for informing a COL applicant that relocation of the base SF for each affected SR to the SFCP shall include the associated Bases for the SF. The applicant is requested to include the base SFs and associated Bases in DCD Chapter 16, as bracketed COL action item information, consistent with STS presentation.

2. Alternatively, the applicant is requested to propose adding a bracketed listing to GTS 5.5.11.a of the SFCP Specification that documents, for each SR, the base SF and the base SF Bases. This approach would be equivalent to the above change, but would be administratively easier to implement by a COL applicant. This listing of base SFs and Bases would need an associated reviewer's note that describes how a COL applicant is expected to resolve the COL action item. For example:

*********************************** EXAMPLE SFCP SPECIFICATION *****************************

5.5 Programs and Manuals

5.5.11 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

[------------------------------REVIEWER'S NOTE------------------------------------
A COL applicant planning to control Surveillance Frequencies under a Surveillance Frequency Control Program shall relocate the base Frequency and the base Frequency Bases, as given below, to the Surveillance Frequency Control Program for each associated Surveillance Requirement. Else, the Frequency shall be stated in the Surveillance Requirement, and its basis in the Bases for the Surveillance Requirement.

Surveillance	Frequency	Bases
SR 3.1.1.1	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

SR 3.1.2.1	31 EFPD thereafter	*(Based on AP1000) The required subsequent Frequency of 31 effective full power days (EFPD) following the initial 60 EFPD after entering MODE 1 is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (enthalpy rise hot channel factor and AXIAL OFFSET) for prompt indication of an anomaly.*
SR 3.1.4.1	12 hours	Associated Bases for base SF
SR 3.1.4.2	92 days	Associated Bases for base SF
SR 3.1.5.1	12 hours	Associated Bases for base SF
SR 3.1.6.1	12 hours	Associated Bases for base SF
SR 3.1.8.1	30 minutes	Associated Bases for base SF
SR 3.1.8.2	24 hours	Associated Bases for base SF
SR 3.1.9.1	24 months	Associated Bases for base SF
SR 3.1.9.2	31 days	Associated Bases for base SF
SR 3.2.1.1	31 EFPD	Associated Bases for base SF
SR 3.2.2.1	12 hours	Associated Bases for base SF
SR 3.3.1.1	12 hours	Associated Bases for base SF
SR 3.3.1.2	24 hours	Associated Bases for base SF
SR 3.3.1.3	24 months	Associated Bases for base SF
SR 3.3.1.4	24 months	Associated Bases for base SF
SR 3.3.2.1	24 months	Associated Bases for base SF
SR 3.3.2.2	24 months	Associated Bases for base SF
SR 3.3.3.1	24 months	Associated Bases for base SF
SR 3.3.3.2	24 months	Associated Bases for base SF
SR 3.3.4.1	24 months	Associated Bases for base SF
SR 3.3.5.1	24 months	Associated Bases for base SF
SR 3.3.5.2	24 months	Associated Bases for base SF
SR 3.3.5.3	24 months	Associated Bases for base SF
SR 3.3.5.4	24 months	Associated Bases for base SF
SR 3.4.1.1	12 hours	Associated Bases for base SF
SR 3.4.1.2	12 hours	Associated Bases for base SF
SR 3.4.2.1	12 hours	Associated Bases for base SF
SR 3.4.3.1	30 minutes	Associated Bases for base SF
SR 3.4.5.1	72 hours	Associated Bases for base SF
SR 3.4.5.2	72 hours	Associated Bases for base SF
SR 3.4.6.2	24 months	Associated Bases for base SF
SR 3.4.7.1	12 hours	Associated Bases for base SF
SR 3.4.7.2	12 hours	Associated Bases for base SF
SR 3.4.7.3	12 hours	Associated Bases for base SF
SR 3.4.7.4	92 days	Associated Bases for base SF
SR 3.4.7.5	24 months	Associated Bases for base SF
SR 3.4.7.6	24 months	Associated Bases for base SF

SR 3.4.7.7	24 months	Associated Bases for base SF
SR 3.4.8.1	7 days	Associated Bases for base SF
SR 3.4.8.2	14 days	Associated Bases for base SF
SR 3.5.1.1	24 months	Associated Bases for base SF
SR 3.5.1.3	NuScale specific	Associated Bases for base SF
SR 3.5.2.1	24 hours	Associated Bases for base SF
SR 3.5.3.1	24 hours	Associated Bases for base SF
SR 3.5.3.2	24 hours	Associated Bases for base SF
SR 3.5.3.3	31 days <u>AND</u> Once within 6 hours after each solution volume increase of ≥ 15,000 gal	Associated Bases for base SF
SR 3.6.2.1	31 days	Associated Bases for base SF
SR 3.6.2.3	24 months	Associated Bases for base SF
SR 3.8.1.1	12 hours	Associated Bases for base SF
SR 3.8.1.2	24 months	Associated Bases for base SF

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b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

*************************** END OF EXAMPLE SFCP SPECIFICATION *************************

A response based on this suggested presentation of base frequencies and associated SF Bases must provide the actual base Frequency and its actual rationale, since the example depicts typical values and no NuScale-specific SF Bases.

The applicant is requested to treat this Sub-question as a global issue for all SRs for which the Frequency is stated as “In accordance with the Surveillance Frequency Control Program.”

NuScale Response:

With regard to parts a, b, and c:

As described in Table 1.1-1, MODES, entry into MODE 4 requires disabling controls and isolation of lines with potential to affect reactivity in the module. This includes preventing the measurement of boron concentration and temperature in the reactor and containment vessels which are in communication when the RRVs are open. Entry into MODE 4 will require the operating staff to verify by calculations and supporting analyses that the boron concentration will remain adequate to assure that reactivity stays within limits. Implementation will be by plant procedures prepared by a COL applicant as described in COL Items identified in Section 13.5, Plant Procedures.

The Bases for SR 3.1.1.1 are being modified to indicate that the SR is not required to be performed in MODE 4, and to clarify the requirement to verify that the SDM is and will remain within limits during MODE 4 operations. The Bases also describe the use of reactivity calculations to address MODE 4 conditions that conservatively account for passive phenomena that could adversely affect the SDM including temperature changes and Xenon decay effects on reactivity.

With regard to parts d and e:

NuScale has added Table 16.1-1 to the FSAR listing the base frequencies for use in implementing the SFCP required by Technical Specification 5.5.11, Surveillance Frequency Control Program. Relocation of the table is consistent with the relocation of technical specifications to licensee-controlled documents and assure that 10 CFR 50.59 is applicable to any future changes. In addition to the table in the FSAR, the table in Enclosure 1 reiterates the change to the FSAR and provides a comparison of corresponding industry frequencies where applicable.

Impact on DCA:

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

In MODE 1 with $k_{eff} \geq 1.0$, SDM is verified by observing that the requirements of LCO 3.1.5 and LCO 3.1.6 are met. In the event that a CRA is known to be untrippable, however, SDM verification must account for the worth of the untrippable CRA as well as another CRA of maximum worth.

In MODE 1 with $k_{eff} < 1.0$, and in MODES 2, 3, and 4, the SDM is verified by performing a reactivity balance calculation, considering the listed reactivity effects:

a. RCS boron concentration;

b. CRA position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.

~~SR 3.1.1.1 is modified by a Note that allows entry into MODE 4 prior to performing the SR.~~SR 3.1.1.1 is modified by a Note that indicates the surveillance is not required to be performed in MODE 4. In MODE 4 Table 1.1-1, MODES requires the module to be isolated from control systems and process lines that could change the SDM. Verification that the SDM will be met in MODE 4 is required before entry from MODE 5, and before entry from MODE 3 in accordance with SR 3.0.4.

During module movement instrumentation is not available to measure variables that could affect the SDM. Therefore reactivity calculations performed to verify the SDM conservatively account for passive phenomena that may occur such as temperature changes and Xenon decay affects that could occur during the MODE 4 conditions.

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

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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.

16.1.2 References

16.1-1 Technical Report TTR-1116-52011, "Technical Specifications Regulatory Conformance and Development Technical Report," Rev. 0.

16.1-2 NEI 04-10, Risk-Informed Technical Specifications Initiative 5b - Risk-Informed Method for Control of Surveillance Frequencies - Industry Guidance Document, Rev. 1, April 2007.

16.1-3 NEI 06-09, Risk-Informed Technical Specifications Initiative 4b - Risk-Managed Technical Specifications (RMTS) Guidelines - Industry Guidance Document, Rev. 0-A, November 2006.

RAI 16-30

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies

Surveillance Requirement	Base Frequency	Basis
3.1.1.1	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required shutdown margin (SDM). This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.
3.1.2.1	31 effective full-power days (EFPDs)	The required subsequent Frequency of 31 EFPDs, following the initial 60 EFPDs after exceeding 5% rated thermal power (RTP), is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g. axial offset (AO)) monitored by the core monitoring system for prompt indication of an anomaly.
3.1.4.1	12 hours	Verification that individual control rod assembly (CRA) positions are within alignment limits at a 12 hour Frequency provides a history that allows the operator to detect a CRA that is beginning to deviate from its expected position. The specified Frequency takes into account other CRA position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.
3.1.4.2	92 days	The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the CRAs.
3.1.5.1	12 hours	Since the shutdown CRAs are not moved during routine operation, except as part of planned surveillances, verification of shutdown CRA position at a Frequency of 12 hours is adequate to ensure that the shutdown CRAs are within their insertion limits. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.
3.1.6.1	12 hours	Verification of the regulating group insertion limits at a Frequency of 12 hours is sufficient to detect CRA that may be approaching the insertion limits since, normally, very little rod motion is expected to occur in 12 hours.
3.1.8.1	30 minutes	Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.
3.1.8.2	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.
3.1.9.1	31 days	A 31 day Frequency is considered reasonable in view of other administrative controls that will ensure a misconfiguration of the chemical and volume control system (CVCS) makeup pump demineralized water flow path is unlikely. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of CVCS makeup pump demineralized water flow path configuration.
3.1.9.2	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment. The actuation logic is tested as part of Engineered Safety Features Actuation System (ESFAS) Actuation and Logic testing, and valve performance is monitored as part of the Inservice Testing Program.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.1.9.3	31 days	The 31 day Frequency of this SR was developed considering the known stability of stored borated water and the low probability of any undesired source of diluting pure water. The Frequency takes into account administrative controls that will ensure changes to boron concentration are performed in accordance with written procedures. This frequency also considers the size of the boric acid storage tank and the normally expected demands of boric acid for plant operations.
3.1.9.4	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.
3.2.1.1	31 EFPD	The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this frequency is short enough that the Enthalpy Rise Hot Channel Factor limit cannot be exceeded for any significant period of operation.
3.2.2.1	7 days	The Surveillance Frequency of 7 days is adequate considering that the AO is monitored by a computer and any deviation from requirements is alarmed.
3.3.1.1	12 hours	The Frequency of 12 hours is based on industry operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.
3.3.1.2	12 hours	The Frequency of every 12 hours is adequate. It is based on industry operating experience, considering industry instrument reliability and operating history data for instrument drift. Together, with engineering judgment, these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than +1% RTP is not expected in any 12 hour period.
3.3.1.3	24 months	As appropriate, each channel's response must be verified every 24 months. This test measures the portion of the response time from the sensor to receipt in the digital Module Protection System. The digital processing portions of the Module Protection System are assumed to function in less than 1 second consistent with their design. Equipment actuation is measured through testing required by 3.3.2, 3.3.3, and LCO surveillance requirements associated with the actuated components. Response times cannot be determined during unit operation because sensor inputs are required to be varied to measure response times. Industry experience has shown that these components usually pass this surveillance when performed at the 24 months Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
3.3.1.4	24 months	The Frequency is justified by the assumption of a 30 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.
3.3.1.5	24 months	The 24 month frequency is acceptable based on consideration of the design reliability of the equipment.
3.3.2.1	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. This test frequency is justified based on the system design, which includes the use of continuous diagnostic test features that will report a failure within the logic and actuation system to the operator promptly. The only part of the actuation logic circuitry that is not continuously self-tested is the actuation and priority logic circuit which consists of simple discrete components that are very reliable.
3.3.2.2	24 months	The frequency of 24 months is justified based on industry operating experience with similar equipment.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.3.2.3	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.2.4	24 months	The Frequency of 24 months is based on the known reliability of similar Functions in licensed designs and the multidivisional redundancy available, and has been shown to be acceptable through industry operating experience.
3.3.3.1	24 months	This test frequency of 24 months is justified based on the system design which includes the use of continuous diagnostic test features that will report a failure within the logic and actuation system to the operator promptly. The only part of the actuation logic circuitry that is not continuously self-tested is the actuation and priority logic circuit which consists of simple discrete components that are very reliable.
3.3.3.2	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.3.3	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.3.4	24 months	The Frequency of 24 months is based on the known reliability of similar Functions in licensed designs and the multidivisional redundancy available, and has been shown to be acceptable through industry operating experience.
3.3.4.1	24 months	The Frequency of 24 months is based on the known reliability of similar Functions in licensed designs and the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.5.1	24 months	The 24 month frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.5.2	24 months	The Frequency of 24 months is based on the use of the data display capability in the control room as part of the normal unit operation and the availability of multiple video display units at the Remote Shutdown Station. The Frequency of 24 months is based upon industry operating experience and consistency with control room hardware and software.
3.3.5.3	24 months	The 24 month frequency is based on the known reliability of similar Functions in licensed designs and the redundancy available, and has been shown to be acceptable through industry operating experience.
3.4.1.1	12 hours	Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by industry operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.4.1.2	12 hours	Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for reactor coolant system (RCS) cold temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by industry operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.
3.4.2.1	12 hours	The SR to verify all RCS temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.
3.4.3.1	30 minutes	This Frequency of 30 minutes is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.
3.4.5.1	72 hours	The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.
3.4.5.2	72 hours	The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.
3.4.6.2	24 months	The frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.
3.4.7.1	12 hours	The Frequency of 12 hours is based on industry operating experience. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.
3.4.7.2	12 hours	The Frequency of 12 hours is based on industry operating experience that demonstrates channel failure of pressure monitors is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.
3.4.7.3	12 hours	The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.
3.4.7.4	92 days	The Frequency of 92 days considers instrument reliability, and industry operating experience has shown that it is proper for detecting degradation.
3.4.7.5	92 days	The Frequency of 92 days considers instrument reliability, and industry operating experience has shown that it is proper for detecting degradation.
3.4.7.6	24 months	The Frequency of 24 months considers instrument reliability, and industry operating experience that has proven that this Frequency is acceptable.
3.4.7.7	24 months	The Frequency of 24 months is based on the assumption of a 30 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.
3.4.7.8	24 months	The Frequency of 24 months considers instrument reliability, and industry operating experience that has proven that this Frequency is acceptable.
3.4.8.1	14 days	The 14 day Frequency is adequate to trend changes in the noble gas specific activity level and based on the low probability of an accident occurring during this time period.
3.4.8.2	14 days	The 14 day Frequency is adequate to trend changes in the iodine activity level and based on the low probability of an accident occurring during this time period.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.5.1.1	24 months	The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.5.1.3	24 months	The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.5.2.1	24 hours	The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication and alarm of decay heat removal system (DHRS) level in the control room.
3.5.2.2	24 months	The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.5.3.1	24 hours	Since the ultimate heat sink (UHS) level is normally maintained at a stable level, and is monitored by main control indication and alarm, a 24 hour Frequency is appropriate. This frequency also takes into consideration the high ratio of UHS volume change to UHS level change due to the UHS geometry.
3.5.3.2	24 hours	The Frequency of 24 hours is sufficient to identify a temperature change that would approach either the upper or lower limit of UHS bulk average temperature assumed in the safety analyses. Since the UHS bulk average temperature is normally stable, and is monitored by main control indication and alarm, a 24 hour Frequency is appropriate. This frequency also takes into consideration the large heat capacity of the UHS in comparison to the magnitude of possible heat addition or removal mechanisms.
3.5.3.3	31 days	Since the UHS volume of borated water is large compared to potential dilution sources, the 31 day Frequency is acceptable. In addition, the relatively frequent surveillance of the UHS water volume provides assurance that the UHS boron concentration is not changed significantly.
3.6.2.1	31 days	Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions.
3.6.2.3	24 months	The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Industry operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
3.8.1.1	12 hours	The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for similar neutron detector instruments in LCO 3.3.1.
3.8.1.2	24 months	Industry operating experience has shown that similar components usually pass this Surveillance when performed at the 24 month Frequency.

Enclosure 1:

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.1.1.1	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required shutdown margin (SDM). This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.	Similar to the bases for SR 3.1.1.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The rate of change in SDM during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.1.2.1	31 effective full-power days (EFPDs)	The required subsequent Frequency of 31 EFPDs, following the initial 60 EFPDs after exceeding 5% rated thermal power (RTP), is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g. axial offset (AO)) monitored by the core monitoring system for prompt indication of an anomaly.	Similar to the bases for SR 3.1.2.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The rate of change in core reactivity during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.1.4.1	12 hours	Verification that individual control rod assembly (CRA) positions are within alignment limits at a 12 hour Frequency provides a history that allows the operator to detect a CRA that is beginning to deviate from its expected position. The specified Frequency takes into account other CRA position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.	Similar to the bases for SR 3.1.4.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The relative rate of change in control rod position during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.1.4.2	92 days	The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the CRAs.	Similar to the bases for SR 3.1.4.2 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants.
3.1.5.1	12 hours	Since the shutdown CRAs are not moved during routine operation, except as part of planned surveillances, verification of shutdown CRA position at a Frequency of 12 hours is adequate to ensure that the shutdown CRAs are within their insertion limits. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.	Similar to the bases for SR 3.1.5.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.1.6.1	12 hours	Verification of the regulating group insertion limits at a Frequency of 12 hours is sufficient to detect CRA that may be approaching the insertion limits since, normally, very little rod motion is expected to occur in 12 hours.	Similar to the bases for SR 3.1.6.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The relative rate of change in control rod position during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.1.8.1	30 minutes	Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.	Similar to the bases for SR 3.1.8.3 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The relative rate of change in RTP during the performance of PHYSICS TESTS is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.1.8.2	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.	Similar to the bases for SR 3.1.8.4 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The rate of change in SDM during the performance of PHYSICS TESTS is not expected to be significantly different for NuScale compared to Westinghouse Plants.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.1.9.1	31 days	A 31 day Frequency is considered reasonable in view of other administrative controls that will ensure a misconfiguration of the chemical and volume control system (CVCS) makeup pump demineralized water flow path is unlikely. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of CVCS makeup pump demineralized water flow path configuration.	NuScale Unique. Somewhat similar to the bases for SR 3.6.6B.1 in NUREG-1431.
3.1.9.2	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment. The actuation logic is tested as part of Engineered Safety Features Actuation System (ESFAS) Actuation and Logic testing, and valve performance is monitored as part of the Inservice Testing Program.	NuScale Unique. Somewhat similar to the bases for SR 3.5.2.5 in NUREG-1431. This base SR frequency is reasonable because the NuScale demineralized water isolation valves are expected to have similar reliability to Westinghouse Plants' equivalent isolation valves.
3.1.9.3	31 days	The 31 day Frequency of this SR was developed considering the known stability of stored borated water and the low probability of any undesired source of diluting pure water. The Frequency takes into account administrative controls that will ensure changes to boron concentration are performed in accordance with written procedures. This frequency also considers the size of the boric acid storage tank and the normally expected demands of boric acid for plant operations.	NuScale Unique. Somewhat similar to the bases for SR 3.6.6E.3 in NUREG-1431. This base SR frequency is reasonable because the NuScale boric acid storage tank is expected to usually be isolated from the demineralized water supply.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.1.9.4	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.	NuScale Unique. Somewhat similar to the bases for SR 3.5.2.5 in NUREG-1431. This base SR frequency is reasonable because the NuScale CVCS makeup pumps are designed to a maximum capacity of 20 GPM, per FSAR Table 9.3.4-1, and only one is expected to be operated at a time during routine operation.
3.2.1.1	31 EFPD	The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this frequency is short enough that the Enthalpy Rise Hot Channel Factor limit cannot be exceeded for any significant period of operation.	Similar to the bases for SR 3.2.2.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The rate of change in power distribution with fuel burnup during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.2.2.1	7 days	The Surveillance Frequency of 7 days is adequate considering that the AO is monitored by a computer and any deviation from requirements is alarmed.	Similar to the bases for SR 3.2.3.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The rate of change in the shape of the axial power profile during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.3.1.1	12 hours	The Frequency of 12 hours is based on industry operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.	NuScale Unique. Somewhat similar to the bases for SR 3.3.1.1 in NUREG-1431. This base SR frequency is reasonable considering that the Module Protection System continuously monitors and compares similar channels and any suspected channel failure is alarmed. The reliability of sensors and signal processing equipment is not expected to be significantly different in the NuScale design compared to Westinghouse Plants.
3.3.1.2	12 hours	The Frequency of every 12 hours is adequate. It is based on industry operating experience, considering industry instrument reliability and operating history data for instrument drift. Together, with engineering judgment, these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than +1% RTP is not expected in any 12 hour period.	Similar to the bases for SR 3.3.1.2 in NUREG-1431. This base SR frequency is reasonable because the neutron monitoring system (NMS) nuclear instruments are similar to the instruments used in Westinghouse Plants. Since NuScale's safety analysis assumed a 1% accuracy, compared to the 2% accuracy assumed by Westinghouse Plants, the base SR frequency is conservatively twice as frequent as the Westinghouse Plant STS because there is not expected to be a significant difference in the rate of drift of these instruments between the NuScale and Westinghouse Plant designs.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.3.1.3	24 months	As appropriate, each channel's response must be verified every 24 months. This test measures the portion of the response time from the sensor to receipt in the digital Module Protection System. The digital processing portions of the Module Protection System are assumed to function in less than 1 second consistent with their design. Equipment actuation is measured through testing required by 3.3.2, 3.3.3, and LCO surveillance requirements associated with the actuated components. Response times cannot be determined during unit operation because sensor inputs are required to be varied to measure response times. Industry experience has shown that these components usually pass this surveillance when performed at the 24 months Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.	NuScale Unique. Somewhat similar to the bases for SR 3.3.1.16 in NUREG-1431. This base SR frequency is reasonable because the reliability of sensors and signal processing equipment is not expected to be significantly different in the NuScale design compared to Westinghouse Plants.
3.3.1.4	24 months	The Frequency is justified by the assumption of a 30 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.	Similar to the bases for SR 3.3.1.12 in NUREG-1431. Per TR-0616-49121 “NuScale Instrument Setpoint Methodology Technical Report” Section 3.2.2, sensor calibration is performed during refueling outages and the drift allowance is based on a 24 month fuel cycle with 25 percent added margin.
3.3.1.5	24 months	The 24 month frequency is acceptable based on consideration of the design reliability of the equipment.	NuScale Unique.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.3.2.1	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. This test frequency is justified based on the system design, which includes the use of continuous diagnostic test features that will report a failure within the logic and actuation system to the operator promptly. The only part of the actuation logic circuitry that is not continuously self-tested is the actuation and priority logic circuit which consists of simple discrete components that are very reliable.	NuScale Unique. Somewhat similar to the bases for SR 3.3.6.1 in NUREG-2194. This base SR frequency is reasonable because the NuScale Reactor Trip System design has several advantages over Westinghouse Plants (such as self diagnostics, internal diversity instead of a separate diverse trip system, etc). See TR-1015-18653-A "Design of the Highly Integrated Protection System Platform" for more information on the NuScale design.
3.3.2.2	24 months	The frequency of 24 months is justified based on industry operating experience with similar equipment.	Similar to the bases for SR 3.3.1.16 in NUREG-1431. This base SR frequency is reasonable because the Reactor Trip System designs are similar in Westinghouse Plants and the NuScale design, with the NuScale design having advantages (such as self diagnostics, internal diversity instead of a separate diverse trip system, etc).
3.3.2.3	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.	NuScale Unique.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.3.2.4	24 months	The Frequency of 24 months is based on the known reliability of similar Functions in licensed designs and the multidivisional redundancy available, and has been shown to be acceptable through industry operating experience.	Similar to the bases for SR 3.3.1.14 in NUREG-1431. This base SR frequency is reasonable because the Reactor Trip Breaker designs are expected to have similar reliability in Westinghouse Plants and the NuScale design.
3.3.3.1	24 months	This test frequency of 24 months is justified based on the system design which includes the use of continuous diagnostic test features that will report a failure within the logic and actuation system to the operator promptly. The only part of the actuation logic circuitry that is not continuously self-tested is the actuation and priority logic circuit which consists of simple discrete components that are very reliable.	NuScale Unique. Somewhat similar to the bases for SR 3.3.6.1 in NUREG-2194. This base SR frequency is reasonable because the NuScale ESFAS design has several advantages over Westinghouse Plants (such as self diagnostics, internal diversity instead of a separate diverse trip system, etc). See TR-1015-18653-A "Design of the Highly Integrated Protection System Platform" for more information on the NuScale design.
3.3.3.2	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.	Similar to the bases for SR 3.3.2.10 in NUREG-1431. This base SR frequency is reasonable because the reliability of individual components in ESFAS in the NuScale design is expected to be similar to those in Westinghouse Plants.
3.3.3.3	24 months	The 24 month frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.	NuScale Unique.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.3.3.4	24 months	The Frequency of 24 months is based on the known reliability of similar Functions in licensed designs and the multidivisional redundancy available, and has been shown to be acceptable through industry operating experience.	Similar to the bases for SR 3.3.1.14 in NUREG-1431. This base SR frequency is reasonable because the reliability of the NuScale pressurizer heater trip breaker is expected to be similar to the reliability of the Westinghouse Plants' Reactor Trip Breaker design.
3.3.4.1	24 months	The Frequency of 24 months is based on the known reliability of similar Functions in licensed designs and the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.	NuScale Unique. Somewhat similar to the bases for SR 3.3.5.1 in NUREG-2194 and various SRs in NUREG-1431 such as 3.3.6.8, 3.3.7.8, 3.3.8.4, etc. This base SR frequency is reasonable because the reliability of individual components in the NuScale design is expected to be similar to those in Westinghouse Plants.
3.3.5.1	24 months	The 24 month frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month frequency is also acceptable based on consideration of the design reliability of the equipment.	Similar to the bases for SR 3.3.4.2 in NUREG-1431. This base SR frequency is reasonable because the reliability of the NuScale Remote Shutdown Station (RSS) and the Westinghouse Remote Shutdown System are expected to be similar. Additionally, the operator's impact on plant safety in the NuScale design is smaller compared to the Westinghouse Plant design.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.3.5.2	24 months	The Frequency of 24 months is based on the use of the data display capability in the control room as part of the normal unit operation and the availability of multiple video display units at the RSS. The Frequency of 24 months is based upon industry operating experience and consistency with control room hardware and software.	Similar to the bases for SR 3.3.18.2 in NUREG-2194. This base SR frequency is reasonable because the reliability of the NuScale RSS and the Westinghouse Remote Shutdown Workstation are expected to be similar. Additionally, the operator's impact on plant safety in the NuScale design is smaller compared to the Westinghouse Plant design.
3.3.5.3	24 months	The 24 month frequency is based on the known reliability of similar Functions in licensed designs and the redundancy available, and has been shown to be acceptable through industry operating experience.	Similar to the bases for SR 3.3.18.3 in NUREG-2194. This base SR frequency is reasonable because the reliability of the NuScale RSS and the Westinghouse Remote Shutdown Workstation are expected to be similar. Additionally, the operator's impact on plant safety in the NuScale design is smaller compared to the Westinghouse Plant design.
3.4.1.1	12 hours	Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by industry operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.	Similar to the bases for SR 3.4.1.1 in NUREG-1431. This base SR frequency is reasonable because the relative rate of change in margin to reactor coolant system (RCS) critical heat flux (CHF) parameter limits due to pressure changes during the course of routine operations is not expected to be significantly different for NuScale compared to Westinghouse Plants.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.4.1.2	12 hours	Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS cold temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by industry operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.	Similar to the bases for SR 3.4.1.2 in NUREG-1431. This base SR frequency is reasonable because the relative rate of change in margin to RCS CHF parameter limits due to temperature increases during the course of routine operations is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.4.2.1	12 hours	The SR to verify all RCS temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.	Similar to the bases for SR 3.4.2.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and control room indications for RCS temperature as Westinghouse Plants.
3.4.3.1	30 minutes	This Frequency of 30 minutes is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.	Similar to the bases for SR 3.4.3.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar control room indications for RCS pressure, RCS temperature, and RCS heatup and cooldown rates as Westinghouse Plants.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.4.5.1	72 hours	The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.	Similar to the bases for SR 3.4.13.1 in NUREG-1431. This base SR frequency is reasonable because RCS Leakage Detection Instrumentation and the risk of loss of integrity of the reactor coolant pressure boundary (RCPB) are not expected to be significantly different between the NuScale and Westinghouse Plant designs.
3.4.5.2	72 hours	The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.	Similar to the bases for SR 3.4.13.2 in NUREG-1431. This base SR frequency is reasonable because the design of the NuScale steam generators is such that the primary to secondary LEAKAGE is expected to be comparable to the Westinghouse Plant steam generator design.
3.4.6.2	24 months	The frequency of 24 months is based on the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.	Similar to the bases for SR 3.1.9.3 in NUREG-2194. This base SR frequency is reasonable because the CVCS designs in the NuScale and Westinghouse Plant are expected to have similar reliability.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.4.7.1	12 hours	The Frequency of 12 hours is based on industry operating experience. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.	Similar to the bases for SR 3.3.1.1 in NUREG-1431. This base SR frequency is reasonable because the NuScale containment evacuation system (CES) condensate monitor is a level monitor expected to be similar in reliability to the Westinghouse Plant steam generator level and pressurizer water level monitors.
3.4.7.2	12 hours	The Frequency of 12 hours is based on industry operating experience that demonstrates channel failure of pressure monitors is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.	Similar to the bases for SR 3.3.1.1 in NUREG-1431. This base SR frequency is reasonable because the NuScale CES inlet pressure monitor is expected to have a similar reliability as the pressure monitors in the Westinghouse Plant design.
3.4.7.3	12 hours	The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.	Similar to the bases for SR 3.4.15.1 in NUREG-1431. This base SR frequency is reasonable because the NuScale CES gaseous radioactivity monitor and the Westinghouse Plant containment atmosphere radioactivity monitor are expected to have similar reliability.
3.4.7.4	92 days	The Frequency of 92 days considers instrument reliability, and industry operating experience has shown that it is proper for detecting degradation.	Similar to the bases for SR 3.4.15.2 in NUREG-1431. This base SR frequency is reasonable because the NuScale CES gaseous radioactivity monitor and the Westinghouse Plant containment atmosphere radioactivity monitor are expected to have similar reliability.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.4.7.5	92 days	The Frequency of 92 days considers instrument reliability, and industry operating experience has shown that it is proper for detecting degradation.	Similar to the bases for SRs 3.4.15.2 and 3.4.15.3 in NUREG-1431. This base SR frequency is reasonable because the NuScale CES condensate monitor and the Westinghouse Plant containment sump monitor are expected to have similar reliability.
3.4.7.6	24 months	The Frequency of 24 months considers instrument reliability, and industry operating experience that has proven that this Frequency is acceptable.	Similar to the bases for SR 3.4.15.3 in NUREG-1431. This base SR frequency is reasonable because the NuScale CES condensate monitor and the Westinghouse Plant containment sump monitor are expected to have similar reliability.
3.4.7.7	24 months	The Frequency of 24 months is based on the assumption of a 30 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.	Similar to the bases for SR 3.3.1.10 in NUREG-1431. This base SR frequency is reasonable because the NuScale CES inlet pressure monitor pressure monitors in the Westinghouse Plant design are expected to have similar reliability.
3.4.7.8	24 months	The Frequency of 24 months considers instrument reliability, and industry operating experience that has proven that this Frequency is acceptable.	Similar to the bases for SR 3.4.15.4 in NUREG-1431. This base SR frequency is reasonable because the NuScale CES gaseous radioactivity monitor and the Westinghouse Plant containment atmosphere radioactivity monitor are expected to have similar reliability.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.4.8.1	14 days	The 14 day Frequency is adequate to trend changes in the noble gas specific activity level and based on the low probability of an accident occurring during this time period.	Similar to the bases for SR 3.4.16.1 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The relative rate of change in RCS specific activity during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.4.8.2	14 days	The 14 day Frequency is adequate to trend changes in the iodine activity level and based on the low probability of an accident occurring during this time period.	Similar to the bases for SR 3.4.16.2 in NUREG-1431. This base SR frequency is reasonable because NuScale uses similar fuel and reactivity control processes as Westinghouse Plants. The relative rate of change in RCS specific activity during routine operation is not expected to be significantly different for NuScale compared to Westinghouse Plants.
3.5.1.1	24 months	The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.	Similar to the bases for SR 3.5.2.5 in NUREG-1431. This base SR frequency is reasonable because NuScale's ECCS and the Westinghouse Plant ECCS are expected to have similar reliability.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.5.1.3	24 months	The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.	Similar to the bases for SR 3.5.2.5 in NUREG-1431. This base SR frequency is reasonable because NuScale's ECCS and the Westinghouse Plant ECCS are expected to have similar reliability.
3.5.2.1	24 hours	The 24 hour Frequency is based on the expected low rate of gas accumulation and the availability of control room indication and alarm of decay heat removal system (DHRS) level in the control room.	Similar to the bases for SR 3.5.4.3 in NUREG-2194. This base SR frequency is reasonable because NuScale's DHRS is expected to have a similar reliability as the passive residual heat removal heat exchanger in the AP1000 design.
3.5.2.2	24 months	The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.	Similar to the bases for SR 3.5.2.5 in NUREG-1431. This base SR frequency is reasonable because NuScale's DHRS and the Westinghouse Plant ECCS are expected to have similar reliability.
3.5.3.1	24 hours	Since the ultimate heat sink (UHS) level is normally maintained at a stable level, and is monitored by main control indication and alarm, a 24 hour Frequency is appropriate. This frequency also takes into consideration the high ratio of UHS volume change to UHS level change due to the UHS geometry.	Similar to the bases for SR 3.5.6.2 in NUREG-2194. This base SR frequency is reasonable because the operating band is large due to the very large volume of water in the UHS.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.5.3.2	24 hours	The Frequency of 24 hours is sufficient to identify a temperature change that would approach either the upper or lower limit of UHS bulk average temperature assumed in the safety analyses. Since the UHS bulk average temperature is normally stable, and is monitored by main control indication and alarm, a 24 hour Frequency is appropriate. This frequency also takes into consideration the large heat capacity of the UHS in comparison to the magnitude of possible heat addition or removal mechanisms.	Similar to the bases for SR 3.5.4.1 in NUREG-1431. This base SR frequency is reasonable because the operating band is large due to the very large volume of water in the UHS.
3.5.3.3	31 days	Since the UHS volume of borated water is large compared to potential dilution sources, the 31 day Frequency is acceptable. In addition, the relatively frequent surveillance of the UHS water volume provides assurance that the UHS boron concentration is not changed significantly.	Similar to the bases for SR 3.5.6.4 in NUREG-2194. This base SR frequency is reasonable because the operating band is large due to the very large volume of water.
3.6.2.1	31 days	Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions.	Similar to the bases for SR 3.6.3.3 in NUREG-1431. This base SR frequency is reasonable because the probability of misalignment of these containment isolation valves, since they have been verified to be in the proper position, is expected to be small.
3.6.2.3	24 months	The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Industry operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.	Similar to the bases for SR 3.6.3.8 in NUREG-1431. This base SR frequency is reasonable because the NuScale containment isolation valve design is expected to have a similar reliability as the Westinghouse Plant containment isolation valve design.

Surveillance Requirement	Base Frequency	Basis	Comparison to Similar Surveillance Intervals Where Applicable
3.8.1.1	12 hours	The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for similar neutron detector instruments in LCO 3.3.1.	Similar to the bases for SR 3.9.3.1 in NUREG-1431. This base SR frequency is reasonable because the NuScale refueling nuclear instrumentation design is expected to have a similar reliability as the Westinghouse Plant refueling nuclear instrumentation design.
3.8.1.2	24 months	Industry operating experience has shown that similar components usually pass this Surveillance when performed at the 24 month Frequency.	Similar to the bases for SR 3.9.3.2 in NUREG-1431. This base SR frequency is reasonable because the NuScale refueling nuclear instrumentation design is expected to have a similar reliability as the Westinghouse Plant refueling nuclear instrumentation design.

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

eRAI No.: 9034
Date of RAI Issue: 09/14/2017

NRC Question No.: 16-31

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 model standard technical specifications (STS) in the following documents provide NRC guidance on format and content of TS as acceptable means to meet 10 CFR 50.36 requirements. These documents may be accessed using the Agencywide Documents Access and Management Systems (ADAMS) by their accession numbers.

- NUREG-1431, "STS Westinghouse Plants," Revision 4
  (ADAMS Accession Nos. ML12100A222 and ML12100A228)

- NUREG-1432, "STS Combustion Engineering Plants," Revision 4
  (ADAMS Accession Nos. ML12102A165 and ML12102A169)

- NUREG-2194, "STS Westinghouse Advanced Passive 1000 (AP1000) Plants," Revision 0
  (ADAMS Accession No. ML16111A132)

The NRC staff needs to evaluate technical differences in the proposed generic TS (GTS) from applicable provisions in these documents, which are referenced by the DC applicant in Design Control Document (DCD) Tier 2, Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the GTS to ensure adequate protection of public health and safety, and the completeness and accuracy of the GTS Bases.

The statements of the LCO, Condition A, Requied Action A.2, and SR 3.1.2.1 of Subsection 3.1.2 differ from the corresponding statements in the Westinghouse, CE, and AP1000 STS Subsection 3.1.2, "Core Reactivity"; differences in these STS requirements are indicated in the following quotations using blue and red colored font. *The changes indicate how to revise each STS requirement to match the corresponding GTS requirement.*

In the quotation of GTS 3.1.2 requirements, the <u>underlined</u> blue colored font indicates modification of GTS Frequency requirements, which has been globally requested by the staff in another RAI Question Sub-question. An editorial correction to GTS 3.1.2 Required Action A.2 is indicated by ~~lined out~~ red colored font.

a. The applicant is requested to justify the GTS phrasing over the phrasing of the three STS subsections for each GTS provision quoted.

b. The applicant is also requested to compare the GTS 3.1.2 Bases against the Bases of the three STS subsections and, for all phrasing differences, justify the GTS Bases phrasing over the phrasing of the three STS Bases subsections.

NuScale GTS

LCO 3.1.2	The core reactivity balance shall be within ±1% Δk/k of the normalized predicted values.
Condition A	Core reactivity not within limit.
Required Action A.2	Establish~~ed~~ appropriate operating restrictions. | 7 days
SR 3.1.2.1	
Note	Predicted reactivity values may be adjusted to correspond to measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.
Surveillance	Verify overall core reactivity balance is within ±1% Δk/k of predicted values.
Frequency	Once prior to exceeding 5% RTP after each refueling AND ----------------NOTE--------------- Only required after 60 EFPD -------------------------------------- [31 EFPD thereafter OR In accordance with the Surveillance Frequency Control Program]

Westinghouse STS (revised to match GTS)

LCO 3.1.2 | The ~~measured~~ core reactivity balance shall be within ± 1% Δk/k of the normalized predicted values.

Condition A | ~~Measured core~~ Core reactivity not within limit.

Required Action A.2 | Establish appropriate operating restrictions ~~and SRs~~. | 7 days

SR 3.1.2.1

Note: ~~The predicted~~ Predicted reactivity values may be adjusted ~~(normalized)~~ to correspond to ~~the~~ measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.

Surveillance: Verify overall ~~measured~~ core reactivity balance is within ±1% Δk/k of predicted values.

Frequency: Once prior to exceeding 5% RTP ~~entering MODE 1~~ after each refueling

AND

---------------NOTE---------------
Only required after 60 EFPD

[31 EFPD thereafter

OR

In accordance with the Surveillance Frequency Control Program]

CE STS (revised to match GTS)

LCO 3.1.2 | The core reactivity balance shall be within ± 1% Δk/k of the normalized predicted values.

Condition A | Core reactivity ~~balance~~ not within limit.

Required Action A.2 | Establish appropriate operating restrictions ~~and SRs~~. | 7 days

SR 3.1.2.1

Note 1	~~The predicted~~ Predicted reactivity values may be adjusted ~~(normalized)~~ to correspond to ~~the~~ measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.
~~Note 2~~	~~This Surveillance is not required to be performed prior to entry into MODE 2.~~
Surveillance	Verify overall core reactivity balance is within ±1.0% Δk/k of predicted values.
Frequency	Once prior to ~~Prior to entering MODE 1 after fuel loading~~ exceeding 5% RTP after each refueling AND ----------------NOTE--------------- Only required after 60 EFPD --------------------------------------- [31 EFPD thereafter OR In accordance with the Surveillance Frequency Control Program]

AP1000 STS (revised to match GTS)

LCO 3.1.2	The ~~measured~~ core reactivity balance shall be within ± 1% Δk/k of the normalized predicted values.
Condition A	~~Measured core~~ Core reactivity not within limit.
Required Action A.2	Establish appropriate operating restrictions ~~and SRs~~. | 7 days
SR 3.1.2.1	
Note	~~The predicted~~ Predicted reactivity values may be adjusted ~~(normalized)~~ to correspond to ~~the~~ measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading.
Surveillance	Verify overall measured core reactivity balance is within ±1% Δk/k of predicted values.

Frequency	Once prior to entering MODE 1 exceeding 5% RTP after each refueling <u>AND</u> ----------------NOTE--------------- Only required ~~to be performed~~ after 60 EFPD --------------------------------------- 31 EFPD thereafter

NuScale Response:

The NuScale plant and core design are not the same as the Westinghouse, CE, or AP1000 designs. The LCO, Actions, and Surveillance Requirement were prepared with STS input as the basis for the limit. However the LCO was clarified to appropriately align with the NuScale usage. Similarly, the Bases reflect the NuScale design and the 3.1.2 LCO.

The word 'balance' has been added to Condition A. The typographical error in Required Action A.2 is being corrected. See Table 1.1-1, MODES regarding the change to the Note that removed the MODE 2 exception that exists in the STS.

Impact on DCA:

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The core reactivity balance shall be within ±1% Δk/k of the normalized predicted values.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Core reactivity balance not within limit.	A.1	Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation.	7 days
	AND		
	A.2	Establish~~ed~~ appropriate operating restrictions.	7 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 2.	6 hours

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

eRAI No.: 9034
Date of RAI Issue: 09/14/2017

NRC Question No.: 16-32

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 model standard technical specifications (STS) in the following documents provide NRC guidance on format and content of TS as acceptable means to meet 10 CFR 50.36 requirements. These documents may be accessed using the Agencywide Documents Access and Management Systems (ADAMS) by their accession numbers.

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  (ADAMS Accession No. ML16111A132)

The NRC staff needs to evaluate technical differences in the proposed generic TS (GTS) from applicable provisions in these documents, which are referenced by the DC applicant in Design Control Document (DCD) Tier 2, Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the GTS to ensure adequate protection of public health and safety, and the completeness and accuracy of the GTS Bases.

The applicant is requested to describe the operational steps taken to shutdown a MODULE within 48 hours beginning from MODE 1 at RTP and ending in MODE 3 with all reactor coolant system temperatures less than 200 degrees F. Discuss why 48 hours is specified to perform this evolution upon entering Specification 3.1.3, "Moderator Temperature Coefficient (MTC)," Condition B in accordance with Required Action B.1. Since Specification 3.8.2, "Decay Time" only requires 48 hours to have elapsed since reactor shutdown (since the reactor was last critical) before allowing irradiated fuel movement in the reactor vessel (which is only possible in

MODE 5), it appears that Required Action B.1 could still be completed “in an orderly manner and without challenging plant systems” in a time much less than 48 hours.

> The associated Bases for Required Action B.1 state, “The allowed Completion Time is a reasonable time based on the activities needed to reach the required MODE from full power operation in an orderly manner and without challenging plant systems.”

Also, correct typo by inserting “on” as indicated.

Explain why “required MODE” is used instead of “required MODULE conditions.”

Please also address your response to the same, or similar, shutdown action completion times in the following action requirements. Explain the many variations in the rationale for the same completion time of the same shutdown action. The Bases for the following shutdown action requirements appear to be inconsistent regarding reliance on the use of safety systems to cooldown the MODULE in MODE 3. The requested explanations under each set of action requirements that are similar may be provided jointly.

1. 3.3.1, MODULE Protection System Instrumentation,

Function 7.b, DHRS actuation on High Pressurizer Pressure

Function 9.b, DHRS actuation on Low Low Pressurizer Pressure

Function 13.b, DHRS actuation on High Narrow Range RCS Hot Temperature

Function 16.a, ECCS actuation on Low RPV Riser Level

Function 17.b, DHRS actuation on High Main Steam Pressure

Function 22.b, DHRS actuation on High Narrow Range Containment Pressure

Function 23.a, ECCS actuation on High Containment Water Level

Required Action I.1: Be in MODE 2. | 6 hours

Required Action I.2: Be in MODE 3 and PASSIVELY COOLED. | 36 hours

> The associated Bases for Required Actions I.1 and I.2 state, “Completion Times are established considering the likelihood of a LOCA event that would require ECCS or DHRS actuation. They also provide adequate time to permit evaluation of conditions and restoration of actuation logic OPERABILITY without unnecessarily challenging plant systems during a shutdown.”

Explain why "actuation logic OPERABILITY" restoration is mentioned instead of "channel OPERABILITY" restoration.

Explain why "unnecessarily" is needed.

Function 10.b, DHRS actuation on Low Low Main Steam Pressure

Required Action K.1: Be in MODE 2. | 6 hours

Required Action K.2: Be in MODE 3. | 36 hours

The associated Bases for Required Actions K.1 and K.2 state, "The allowed Completion Times are reasonable to reach the required MODULE conditions from full power conditions in an orderly manner and without challenging MODULE systems."

Explain why "MODULE systems" is used instead of "plant systems."

Function 22.c, CIS actuation on High Narrow Range Containment Pressure,

Required Action L.1: Be in MODE 2. | 6 hours

Required Action L.2: Be in MODE 3 with RCS temperature hot < 200°F. | 48 hours

The associated Bases for Required Actions L.1 and L.2 state, "Completion Times are established considering the likelihood of a design basis event that would require CIS actuation during the period of inoperability. They also provide adequate time to permit evaluation of conditions and restoration of logic OPERABILITY without unnecessarily challenging plant systems during a shutdown. Analysis shows that 48 hours from entry into this condition is a reasonable time to reach MODE 3 with RCS wide range temperature hot < 200°F using normal plant systems and procedures.

Explain why "logic OPERABILITY" restoration is mentioned instead of "channel OPERABILITY" restoration.

Explain why "unnecessarily" is needed.

Explain why last sentence says "using normal plant systems and procedures" instead of "using only safety-related plant systems and procedures."

Function 25, Low AC Voltage to ELVS Battery Chargers, and

Function 26, High Under-the-Bioshield Temperature,

Each of which causes actuation of RTS, DHRS, CIS, DWSI, and Pressurizer Heater Trip(PHT)

Required Action M.1: Be in MODE 2. | 72 hours

Required Action M.2: Be in MODE 3 and PASSIVELY COOLED. | 96 hours

Required Action M.3: Be in MODE 3 with RCS temperature hot < 200°F. | 96 hours

The associated Bases for Required Actions M.2 and M.3 state, "Completion Times are established considering the likelihood of a design basis event that would require automatic actuation during the period of inoperability. They also provide adequate time to permit evaluation of conditions and restoration of logic OPERABILITY without unnecessarily challenging plant systems during a shutdown.

Explain why "logic OPERABILITY" restoration is mentioned instead of "channel OPERABILITY" restoration.

Explain why "unnecessarily" is needed.

Explain why 72 hours are allowed to be in MODE 2; and 96 hours are allowed to be in MODE 3 and PASSIVELY COOLED, and in MODE 3 below 200°F.

2. 3.3.3, ESFAS Logic and Actuation, Function 1, ECCS, and Function 2, DHRS,

Required Action C.1: Be in MODE 2. | 6 hours

Required Action C.2: Be in MODE 3 and PASSIVELY COOLED. | 36 hours

The associated Bases for Required Actions C.1 and C.2 state, "Completion Times are established considering the likelihood of a LOCA event that would require ECCS or DHRS actuation. They also provide adequate time to permit evaluation of conditions and restoration of *actuation* logic OPERABILITY without unnecessarily challenging plant systems during a shutdown.

Explain why "unnecessarily" is needed.

3. 3.3.3, ESFAS Logic and Actuation, Function 3, CIS

Required Action D.1: Be in MODE 2. | 6 hours

Required Action D.2: Be in MODE 3 with RCS temperature hot < 200°F. | 48 hours

Explain omission of "hot" in D.2, or insert "hot" as indicated. Also, address this apparent omission in the discussion of CIS Actuation in the Applicable Safety Analyses, LCO, and Applicability section of Bases Subsection B 3.3.3.

> The associated Bases for Required Actions D.1 and D.2 state, "Completion Times are established considering the *limited likelihood* of a design basis event that would require CIS actuation during the period of inoperability. They also provide adequate time to permit evaluation of conditions and restoration of logic OPERABILITY without unnecessarily challenging plant systems during a shutdown. Analysis shows that 48 hours from entry into this condition is a reasonable time to reach MODE 3 with RCS wide range ~~Thot~~ T_{hot} < 200°F using normal plant systems and procedures.

Explain why "limited likelihood" is used instead of "low probability," "low likelihood," or small probability," etc.

Explain why "logic OPERABILITY" and not "actuation logic OPERABILITY" is used, or add "actuation."

Explain why "unnecessarily" is needed.

Correct typo "Thot" in last sentence, as indicated, or replace with "temperature hot."

Explain why last sentence says "using normal plant systems and procedures" instead of "using only safety-related plant systems and procedures."

4. 3.3.3, ESFAS Logic and Actuation, Actions section of Bases Subsection B 3.3.3, for Action B C, D, E, F, and G include a paragraph similar to the following paragraph:

 The redundant signal paths and logic of the OPERABLE division provides robust capability to automatically actuate the required ESFAS function with a single division of logic OPERABLE.

 The Bases for Actions C, D, E, F, and G include a similar paragraph, as follows:

> With one division of *actuation* logic inoperable, the redundant signal paths and logic of the OPERABLE division provide robust capability to automatically actuate the [ECCS, DHRS, CIS, DWSI, CVCSI, or PHT] if required.

Note that the paragraph for Action C inadvertently omits the word “actuation” in the opening phrase; this should be changed to match the other paragraphs.

Explain why the subjective phrase “robust capability” is used instead of the objective phrase “sufficient capability” in these paragraphs, or replace “robust” with “sufficient.”

5. 3.3.4, Manual Actuation Functions, Function 2, ECCS, and Function 3, DHRS,

   Required Action D.1: Be in MODE 2. | 24 hours

   Required Action D.2: Be in MODE 3 and PASSIVELY COOLED. | 72 hours

   > The associated Bases for Required Actions D.1 and D.2 state, “...Condition D provides 24 hours to restore the manual actuation capability to OPERABLE status before the MODULE must be in MODE 2. ~~The Actions~~ Required Action D.2 requires the MODULE be in MODE 3 and PASSIVELY COOLED within 72 hours of entering the condition. The Completion Times provide opportunity for correction of the identified inoperability while maintaining the reactor coolant system closed, minimizing the transients and complexity of a return to operation when OPERABILITY is restored.”
   >
   > “The Completion Times are reasonable because the credited automatic actuation function remains OPERABLE as specified in LCO 3.3.3, and alternative means of manually initiating the safety function remain available, e.g., manually initiating individual MPS division trip logic and component-level actuations.”

   Correct typo as indicated.

   Explain why LCO 3.3.4 only addresses division level manual actuation controls, and not component-level manual actuation controls.

   *Explain why Condition B, for both manual actuation divisions inoperable (loss of division level system manual actuation capability), allows 6 hours **followed by** the 24 hour allowance of Condition D to be in MODE 2, without other Required Actions, which verify OPERABILITY of the automatic actuation logic for ECCS and DHRS, as well as component-level manual actuation controls. Explain why 72 hours is needed to be in MODE 3 and PASSIVELY COOLED.*

   Explain the phrase “while maintaining the reactor coolant system closed, minimizing the transients and complexity of a return to operation.”

6. 3.3.4, Manual Actuation Functions, Function 4, Containment Isolation System,

   Required Action I.1: Be in MODE 2. | 6 hours

   Required Action I.2: Be in MODE 3 with RCS temperature hot < 200°F. | 48 hours

The associated Bases for Required Actions I.1 and I.2 state, "...the MODULE must be placed in MODE 2 within 6 hours and in MODE 3 with the RCS temperature hot < 200 °F within 48 hours. Reducing the RCS temperature to < 200 °F places the MODULE in a MODE or specified condition in which the LCO no longer applies."

"The Completion Times are reasonable because the credited automatic actuation function remains OPERABLE as specified in LCO 3.3.3, and alternative means of manually initiating the safety function remain available, e.g., manually initiating individual MPS division trip logic and component-level actuations.

Explain why an inoperable ECCS or DHRS manual initiation gets 24 hours to be in MODE 2 and 72 hours to be in MODE 3 and PASSIVELY COOLED, but an inoperable CIS manual initiation gets 6 hours to be in MODE 2 and 48 hours to be in MODE 3 with ~~RCS temperature hot~~ all RCS temperatures below 200 °F.

Explain why Required Action I.2 says "RCS temperature hot < 200°F" instead of "All RCS temperatures < 200°F," as indicated, to be more consistent with Footnote (c), which says, "With any RCS temperature ≥200° F."

Correct the typo in Footnote (c), which should say "...≥200 °F" or "...≥200°F" according to the convention used in the NuScale DCA.

7. 3.3.4, Manual Actuation Functions, Function 7, Pressurizer Heater Trip

Explain why the 48 hours of Condition A, or the 6 hours of Condition B, followed by the 24 hours of Condition G are needed before manually opening the pressurizer heater breakers (de-energizing pressurizer heaters).

*Required Action G.1 should say "De-energize **affected** pressurizer heaters." because Condition G can be entered from Condition A.*

8. 3.4.5, RCS Operational LEAKAGE,

Required Action B.1: Be in MODE 2. | 6 hours

Required Action B.2: Be in MODE 3 with RCS temperature hot < 200 °F. | 48 hours

The associated Bases for Required Actions B.1 and B.2 state, "The allowed Completion Times are reasonable, based on operating requirements and normal cooling capabilities, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems."

Explain whether “normal cooling capabilities” includes non-safety systems.

Explain why “required plant conditions” is used instead of “required MODULE conditions.”

Explain why last sentence says “based on operating requirements and normal plant systems and procedures” instead of “based on operating requirements and only safety-related plant systems and procedures.”

9. 3.4.6, Chemical and Volume Control System Isolation Valves,

   Required Action C.1: Be in MODE 2. | 6 hours

   Required Action C.2: Be in MODE 3 with RCS temperature hot < 200°F. | 48 hours

   > The associated Bases for Required Actions C.1 and C.2 do not address the rationale for the Completion Times.

   Add Bases for the Completion Times.

   Correct typo in Required Action A.2, Note 1, by inserting “devices” after “Isolation.”

10. 3.4.7, RCS Leakage Detection Instrumentation,

    Required Action C.1: Be in MODE 2. | 6 hours

    Required Action C.2: Be in MODE 3 with RCS temperature hot < 200°F. | 48 hours

    Explain why “required plant conditions” is used instead of “required MODULE conditions.”

    Explain why last sentence says “based on operating requirements and normal cooling capabilities” instead of “based on operating requirements and only safety-related cooling requirements.”

11. 3.4.8, RCS Specific Activity,

    Required Action C.1: Be in MODE 2. | 6 hours

    Required Action C.2: Be in MODE 3. | 36 hours

The associated Bases for Required Actions C.1 and C.2 state, "If ~~the~~ a Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 12 μCi/gm, the reactor must be brought to MODE 2 within 6 hours and MODE 3 within 36 hours. The allowed Completion Times are reasonable, based on operating requirements, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems."

Correct typo in first line as indicated by markup.

Explain why the last sentence uses "required plant conditions" instead of "required MODULE conditions."

12. 3.4.9, SG Tube Integrity

Required Action B.1: Be in MODE 2. | 6 hours

Required Action B.2: Be in MODE 3 and PASSIVELY COOLED. | 36 hours

The associated Bases for Required Actions B.1 and B.2 state, "The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems."

*Explain why the last sentence uses "*based on operating experience, to reach the desired *plant conditions" instead of "based on operating requirements, to reach the required MODULE conditions."*

13. 3.5.1, ECCS

Required Action C.1: Be in MODE 2. | 6 hours

Required Action C.2: Be in MODE 3 and PASSIVELY COOLED. | 36 hours

The associated Bases for Required Actions C.1 and C.2 state, "If the Required Actions cannot be completed within the associated Completion Times, if two or more RVVs, or both RRVs are inoperable the plant must be placed in a condition that does not rely on the ECCS valves opening. To accomplish this, the plant must be shutdown and placed in a safe condition. To do this the plant is shutdown and enters MODE 2 within 6 hours."

"Additionally, within 36 hours the PASSIVE COOLING must be established to ensure decay heat is removed and transferred to the UHS."

Explain why the rationale for these shudown actions differes from that of similar shutdown actions in other subsections. Note that a rationale for the completion times is not explicitly included.

14. 3.5.2, DHRS

Required Action B.1: Be in MODE 2. | 6 hours

Required Action B.2: Be in MODE 3 and PASSIVELY COOLED. | 36 hours

> The associated Bases for Required Actions B.1 and B.2 state, "If the Required Actions cannot be completed within the associated Completion Time, or if both trains of DHRS are declared inoperable the plant must be placed in a mode that does not rely on the DHRS. To accomplish this the plant must be in MODE 2 within 6 hours and PASSIVE COOLING must be established within 36 hours. This condition ensures decay heat is removed and transferred to the UHS."

Explain why the rationale for these shudown actions differes from that of similar shutdown actions in other subsections. Note that a rationale for the completion times is not explicitly included.

15. 3.5.3, UHS

Required Action D.1: Be in MODE 2. | 6 hours

Required Action D.2: Be in MODE 3. | 36 hours

> The associated Bases for Required Actions B.1 and B.2 state, "If the UHS level or bulk average temperature cannot be returned to within limits within the associated Completion Time, the plant must be brought to a condition where the decay heat of the plant with the potential to be rejected to the UHS is minimized. To achieve this status, the plant must be brought to MODE 2 within 6 hours and MODE 3 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems."

Explain why the last sentence uses "based on operating experience, to reach the required plant conditions" instead of "based on operating requirements, to reach the required MODULE conditions."

16. 3.6.1, Containment,

Required Action B.1: Be in MODE 2. | 6 hours

Required Action B.2: Be in MODE 3 with RCS temperature hot < 200°F. | 48 hours

The associated Bases for Required Actions B.1 and B.2 state, “If containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours and to MODE 3 with RCS temperature hot < 200ºF within 48 hours(Ref. 3). The allowed Completion Times are reasonable, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.”

Explain why the last sentence uses “are reasonable, to reach the required plant conditions” instead of “are reasonable, based on operating requirements, to reach the required MODULE conditions.”

17. 3.6.2, Containment Isolation Valves,

Required Action C.1: Be in MODE 2. | 6 hours

Required Action C.2: Be in MODE 3 with RCS temperature hot < 200°F. | 48 hours

The associated Bases for Required Actions C.1 and C.2 state, “If the Required Actions and associated Completion Times are not met, the plant must be brought to a MODE or condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours and MODE 3 with RCS temperature hot < 200ºF within 48 hours.

Add Bases for the Completion Times.

NuScale Response:

The response is divided into a general discussion of issues identified, followed by specific responses to the remaining issues raised in the introductory discussion as well as the 17 sets of specific issues at the locations identified in the RAI, where needed.

General Discussion of Shutdown Completion Times

This RAI response and discussion is based on the definitions of the NuScale MODES as provided in Table 1.1-1 of the NuScale GTS. Reference to 'industry' MODES are general in nature and meant as generic reference to MODES typically used at PWRs as defined in NUREG-1430 through -1432, and in NUREG-2194. No specific PWR vendor design comparisons are intended.

The unique NuScale design makes direct MODE comparison inappropriate, however the concepts of critical operations, higher energy conditions, and reduced energy conditions are addressed in the discussions that follow. Additionally, the NuScale design includes readily achievable passive cooling and a MODE to address the relocation of a module to the refueling area. A general and subjective comparison of plant conditions in large Legacy PWRs with conditions in a NuScale plant is provided in the table below:

Legacy PWR MODE(s)		**NuScale MODE**	
1 and 2	Reactor Critical Power Operations	1	Reactor Critical Power Operations
3	'Hot' Subcritical Conditions with 'steaming' heat removal	2	'Hot' Subcritical Conditions
4	Reduced Temperature Conditions with forced shutdown cooling flow	3	Reduced Energy
		3 and Passively Cooled	Reduced Energy and Passively Cooled.
5	'Cold' Conditions with forced shutdown cooling flow	3 with RCS Temp < 200F	'Cold' Conditions and Passively Cooled
No Corresponding MODE in legacy PWRs		4	Transition to Refueling Area Passively Cooled
6	Refueling Operations	5	Refueling Operations

NuScale has adopted completion times similar to and that generally align with standard industry completion times for transition to the most stable and cool industry conditions - PWR MODE 5, Cold Shutdown. For the NuScale design this means transition to MODE 3, Passive Shutdown with appropriate 'other specified conditions' depending on the LCO associated with the Condition. For NuScale MODE 3 or MODE 3 with some additional specified conditions represents the reduced energy ‘safe’ condition for the plant.

The NuScale plant is capable of completely passive cooling and the Technical Specifications are designed to protect this capability. When all means of passive cooling are available, MODE 3 represents a low energy shutdown condition from which reaching passive cooling conditions has occurred or is easily achievable and minimally impactful to plant staff and systems. When any of the passive cooling capabilities are challenged the Technical Specifications drive the plant to a MODE or other specified condition that implements one of the remaining means of passive cooling, or otherwise resolves the challenge to availability.

The RAI requests a discussion of the basis for the Completion Times specified to reduce power and exit the APPLICABILITY of the associated LCOs when the requirements of the LCO cannot be met. In general, two sets of Completion Times exist - those that require the NuScale plant to

- Be in MODE 2 in 6 hours, followed by transition to a MODE 3 or MODE 3 with an 'other specified condition that results in exiting the LCO Applicability,' or
- Be in MODE 2 in 6 hours, followed by transition to MODE 3 with the RCS temperature below 200 F.

Enter MODE 2 in 6 hours, and Exit Applicability in MODE 3

The Completion Times are considered appropriate as they result in reaching a condition comparable to those within the nuclear industry's accepted practice. The simplified NuScale plant design with passive safety and a reduced core size assure the risk and safety impact of the Completion Times in the GTS are bounded by existing plant Completion Times.

Completion times in existing plants are generally not established or bound by the capability of the plants to achieve the specified conditions. Rather the time allowed to make MODE transitions ensures that operators have appropriate time to evaluate and respond to conditions in a deliberate and planned manner. The completion times balance risk from temporary and limited operations in the condition with the required plant transient impact on staff and plant equipment. The appropriate completion times permit the transitions to be accomplished using established processes, equipment, and normal plant staff. Although not directly considered in development, the typical industry standard times also provide consideration for the potential need for multi-module actions to be addressed.

In other LCO where the Completion Times are specified to require the plant to reach MODE 3 and an other specified condition such as:

- MODE 3 and passively cooled in 36 hours,
- MODE 3 and <500 psia in 36 hours,
- MODE 3 and < 350 F in 36 hours, or
- MODE 3 and LTOP enabled in 36 hours

Typically, the same completion times are used to reach these conditions as the completion time to achieve MODE 3 only. This is based on the fact that none of these additional requirements is, of itself, a sufficient enough change to require that the allowable time needed to meet the action be extended. Additional actions may be required to ensure the other specified conditions are met, however the effect on overall risk of achieving the condition has minimal impact due to those additional steps.

MODE 3 and < 200 F in 48 hours.

A Required Action to reach MODE 3 and be below 200 F has many of the same requirements as the other MODE 3 transitions. However the completion time is extended to ensure that passive means can be used meet the condition and to ensure that the condition can be maintained using passive, safety related, equipment.

NuScale's ability to cool the plant and achieve conditions below 200F if only safety related equipment is available may be somewhat hampered if the chemical volume control system or the normal steam and feedwater systems are unavailable. Those conditions are considered highly unlikely as discussed in the FSAR, and the plant can be cooled and the fuel is safe, but the completion time may be impacted. These systems functions do not satisfy the 10 CFR 50.36 or risk consideration criteria for inclusion and control by Technical Specifications. However their availability is a consideration in the time allowed to achieve the referenced condition under some conditions.

General Discussion of concern with use of "unnecessarily challenging" in Bases

NuScale had included the phrase “without unnecessarily challenging plant systems” (and a similar variations of this phrase) to its shutdown actions as an extension of the phrase “in an orderly manner”. It was intended to add clarity to the latter by addressing one of the potential consequences of not allowing these shutdowns to be performed in an orderly manner. There are many systems and components that are capable of operating to shutdown the reactor quickly, but those actions place dynamic loads, thermal transients, and other stresses on equipment that are only intended to exist in response to transients and accidents. It is inappropriate to apply these capabilities to what would normally be a routine shutdown if not for the required action in the Technical Specification Actions.

However the addition of this phrase “without unnecessarily challenging plant systems” to the NuScale GTS does create a departure from the existing GTS. For this reason in cases where the phrase “in an orderly manner” is already included it will be removed. In a few places where the bases is referring to the challenge of a plant shutdown the phrase remains, but the word ‘unnecessarily’ has been removed.

With regard to 3.8.2, Decay Time, in General Discussion:

The staff noted that Specification 3.8.2, “Decay Time” only requires 48 hours to have elapsed since reactor shutdown before allowing irradiated fuel movement in the reactor vessel (which is only possible in MODE 5). The implication being that if we need to limit fuel movement in MODE 5 within 48 hours that reaching MODE 3 and <200 F should be possible in much less time. This limit in Specification 3.8.2 is based on the analysis of fuel handling accidents and criterion 2 of 10 CFR 50.36(c)(2)(ii) as described in the Bases for 3.8.2. NuScale analysis supports any potential fuel handling accidents that occur after 48 hours. Therefore the specification protects this limitation of the calculation. It is not based on the assumption that 48 hours represents the expected time for fuel movement.

With regard to sub-issue 1 regarding the Bases for 3.3.1:

In the Bases for Required Actions I.1 and I.2 - 'actuation logic has been replaced with channel. See above for discussion of unnecessarily challenged.'

In the Bases for Required Actions K.1 and K.2 - See response to RAI 9050 16-16 for change in the usage of 'module' throughout GTS. Replaced 'actuation logic' with 'channel'.

In the Bases for Required Actions L.1 and L.2 - 'actuation logic' and 'logic ' replaced with 'channel.' See above for discussion of unnecessarily challenged. Other issue modified.

In the Bases for Required Actions M.1, M.2, and M3 - 'actuation logic' and 'logic ' replaced with 'channel.' See above for discussion of unnecessarily challenged. Additionally, completion times were changed to align other similar required actions completion times.

In the Bases for Required Actions N.1 and N.2 - 'actuation logic' replaced with 'channel'.

With regard to sub-issue 2 regarding the Bases for 3.3.3:

In the Bases for Required Actions C.1 and C.2 - see above for discussion of unnecessarily challenged.

With regard to sub-issue 3 regarding the Bases for 3.3.3:

In the Bases for Required Actions D.1 and D.2 - the bases for D.2 includes a discussion of T-2 as described in RAI 9050 16-23 which addresses the concern with the 'hot' terminology. The term 'limited' was changed to 'low.' The term 'actuation' was added. See above for discussion of unnecessarily challenged. The final question was addressed by deletion of the phrase.

With regard to sub-issue 4 regarding the Bases for 3.3.3:

In the Bases for the Required Actions C.1 and C.2, D.1 and D.2 - the term 'actuation' was added. 'Robust was changed to 'sufficient.'

With regard to sub-issue 5 regarding the Bases for 3.3.4:

In the Bases for the Required Actions D.1 and D.2 - the typo was corrected as indicated. and last phrase removed to resolve the request for an explanation.

The issues related to component-level actuation operability requirements are addressed in detail in the FSAR. Manual component and divisional level actuations are not required to address design basis event or credited in the safety analyses. Divisional level actuations are included in NuScale's GTS as a defense in depth capability in accordance with criterion 4 of 10 CFR 50.36(c)(2)(ii) as described in the Bases for 3.3.4. The likelihood that these switches will be required is extremely limited. For this reason the Completion Times are reasonable based on the function provided by these components and functions.

With regard to sub-issue 6 regarding the Bases for 3.3.4:

In the Bases for Required Actions I.1 and I.2 - In the NuScale design, containment isolation is considered more important to successfully address design bases events than ECCS and DHR. Based on the number of functions, components, and systems effects impacted by this condition, NuScale has used engineering judgement to limit the time frame that manual actuation of CIS can be inoperable as compared to ECCS and DHR.

The other items have been addressed by changes to the Bases as shown.

With regard to sub-issue 7 regarding 3.3.4:

In the Bases for Required Actions G.1 - as described above, the manual actuation function including the manual capability to de-energize the pressurizer heaters is not credited in the design and safety basis. If the automatic actuation function is inoperable, the appropriate condition in LCO 3.3.3 would be entered. Manual actuation was included in NuScale's GTS as a defense in depth capability in accordance with criterion 4 of 10 CFR 50.36(c)(2)(ii) as described in the Bases for 3.3.4. Based on the availability of the credited function the completion time to address this defense-in-depth capability is reasonable.

'Affected' was added to Required Action Condition G.

With regard to sub-issue 8 regarding 3.3.5:

In the Bases for Required Actions B.1 and B.2 - The associated text was modified to reflect a more consistent and appropriate bases.

With regard to sub-issue 9 regarding 3.4.6:

In the Bases for Required Actions C.1 and C.2 - Bases for the completion times have been added.

The typo discussion appears to be in reference to a version of the GTS that was not submitted as part of the DCA. No changes were made to the second note as provided in the submitted DCA which indicates that

> *2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.*

With regard to sub-issue 10 regarding 3.4.7:

In the Bases for Required Actions C.1 and C.2 - see response to RAI 9050 16-16 for change in the use of 'module' throughout GTS. Additional changes were made to the bases to address the second issue.

With regard to sub-issue 11 regarding 3.4.8:

In the Bases for Required Actions C.1 and C.2 - The typo was corrected. See response to RAI 9050 16-16 for change in the use of 'module' throughout GTS.

With regard to sub-issue 12 regarding 3.4.9:

In the Bases for Required Actions B. 1 and B.2 - The bases have been modified.

With regard to sub-issue 13 regarding 3.5.1:

In the Bases for Required Actions C.1 and C.2 - The bases have been modified to address the concern. See also the general discussion provided above.

With regard to sub-issue 14 regarding 3.5.2:

In the Bases for Required Actions B.1 and B.2 - the bases have been modfiied to address the concern. See also the general discussion provided above.

With regard to sub-issue 15 regarding 3.5.3:

In the Bases for Required Actions C.1 and d.2 - the bases have been modified to address the concern. See also the general discussion provided above.

With regard to sub-issue 16 regarding 3.6.1:

In the Bases for Required Actions B.1 and B.2 - the bases have been modified to address the concern. See also the general discussion provided above.

With regard to sub-issue 17 regarding 3.6.2:

In the Bases for Required Actions C.1 and C.2 - the bases have been modified to address the concern. See also the general discussion provided above.

Impact on DCA:

The Technical Specifications have 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
F. As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1.	F.1	---------------NOTE------------ Flow path(s) may be unisolated intermittently under administrative controls. -- Isolate the flow paths from the CVCS to the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve.	1 hour
G. As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1.	G.1	-------------NOTE------------- Heater(s) may be energized intermittently under administrative controls. ---------------------------------- De-energize affected pressurizer heaters.	24 hours
H. As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1.	H.1	Open two ~~R~~reactor ~~V~~vent ~~V~~valves.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	---------------NOTES------------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ------------------------------------- Verify the affected CVCS flow path is isolated.	Once per 31 days
B. One or more CVCS flow paths with two CVCS valves inoperable.	B.1	Isolate the affected CVCS flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 2.	6 hours
	AND		
	C.2	Be in MODE 3 with RCS hot temperature ~~hot~~ < 200°F.	48 hours

BASES

LCO 3.0.3 (continued)

c. A Condition exists for which the Required Actions have now been performed, or

d. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition was initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit~~MODULE~~ to be in MODE 3 and PASSIVELY COOLED when a shutdown is required during MODE 1 operation. If the unit~~MODULE~~ is in MODE 2 when a shutdown is required, the time limit for entering MODE 3 and PASSIVE COOLING applies. If MODE 2 is entered in less time than allowed, however, the total allowable time to enter MODE 3 and be PASSIVELY COOLED is not reduced. For example, if MODE 2 is entered in 2 hours, then the time allowed for entering MODE 3 and to establish PASSIVE COOLING is the next 35 hours, because the total time for entering MODE 3 and to be PASSIVELY COOLED is not reduced from the allowable limit of 37 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to enter~~reach~~ a lower MODE of operation in less than the total time allowed.

The Completion Times are established considering the limited likelihood of a design basis event during the 37 hours allowed to enter MODE 3 and be PASSIVELY COOLED. They also provide adequate time to permit evaluation of conditions and restoration of OPERABILITY without challenging plant systems during a shutdown. Analysis shows that 37 hours from entry into 3.0.3 is a reasonable time to enter MODE 3 and be PASSIVELY COOLED using normal plant systems and procedures.

In MODES 1, 2, and MODE 3 when not PASSIVELY COOLED, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODE 3 when PASSIVELY COOLED, and MODES 4 and 5 because the unit~~MODULE~~ is already in the most restrictive condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, or MODE 3 when not PASSIVELY COOLED) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to 3.0.3 are provided in instances where requiring a unit~~MODULE~~ shutdown in accordance with LCO 3.0.3, would not

BASES

ACTIONS (continued)

acceptable for continued operation, then the boron letdown curve may be renormalized and power operation may continue. If changes to operational restrictions are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined and implemented.

The required Completion Time of 7 days is adequate for preparing and implementing whatever operating restrictions that may be required to allow continued reactor operation.

B.1

If the core reactivity cannot be restored to within the 1% Δk/k limit, the unit~~plant~~ must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours. If the SDM for MODE 2 is not met, then boration may be required to meet SR 3.1.1.1 prior to entry into MODE 2. The allowed Completion Time is reasonable, for reaching MODE 2 from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted RCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable, including CRA position, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to exceeding 5% RTP as an initial check on core conditions and design calculations at BOC. The Surveillance is performed again prior to exceeding 60 effective full power days (EFPDs) to confirm the core reactivity is responding to reactivity predictions and then periodically thereafter during the operating cycle in accordance with the Surveillance Frequency Control Program. The SR is modified by a Note indicating that the predicted core reactivity may be adjusted to the measured value provided this normalization is performed prior to exceeding a fuel burnup of 60 EFPDs. This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

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

BASES

APPLICABILITY (continued)

In MODE 3 with ~~reactor coolant temperature~~all RCS temperatures < 200°F and in MODES 4 and 5, this LCO is not applicable because no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these conditions.

ACTIONS

A.1

MTC is a function of the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 2. This eliminates the potential for violation of the accident analysis bounds. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, and the time for reaching MODE 2 from full power conditions in an orderly manner~~ and without challenging plant systems~~.

B.1

Operating outside the lower MTC limit means the safety analysis assumptions for the EOC accidents that use a bounding negative MTC value may be invalid. If the lower MTC limit is exceeded, the unit~~plant~~ must be placed in a MODE or condition in which the LCO requirements are not applicable. In addition to Required Action A.1, Required Action B.1 also requires the unit~~plant~~ to be in MODE 3 with ~~reactor coolant temperaure~~all RCS temperatures < 200°F within 48 hours.

The allowed Completion Time is a reasonable time based on the activities needed to reach the required MODE from full power operation in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 and SR 3.1.3.2

The SRs for measurement of the MTC at the beginning and two-thirds of each fuel cycle provide for confirmation of the limiting MTC values. The MTC changes smoothly from least negative to most negative value during fuel cycle operation, as the RCS boron concentration is reduced to compensate for fuel depletion.

The requirement for measurement prior to exceeding > 5% RTP satisfies the confirmatory check on the upper MTC value.

BASES

ACTIONS (continued)

A.2

When Required Action cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 2 within 6 hours, which obviates concerns about the development of undesirable xenon and power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 2 from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that the position of individual rods is within alignment limits allows the operator to detect that a rod is beginning to deviate from its expected position.

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

The SR is modified by a Note that permits it not to be performed for rods associated with an inoperable rod position indicator. The alignment limit is based on rod position indicator which is not available if the indicator is inoperable. LCO 3.1.7, “Rod Position Indication,” provides Actions to verify the rods are in alignment when one or more rod position indicators are inoperable.

SR 3.1.4.2

Verifying each CRA is OPERABLE would require that each CRA be tripped. In MODE 1 tripping each full length CRA would result in radial or axial power tilts, or oscillations. Exercising each individual CRA provides increased confidence that all CRAs continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 4 steps will not cause significant radial or axial power tilts, or oscillations, to occur.

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

Between required performances of SR 3.1.4.2, if a CRA(s) is discovered to be immovable, but remains trippable, the CRA(s) is considered to be OPERABLE. At any time, if a CRA(s) is immovable, a determination of the trippability of the CRA(s) must be made, and appropriate action taken.

BASES

LCO (continued)

normally violate the LCO. This Note applies to each shutdown group as it’s moved below the insertion limit to perform the SR. This Note is not applicable should a malfunction stop performance of the SR.

APPLICABILITY

The shutdown group CRAs must be within their insertion limits, with the reactor in MODE 1. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 2, 3, 4 the shutdown group CRAs are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODES 2, 3, and 4. LCO 3.5.3, "Ultimate Heat Sink," ensures adequate SDM in MODES 4 and 5.

ACTIONS

A.1.1, A.1.2, and A.2

When one shutdown group CRA is not within insertion limits, 2 hours are allowed to restore the shutdown group CRA to within insertion limits. This is necessary because the available SDM may be significantly reduced with one showdown group CRA not within their insertion limits. Also, verification of the SDM or initiation of boration within 1 hour is required, since the SDM in MODE 1 is continuously monitored and adhered to, in part, by the control and shutdown group insertion limits (see LCO 3.1.1).

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the unit~~plant~~ to remain in an unacceptable condition for an extended period of time.

B.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable for reaching the required MODE from full power conditions in an orderly manner ~~and without challenging plant systems~~.

BASES

ACTIONS

A.1.1, A.1.2, and A.2

When the regulating group are outside the acceptance insertion limits, they must be restored to within those limits. This restoration can occur in two ways:

a. Reduce power to be consistent with rod position; or

b. Moving rods to be consistent with power.

Also, verification of SDM or initiation of boration to regain SDM is required in 1 hour, since the SDM in MODE 1 with k_{eff} is ≥ 1.0 is normally ensured by adhering to the control and shutdown group insertion limits (see LCO 3.1.1, "Shutdown Margin (SDM)") has been upset.

The allowed Completion Time of 2 hours for restoring the regulating group to within insertion limits, provides an acceptable time for evaluating and repairing minor problems without allowing the unit~~plant~~ to remain in an unacceptable condition for an extended period of time.

B.1

If the Required Actions cannot be completed within the associated Completion Times, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable for reaching the required MODE from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification of the regulating group insertion limits is sufficient to detect regulating groups that may be approaching the insertion limits.

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

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.

2. 10 CFR 50.46.

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

BASES

ACTIONS (continued)

C.1

The Required Action clarify that when one or more CRAs with inoperable position indicators have been moved in excess of 6 steps in one direction since the position was last determined, the Required Actions of A.1 or B.1 are still appropriate but must be initiated promptly under Required Action C.1 to begin verifying that these CRAs are still properly positioned relative to their group positions.

D.1 and D.2

With one demand position indicator per group inoperable, the CRA positions can be determined by the RPI System. Since normal full power operation does not require excessive movement of CRAs, verification by administrative means that the CRDS position indicators are OPERABLE and the most withdrawn CRA and the least withdrawn CRA are ≤ 6 steps apart within the allowed Completion Time of once every 8 hours is adequate

E.1

If the Required Actions cannot be completed within the associated Completion Time, the unit~~plant~~ must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours. The allowed Completion Time is based on reaching the required MODE from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verification that the Counter Position Indication agrees with the direct-reading RPI and demand position within 6 steps provides assurance that the RPI is operating correctly.

This surveillance is performed prior to reactor criticality after coupling of one or more CRA to the associated CRDM, as there is the potential for unnecessary unit~~plant~~ transients if the SR were performed with the reactor at power.

BASES

ACTIONS (continued)

D.1

Condition D is entered when Condition C applies to the following Functions that result in a reactor trip or DHRS actuation, as listed in Table 3.3.1-1.

- 1a, Power Range Linear Power – High (RTS)
- 3a, Intermediate Range Log Power Rate – High (RTS)
- 4a, Source Range Count Rate – High (RTS)
- 5a, Source Range Log Power Rate – High (RTS)
- 7a, Pressurizer Pressure – High (RTS)
- 8a, Pressurizer Pressure – Low (RTS)
- 8b, Pressurizer Pressure – Low (DHRS)
- 9a, Pressurizer Pressure – Low Low (RTS)
- 10a, Pressurizer Level – High (RTS)
- 11a, Pressurizer Level – Low (RTS)
- 13a, NR RCS Hot Temperature – High (RTS)
- 15a, RCS Flow – Low Low (RTS)
- 17a, Main Steam Pressure – High (RTS)
- 19a, Main Steam Pressure – Low Low (RTS)
- 20a, Steam Superheat – High (RTS)
- 20b, Steam Superheat – High (DHRS)
- 21a, Steam Superheat – Low (RTS)
- 21b, Steam Superheat – Low (DHRS)
- 22a, NR Containment Pressure – High (RTS)

If a~~the~~ Required Action~~s~~ associated with ~~this~~ Condition A or B cannot be completed within the required Completion Time for the referenced MPS Function, or three or more channels of the referenced MPS Function are inoperable, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the LCO and Required Actions for the referenced MPS Function do not apply. This is accomplished by opening the reactor trip breakers. The above MPS Functions that result in a reactor trip or DHRS actuation are not required to be OPERABLE when the reactor trip breakers are open. The ~~allowed~~ Completion Time ~~for D.1~~ of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner~~ without challenging plant systems~~.

BASES

ACTIONS (continued)

E.1

Condition E is entered when Condition C applies to Functions that result in a reactor trip signal when reactor THERMAL POWER is above the N-2L interlock~~≥ 15% of RTP~~, as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the Required Actions do not apply. This is accomplished by reducing THERMAL POWER to ~~< 15% RTP~~below the N-2L interlock. The allowed Completion Time for E.1 of 6 hours is reasonable, based on operating experience, for reaching the required condition from full power conditions in an orderly manner~~ without challenging plant systems~~.

F.1

Condition F is entered when Condition C applies to Functions that result in isolation of the CVCS system as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the Required Actions do not apply. This is accomplished by isolating the CVCS flowpath to the RCS. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for aligning the system in an orderly manner~~ without challenging plant systems~~.

Required Action F.1 is modified by a Note that allows isolated penetration flow paths to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the device controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for isolation is indicated. This allowance permits the isolation signal to be reset when appropriate conditions exist to do so.

BASES

ACTIONS (continued)

G.1

Condition G is entered when Condition C applies to Functions that result in automatic removal of electrical power from the pressurizer heaters as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the Required Actions do not apply. This is accomplished by opening the power supply breakers to the pressurizer heaters. The allowed Completion Time for G.1 of 6 hours is reasonable, based on operating experience, for reaching the required conditions ~~from full power conditions~~ in an orderly manner ~~without challenging plant systems~~.

H.1

Condition H is entered when Condition C applies to Functions that result in automatic isolation of the demineralized water system as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the Required Actions do not apply. This is accomplished by isolating the demineralized water flowpath to the RCS. The allowed Completion Time for H.1 of 1 hour is reasonable, based on operating experience, for reaching the required condition ~~from full power conditions~~ in an orderly manner ~~without challenging plant systems~~.

I.1 and I.2

Condition I is entered when Condition C applies to Functions that result in a DHRS or ECCS actuation, as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the Required Actions do not apply. This is accomplished by Required Actions I.1 and I.2.

I.1 places the unit~~MODULE~~ in MODE 2 within 6 hours. This action limits the time the unit~~MODULE~~ may continue to operate with a limited or inoperable automatic channel~~actuation logic~~.

I.2 requires the unit~~MODULE~~ to be in MODE 3 and PASSIVELY

BASES

ACTIONS (continued)

COOLED within 36 hours of entering the Condition~~CONDITION~~. These conditions assure adequate passive decay heat transfer to the UHS and result in the unit~~MODULE~~ being in a condition for which the LCO no longer applies.

Completion Times are established considering the likelihood of a LOCA event that would require ECCS or DHRS actuation. They also provide adequate time to permit evaluation of conditions and restoration of channel~~actuation logic~~ OPERABILITY without ~~unnecessarily~~ challenging plant systems during a shutdown.

J.1

As listed in Table 3.3.1-1, Condition J is entered when Condition C applies to Function 24.a, "High RCS Pressure - Low Temperature Overpressure Protection (LTOP)," which results in actuation of the LTOP system.~~Functions that result in actuation of the low temperature overpressure protection system as listed in Table 3.3.1-1.~~

If a~~the~~ Required Action~~s~~ associated with ~~this~~ Condition A or B cannot be completed within the required Completion Time, or three or more channels of this Function are inoperable, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the LCO and Required Actions for this Function do not apply. This is accomplished by opening at least two RVVs. The ~~allowed~~ Completion Time ~~for J.1~~ of 1 hour is reasonable, based on operating experience, ~~for reaching the required MODE from full power conditions in an orderly manner without challenging plant systems~~for establishing an RCS vent flow path sufficient to ensure low temperature overpressure protection.

K.1 and K.2

Condition K is entered when Condition C applies to Functions that result in actuation of the DHRS on Low Low Main Steam Pressure as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition~~CONDITION~~ cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE in which the LCO does not apply. This is accomplished by Required Actions K.1 and K.2. K.1 places the unit~~MODULE~~ in MODE 2 within 6 hours. This action limits the time the unit~~MODULE~~ may continue to operate with a limited or inoperable DHRS automatic channel~~actuation logic~~. K.2 places the unit~~MODULE~~ in MODE 3 within 36 hours. The allowed Completion Times are reasonable to reach the required unit~~MODULE~~ conditions from full power conditions in an orderly manner~~ and without challenging unitMODULE systems~~.

BASES

ACTIONS (continued)

L.1 and L.2

Condition L is entered when Condition C applies to Functions that result in actuation of the Containment Isolation system as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition~~CONDITION~~ cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE in which the LCO does not apply. This is accomplished by Required Actions L.1 and L.2. L.1 places the unit~~MODULE~~ in MODE 2 within 6 hours. This action limits the time the unit~~MODULE~~ may continue to operate with a limited or inoperable CIS automatic ~~actuation logic~~channel. L.2 places the unit~~MODULE~~ in MODE 3 with RCS hot temperature~~ hot~~ < 200°F within 48 hours of entering the condition. This condition assures the unit~~MODULE~~ will maintain the RCS depressurized and the unit~~MODULE~~ being in a condition for which the LCO no longer applies.

Completion Times are established considering the likelihood of a design basis event that would require CIS actuation during the period of inoperability. They also provide adequate time to permit evaluation of conditions and restoration of channel~~logic~~ OPERABILITY without ~~unnecessarily~~ challenging plant systems during a shutdown.~~ Analysis shows that 48 hours from entry into this condition is a reasonable time to reach MODE 3 with RCS wide range temperature hot < 200°F using normal plant systems and procedures.~~

M.1, M.2, and M.3

Condition~~CONDITION~~ M is entered when Condition~~CONDITION~~ C applies to Functions that result in a reactor trip, CIS actuation, DHR actuation, DWSI, and Pressurizer Heater Trip due to the Low ELVS Voltage or High Under-the-Bioshield Temperature as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition~~CONDITION~~ cannot be completed within the required Completion Time, the unit~~MODULE~~ must be brought to a MODE or other specified condition where the Required Actions do not apply. This is accomplished by Required Actions M.1, M.2, M.3, M.4, and M.5.

M.1 places the unit~~MODULE~~ in MODE 2 within ~~72~~6 hours. This action limits the time the unit~~MODULE~~ may continue to operate with a limited or inoperable automatic channel~~actuation logic~~. M.2 requires the unit~~MODULE~~ to be in MODE 3 and PASSIVELY COOLED within ~~96~~36 hours of entering the Condition~~CONDITION~~. These conditions assure adequate passive decay heat transfer to the UHS and result in the unit~~MODULE~~ being in a condition for which the DHRS OPERABILITY is no longer required.

BASES

ACTIONS (continued)

M.3 places the unit~~MODULE~~ in MODE 3 with RCS temperature ~~hot < 200°F~~below the T-2 interlock within 36~~96~~ hours of entering the condition. This condition assures the unit~~MODULE~~ will maintain the RCS depressurized and the unit~~MODULE~~ being in a condition for which the LCO no longer applies.

M.4 isolates the demineralized water flowpath to the RCS within ~~96~~36 hours. This completes the function of the DWSI.

M.5 opens the power supply breakers to the pressurizer heaters within ~~96~~36 hours.

Completion Times are established considering the likelihood of a design basis event that would require automatic actuation during the period of inoperability. They also provide adequate time to permit evaluation of conditions and restoration of ~~logic~~channel OPERABILITY without ~~unnecessarily~~ challenging plant systems during a shutdown.

N.1 and N.2

Condition N is entered when Condition C applies to Functions that result in the actuation of DHRS on Low Low Pressurizer Level as listed in Table 3.3.1-1.

If the Required Actions associated with this Condition cannot be completed within the required Completion Time, the unit must be brought to a MODE or other specified condition where the Required Actions do not apply. This is accomplished by Required Actions N.1 and N.2.

N.1 places the unit in MODE 2 within 6 hours. This action limits the time the unit may continue to operate with a limited or inoperable DHRS automatic channel. N.2 places the unit in MODE 3 with RCS temperature below the T-2 interlock or in MODE 3 with Containment Water Level above the L-1 interlock within 48 hours of entering the condition. This condition assures the RCS is in a condition for which the LCO no longer applies.

Completion Times are established considering the likelihood of a design basis event that would require DHRS actuation during the period of inoperability. They also provide adequate time to permit evaluation of conditions and restoration of channel OPERABILITY without challenging plant systems during a shutdown.

BASES

ACTIONS

When the required ESFAS logic for the Actuation Functions listed in Table 3.3.3-1 are inoperable, the unit~~MODULE~~ is outside the safety analysis, if applicable in the current MODE of operation. Required Actions must be initiated to limit the duration of operation or to place the unit~~MODULE~~ in a MODE or other applicable condition in which the Condition no longer applies.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Actuation Function. The Completion Time for the inoperable function will be tracked separately for each function, starting from the time the Condition was entered for that Actuation Function.

A.1

Condition A applies if one or more divisions of the LTOP Logic and Actuation Function are inoperable. The Required Action is to open two reactor vent valves (RVVs) within one hour. This places the unit~~reactor~~ in a condition in which the LCO no longer applies. The one hour completion time provides adequate time to either immediately restore the inoperable logic or take manual action to open the RVVs, which establishes an RCS vent flow path sufficient to ensure low temperature overpressure protection.

B.1

Condition B applies if one division of an ESFAS actuation logic function is inoperable. This Condition is not applicable to LTOP actuation logic.

The redundant signal paths and logic of the OPERABLE division provides ~~robust~~sufficient capability to automatically actuate the required ESFAS function with a single division of logic OPERABLE.

If one division of ACTUATION FUNCTION logic cannot be restored to OPERABILITY within six hours, then the Conditions listed in Table 3.3.3-1 must be entered to limit the duration of operation with an inoperable division and to place the unit~~MODULE~~ in a MODE or other applicable condition in which the LCO no longer applies. The six hour limit provides a reasonable time during which the actuation system may be restored to OPERABILITY.

BASES

ACTIONS (continued)

C.1 and C.2

If Required Action B.1 directs entry into Condition C as specified in Table 3.3.3-1, or if both divisions of ECCS or DHRS are inoperable the unit~~plant~~ is outside its design basis ability to automatically mitigate a postulated event.

With one division of actuation logic inoperable the redundant signal paths and logic of the OPERABLE division provide sufficient~~robust~~ capability to automatically actuate the ECCS or DHRS if required.

C.1 requires the unit~~MODULE~~ to be in MODE 2 within 6. This action limits the time the unit~~MODULE~~ may continue to operate with limited or inoperable automatic actuation logic.

C.2 requires the unit~~MODULE~~ to be in MODE 3 and PASSIVELY COOLED within 36 hours of entering the Condition~~CONDITION~~. This condition assures adequate passive decay heat transfer to the UHS and result in the unit~~MODULE~~ being in a condition for which the LCO no longer applies.

Completion Times are established considering the likelihood of a LOCA event that would require ECCS or DHRS actuation. They also provide adequate time to permit evaluation of conditions and restoration of actuation logic OPERABILITY without ~~unnecessarily~~ challenging plant systems during a shutdown.

D.1 and D.2

If Required Action B.1 directs entry into Condition D as specified in Table 3.3.3-1, or if both divisions of the containment isolation actuation function are inoperable then the unit~~MODULE~~ is outside its design basis ability to automatically mitigate some design basis events.

With one division of actuation logic inoperable, the redundant signal paths and logic of the OPERABLE division provide sufficient~~robust~~ capability to automatically actuate the CIS if required.

D.1 requires the unit~~MODULE~~ to be in MODE 2 within 6 hours of entering the Condition. This action limits the time the unit~~MODULE~~ may continue to operate with limited or inoperable CIS automatic actuation logic.

D.2 requires the unit~~MODULE~~ to be placed in MODE 3 with RCS temperature ~~< 200°F~~below the T-2 interlock within 48 hours of entering the Condition. This condition assures the unit~~MODULE~~ will maintain the RCS depressurized,

BASES

ACTIONS (continued)

and the unit~~MODULE~~ being in a condition for which the LCO no longer applies.

Completion Times are established considering the ~~limited likelihood~~ low probability of a design basis event that would require CIS actuation during the period of inoperability. They also provide adequate time to permit evaluation of conditions and restoration of actuation logic OPERABILITY without ~~unnecessarily~~ challenging plant systems during a shutdown. ~~Analysis shows that 48 hours from entry into this condition is a reasonable time to reach MODE 3 with RCS wide range Thot < 200°F using normal plant systems and procedures.~~

E.1

If Required Action B.1 directs entry into Condition E as specified in Table 3.3.3-1, or if both divisions of demineralized water supply isolation actuation are inoperable then the unit~~MODULE~~ is outside its design basis ability to automatically mitigate some design basis events.

With one division of actuation logic inoperable, the redundant signal paths and logic of the OPERABLE division provide sufficient~~robust~~ capability to automatically actuate the DWSI if required.

In this condition the demineralized water supply flow path(s) to the RCS must be isolated within 1 hour to preclude an inadvertent boron dilution event.

Isolation can be accomplished by manually isolating the demineralized water isolation valve(s). Alternatively, the dilution path may be isolated by closing appropriate isolation valve(s) in the flow path(s) from the demineralized water storage tank to the RCS.

The Required Action is modified by a Note allowing the flow path(s) to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the main control room. In this way, the flow path can be isolated when a need for isolation is indicated.

F.1

If Required Action B.1 directs entry into Condition F as specified in Table 3.3.3-1, or if both divisions of the CVCS isolation actuation function are inoperable then the unit~~MODULE~~ is outside its design basis ability to automatically mitigate some design basis events.

BASES

ACTIONS (continued)

With one division of actuation logic inoperable, the redundant signal paths and logic of the OPERABLE division provide robust capability to automatically actuate the CVCSI if required.

F.1 requires the isolation of flow paths from the CVCS to the reactor coolant system within 1 hour of entering the Condition~~CONDITION~~. The Action is modified by a Note that permits the flow path(s) to be unisolated intermittently under administrative controls. This Note limits the likelihood of an event by requiring additional administrative control of the CVCS flow paths. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the main control room. In this way, the flow path(s) can be isolated when a need for isolation is indicated. This permits the unit~~MODULE~~ to continue to operate while in the Condition.

G.1

If Required Action B.1 directs entry into Condition G as specified in Table 3.3.3-1, or if both divisions of the pressurizer heater trip function are inoperable then the unit~~MODULE~~ is outside its design basis ability to automatically mitigate some design basis events.

With one division of actuation logic inoperable, the redundant signal paths and logic of the OPERABLE division provide ~~robust~~sufficient capability to automatically actuate the PHT if required.

G.1 requires de-energization of the pressurizer heaters within 6 hours of entering the Condition~~CONDITION~~. This action limits the time the unit~~MODULE~~ may continue to operate with limited or inoperable PHT automatic actuation logic. The Action is modified by a Note that permits the heaters to be energized intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the breaker controls, who is in continuous communication with the main control room. In this way, the pressurizer heaters can be de-energized when a need for de-energization is indicated. This permits the unit~~MODULE~~ to continue to operate while in the Condition.

The completion time was established considering the likelihood of a design basis event that would require automatic de-energization.

BASES

ACTIONS (continued)

B.1

Condition B applies to the manual actuation functions identified in Table 3.3.4-1. Condition B addresses the situation where one or more Functions have both manual actuation divisions inoperable. One manual actuation division consists of an actuation switch and the associated hardware (such as contacts and wiring) up to but not including the affected EIMs. EIM OPERABILITY is addressed in LCO 3.3.2 and LCO 3.3.3.

With both manual actuation divisions inoperable, the Condition listed in Table 3.3.4-1 must be entered in 6 hours. In this Condition, the automatic MPS actuations remain available to perform the design basis safety functions consistent with the limits of LCO 3.3.1, 3.3.2, and 3.3.3. The Completion Time of 6 hours provides adequate opportunity to identify and implement corrective actions to restore a manual actuation function without entering the Condition specified in Table 3.3.4-1.

C.1

If Required Actions A.1 or B.1 direct entry into Condition C as specified in Table 3.3.4-1, then the reactor trip breakers must be opened immediately. Opening the reactor trip breakers satisfies the safety function of the system and places the unit~~MODULE~~ in a MODE or specified conditions in which the LCO no longer applies.

The immediate completion time is consistent with the importance of the ability to initiate a manual reactor trip using the actuation function.

D.1 and D.2

If Required Actions A.1 or B.1 direct entry into Condition D as specified in Table 3.3.4-1, then Condition D provides 24 hours to restore the manual actuation capability to OPERABLE status before the unit~~MODULE~~ must be in MODE 2. ~~The Actions~~Required Action D.2 requires the unit~~MODULE~~ be in MODE 3 and PASSIVELY COOLED within 72 hours of entering the condition. The Completion Times provide opportunity for correction of the identified inoperability while maintaining the reactor coolant system closed, minimizing the transients and complexity of a return to operation when OPERABILITY is restored.

The Completion Times are reasonable because the credited automatic actuation function remains OPERABLE as specified in LCO 3.3.3, and

BASES

APPLICABILITY

The instrumentation located in the RSS LCO is applicable in MODES 1, 2, and MODE 3 when not PASSIVELY COOLED. ~~and 2.~~ This is required so that the unit~~MODULE~~ can be monitored to ensure the unit~~MODULE~~ transitions to MODE 3 and PASSIVELY COOLED, and remains stable in MODE 3 and PASSIVELY COOLED for an extended period of time from a location other than the control room.

This LCO is not applicable in MODE 3 and PASSIVELY COOLED, 4 or 5. In these MODES, the unit~~MODULE~~ is already subcritical and in a condition of reduced Reactor Coolant System energy. Under these conditions, considerable time is available to restore necessary instrument functions if control room instruments or other actions are required.~~controls become unavailable.~~

ACTIONS

A.1

Condition A addresses the situation where the instrumentation in the RSS is inoperable. The Required Action is to restore the instrumentation in the RSS to OPERABLE status within 30 days. The Completion Time is based on the system design for maintainability and the low probability of an event that would require evacuation of the control room.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A is not met, the unit~~MODULE~~ must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, the unit~~MODULE~~ must be brought to at least MODE 2 within 6 hours and to MODE 3 and PASSIVELY COOLED within 36 hours.

The allowed Completion Times are reasonable to reach the required unit~~MODULE~~ conditions from full power conditions in an orderly manner~~ and without challenging unit systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

SR 3.3.5.1 verifies that the transfer protocol can be performed and that it performs the required functions. This ensures that if the control room becomes inaccessible, the passive cooling system performance can be monitored and evaluated to verify that the unit~~MODULE~~ is transitioning to MODE 3 and PASSIVELY COOLED, and remains stable once MODE 3 and PASSIVELY COOLED condition is reached from the RSS.

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

BASES

ACTIONS

A.1

With one RSV inoperable, the remaining OPERABLE RSV is capable of providing the necessary overpressure protection. Because of additional design margin, the ASME pressure limit for the RCPB and SGS can also be satisfied with one RSV inoperable.

However, the overall reliability of the pressure relief system is reduced because additional failure of the remaining OPERABLE RSV could result in failure to adequately relieve primary or secondary system pressure during a limiting event. For this reason, continued operation is permitted for a limited time only.

The 72 hour Completion Time to restore the inoperable RSV to OPERABLE status is based on the relief capability of the remaining RSV and the low probability of an event requiring RSV actuation.

B.1 and B.2

If the Required Action of Condition A cannot be met within the required Completion Time or if two RSVs are inoperable, the unit~~plant~~ must be placed in a MODE in which the requirement does not apply. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours and to MODE 3 with RCS cold temperature ≤ LTOP enable temperature within 36 hours. RCS cold temperature is considered ≤ LTOP enabling temperature when two or more RCS cold temperature instruments indicate ≤ LTOP enabling temperature specified in the PTLR.

The allowed Completion Times are reasonable based on time to reach the required unit~~plant~~ conditions from full power conditions in an orderly manner~~ and without challenging plant systems~~. The change from MODE 1, or 2, to MODE 3 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer in-surges, and thereby removes the need for overpressure protection by the RSVs.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

SRs are specified in the INSERVICE TESTING PROGRAM. RSVs are to be tested in accordance with the requirements of ASME OM Code (Ref. 3), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The RSV setpoint is ± 3% for OPERABILITY, and the values are reset to remain within ± 1% during the surveillance to allow for drift.

BASES

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce RCS Operational LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1, B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the RCS Operational LEAKAGE and its potential consequences. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours and exit the Applicability in MODE 3 with RCS hot temperature ~~hot~~ ≤ 200~~-~~°F, within 48 hours. The allowed Completion Times are reasonable~~, based on operating requirements and normal cooling capabilities,~~ to reach the required unit~~plant~~ conditions from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

Verifying RCS Operational LEAKAGE is within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE.

Unidentified LEAKAGE and identified LEAKAGE are determined by performance of a RCS water inventory balance. The RCS water inventory balance must be met with the reactor at steady state operating conditions.

Two Notes modify SR 3.4.5.1. The first Note states the SR is not required to be performed until 12 hours after establishing steady state operation. The 12 allowance provides sufficient time to collect and process all necessary data after stable unit~~plant~~ conditions are established. The second Note states the SR is not applicable to primary to secondary LEAKAGE. SR 3.4.5.2 verifies the primary to secondary LEAKAGE.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, and makeup or letdown.

BASES

ACTIONS (continued)

isolation device that cannot be adversely affected by a single active failure. Isolation devices that meet this criterion are a closed and deactivated automatic valve, a closed manual valve, and a blind flange.

The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.2. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the devices are operated under administrative controls and the probability of the misalignment is low.

C.1 and C.2

If the Required Actions and associated completion Times are not met, the unit~~plant~~ must be brought to a MODE or condition in which containment isolation requirement no longer applies. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours and MODE 3 with RCS hot temperature ~~hot~~ < 200ºF within 48 hours. The allowed Completion Times are reasonable to reach the required unit conditions from full power conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

Verifying that the isolation time of each automatic power operated CVCS isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis.

A Note is provided that indicates that the SR is not required to be met when valves are closed or open under administrative controls. This is acceptable because of the slowly occurring nature of the design basis events the CVCS isolation function mitigates. Frequency of this SR is in accordance with the INSERVICE TESTING PROGRAM.

BASES

ACTIONS (continued)

Additionally, the periodic surveillance for RCS water inventory balance, SR 3.4.5.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.5.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit~~plant~~ conditions are established.

Restoration of the channel to OPERABLE status is required to regain the function in a Completion Time of 14 days after the channel's failure. This time is acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1

With one required leakage detection method inoperable, the remaining OPERABLE method will provide indication of changes in leakage. Additionally, Action A.1 will continue to apply and the periodic surveillance for RCS water inventory balance, SR 3.4.5.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

However diversity of leakage detection instrumentation is not available. In addition to the Required Actions of Condition A, the required leakage method is required to regain the function in a Completion Time of 72 hours after the method's failure. This time is acceptable considering the frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

C.1 and C.2

If the Required Action cannot be met within the required Completion Time or if all required leakage detection methods are inoperable, the unit~~plant~~ must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours and to MODE 3 with RCS hot temperature~~ hot~~ <200°F within 48 hours. This action will place the RCS in a low pressure state which reduces the likelihood of leakage and crack propagation. The allowed Completion Times are reasonable, based on operating requirements and normal cooling capabilities, to reach the required unit~~plant~~ conditions from full power conditions in an orderly manner~~ and without challenging plant systems~~.

BASES

ACTIONS (continued)

A Note to the Required Action of Condition A states that LCO 3.0.4.c is applicable. This exception allows entry into the applicable MODE(S) when an allowance is stated in the ACTIONS even though the ACTIONS may eventually require unit~~plant~~ shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit~~plant~~ remains at, or proceeds to power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a small line break occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODES, relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the unit~~plant~~ remains at, or proceeds to, power operation.

C.1 and C.2

If ~~the~~a Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > 12 μCi/gm, the reactor must be brought to MODE 2 within 6 hours and MODE 3 within 36 hours. The allowed Completion Times are reasonable, based on operating requirements, to reach the required unit~~plant~~ conditions from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

SR 3.4.8.1 requires performing a gamma isotopic analysis and calculating the DOSE EQUIVALENT XE-133 using the dose conversion factors in the DOSE EQUIVALENT XE-133 definition. This measurement is the sum of

BASES

ACTIONS (continued)

performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of unit~~plant~~ operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows unit~~plant~~ operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 3 following the next unit refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 2 within 6 hours and MODE 3 and PASSIVELY COOLED within 36 hours.

The allowed Completion Times are reasonable, based on operating requirements~~experience~~, to reach the desired unit~~plant~~ conditions from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the “as found” condition of the SG tubes. The purpose of the condition

BASES

ACTIONS (continued)

C.1 and C.2

If the Required Actions cannot be completed within the associated Completion Times, if two or more RVVs, or both RRVs are inoperable the unit~~plant~~ must be placed in a condition that does not rely on the ECCS valves opening. To accomplish this, the unit~~plant~~ must be shutdown and placed in a safe condition. ~~To do this the plant is shutdown and enters MODE 2 within 6 hours.~~

~~Additionally, within 36 hours the PASSIVE COOLING must be established to ensure decay heat is removed and transferred to the UHS.~~This is accomplished by Required Actions C.1 and C.2.

Required Action C.1 places the unit in MODE 2 within 6 hours. Required Action C.2 places the unit in MODE 3 and passively cooled within 36 hours.

Completion Times are established considering the likelihood of a LOCA event that would require ECCS actuation. They also provide adequate time to reach the required unit condition from full power conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification that the RVVs and RRVs are OPERABLE by stroking the valves open ensures that each train of ECCS will function as designed when these valves are actuated. One RVV is designed to be actuated by either division of the MPS and it must be verified to open from each division without dependence on the other. The RVVs and RRVs safety function is to open as described in the safety analysis. A Note is provided indicating that the SR is not required to be met for a valve that is open. This Note is necessary to allow a valve to be credited with performing its safety function when it may not be able to satisfy the SR requirements. When an ECCS valve is open it has performed its safety function.

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

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Actions cannot be completed within the associated Completion Time, or if both trains of DHRS are declared inoperable the unit~~plant~~ must be placed in a mode that does not rely on the DHRS. ~~To accomplish this the plant must be in MODE 2 within 6 hours and PASSIVE COOLING must be established within 36 hours. This condition ensures decay heat is removed and transferred to the UHS.~~ This is accomplished by Required Actions B.1 and B.2.

Required Action B.1 places the unit in MODE 2 within 6 hours. Required Action B.2 places the unit in MODE 3 and passively cooled within 36 hours.

Completion Times are established considering the likelihood of an event that would require DHRS actuation. They also provide adequate time to reach the required unit condition from full power conditions in an orderly manner.

BASES

ACTIONS (continued)

the unit~~plant~~ must be brought to MODE 2 within 6 hours and MODE 3 within 36 hours. The allowed Completion Times are reasonable, based on operating ~~experience~~requirements, to reach the required unit~~plant~~ conditions from full power conditions in an orderly manner~~ and without challenging plant systems~~.

E.1, E.2, E.3, E.4, and E.5

If the UHS bulk average boron concentration is not within limits, actions must be initiated and continued to restore the concentration immediately. Additionally, activities that could place pool inventory in communication with the reactor core must be suspended. Therefore, CFDS flow into the containment must be immediately terminated, and disassembly of the containment vessel that would open the RCS to communication with the UHS also suspended. Additionally, module~~MODULE~~ movement must be suspended and the movement of irradiated fuel suspended.

The suspension of module and/or fuel movement shall not preclude completion of movement to safe position.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

Verification that the UHS level is above the required minimum level will ensure that the assumed heat capacity of the pool is available and the pool will provide the credited mitigation if an irradiated fuel handling accident occurs. Indication of UHS level including alarms when not within limits are available in the main control room.

The Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.5.3.2

Verification that the UHS bulk average temperature is within limits ensures that the safety analyses assumptions and margins provided by the UHS remain valid. Key UHS temperatures are monitored and alarmed in the control room.

The Frequency is controlled under the Surveillance Frequency Control Program.

BASES

LCO (continued)

Compliance with this LCO will ensure a containment configuration, including maintenance access manways, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

APPLICABILITY

In MODES 1, 2, and 3 with RCS hot temperature ~~hot~~ ≥ 200ºF, the RCS contains sufficient energy such that DBA could cause a release of radioactive material into containment. The containment limits the postulated release of radioactive fission products that could be released from the containment from the reactor core and reactor vessel. The containment supports the emergency core cooling system (ECCS) by providing a part of the means of passive heat transfer from the reactor core, coolant, and vessel to the reactor cooling pool. ECCS OPERABILITY is required as described in LCO 3.5.1, “Emergency Core Cooling.”

In MODE 3 with the RCS hot temperature ~~hot~~ < 200º F, MODES 4 and 5, the probability and consequences of these events are reduced due to unit~~plant~~ conditions in these MODES. Therefore, containment is not required to be OPERABLE in these MODES.

ACTIONS

A.1

In the event containment is inoperable, it must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment OPERABLE during MODES 1, 2, and 3 with the RCS hot temperature ~~hot~~ ≥ 200ºF. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the unit~~plant~~ must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours and to MODE 3 with RCS hot temperature ~~hot~~ < 200ºF within 48 hours (Ref. 3). The allowed Completion Times are reasonable, to reach the required unit~~plant~~ conditions from full power conditions in an orderly manner ~~and without challenging plant systems~~.

BASES

ACTIONS (continued)

B.1

Condition B has been modified by a note indicating that this Condition is only applicable to those penetration flow paths with two condition isolation valves.

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation device that cannot be adversely affected by a single active failure. Isolation devices that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, or a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with

Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the devices are operated under administrative controls and the probability of the misalignment is low.

C.1 and C.2

If the Required Actions and associated Completion Times are not met, the unit~~plant~~ must be brought to a MODE or condition in which the LCO does not apply. To achieve this status, the unit~~plant~~ must be brought to at least MODE 2 within 6 hours and MODE 3 with RCS hot temperature~~ hot~~ < 200ºF within 48 hours.

Completion Times are established considering the likelihood of an event that would require CIS actuation. They also provide adequate time to reach the required unit condition from full power conditions in an orderly manner.

BASES

ACTIONS (continued)

D.1 and D.2

With Required Actions and associated Completion Times not met, isolation capability of the main steam line(s) is not maintained. The associated DHRS and the ability to isolate postulated releases from the SGs are affected. The unit~~MODULE~~ must be placed in a condition in which the LCO does not apply using Required Action D.1 and D.2.

Required Action D.1 requires the unit~~MODULE~~ must be placed in MODE 2 within 6 hours.

Required Action D.2 requires the unit to be in MODE 3 and PASSIVELY COOLED within 36 hours.

The Completion Times are reasonable based operating activities required to reach these conditions in an orderly manner~~, without challenging plant systems~~. The time permits use of normal means to exit the conditions of Applicability. It is also consistent with the Completion Times for an inoperable train of the DHRS.

SURVEILLANCE REQUREMENTS

SR 3.7.1.1

~~This SR verifies MSIV and MSIV Bypass Valve closure times are within limits on an actual or simulated actuation signal. The isolation time is assumed in the accident analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs and MSIV Bypass Valves are not tested at power since even a partial stroke exercise increases the risk of a valve closure when the unit is generating power. Because the isolation valves are not tested at power, they are exempt from the ASME OM Code (Ref. 6) requirements during operation in MODE 1.~~

~~The Frequency is in accordance with the INSERVICE TESTING PROGRAM.~~

~~This test is conducted with the unit in MODE 5. The valves cannot be fully stroked during plant operation because closing a secondary MSIV causes SG pressure and level transients and, most likely, a turbine trip and reactor trip.~~
This SR verifies the safety related and non-safety related MSIV and MSIV Bypass Valve closure times are within limits on an actual or simulated actuation signal. The isolation time is assumed in the accident and containment analyses. The MSIVs and MSIV Bypass Valves are not

BASES

ACTIONS (continued)

and deactivated automatic valve, closed manual valve, or blind flange. An inoperable FWIV/FWRV may be utilized to isolate the line only if its leak tightness has not been compromised. This action returns the system to a condition in which at least one valve in the affected flow path is performing the required safety function. The 8 hour Completion Time is a reasonable amount of time to complete the actions required to close the FWIV, or FWRV, which includes performing a controlled unit~~plant~~ shutdown without challenging plant systems.

D.1, and D.2

If the FWIVs and FWRVs cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit~~plant~~ must be placed in at least MODE 2 within 6 hours, in MODE 3 and PASSIVELY COOLED within 36 hours. The allowed Completion Times are reasonable, to reach the required unit conditions from full power conditions in an orderly manner~~ and without challenging plant systems~~.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the closure time of each FWIV and FWRV is within limits, on an actual or simulated actuation signal. The FWIV and FWRV isolation times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves are tested when the unit~~plant~~ is in a shutdown condition, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. Because the isolation valves are not tested when the unit~~plant~~ is in a shutdown condition, they are exempt from ASME OM Code (Ref. 5) requirements during operation in MODE 1.

The Frequency is in accordance with the INSERVICE TESTING PROGRAM.

NuScale Power

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

eRAI No.: 9034
Date of RAI Issue: 09/14/2017

NRC Question No.: 16-33

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 model standard technical specifications (STS) in the following documents provide NRC guidance on format and content of TS as acceptable means to meet 10 CFR 50.36 requirements. These documents may be accessed using the Agencywide Documents Access and Management Systems (ADAMS) by their accession numbers.

- NUREG-1431, “STS Westinghouse Plants,” Revision 4 (ADAMS Accession Nos. ML12100A222 and ML12100A228)
- NUREG-1432, “STS Combustion Engineering Plants,” Revision 4 (ADAMS Accession Nos. ML12102A165 and ML12102A169)
- NUREG-2194, “STS Westinghouse Advanced Passive 1000 (AP1000) Plants,” Revision 0 (ADAMS Accession No. ML16111A132)

The NRC staff needs to evaluate technical differences in the proposed generic TS (GTS) from applicable provisions in these documents, which are referenced by the DC applicant in Design Control Document (DCD) Tier 2, Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the GTS to ensure adequate protection of public health and safety, and the completeness and accuracy of the GTS Bases.

As proposed, some provisions of Subsection 3.1.3 are not clear.

a. The applicant is requested to reformat the Applicabilty statement of Subsection 3.1.3 so that it has the following presentation:

APPLICABILITY: MODE 1 for upper MTC limit,
MODES 1 and 2 for lower MTC limit,
MODE 3 with reactor coolant temperature ≥ 200°F for
lower MTC limit.

b. For the MODE 3 applicability statement, the applicant is requested to state which of the following reactor coolant temperatures is meant:

- Reactor coolant system (RCS) coolant temperature cold (Core inlet)
- RCS coolant temperature hot (Core outlet), or
- RCS coolant temperature average.

Also state whether

- All RCS temperature sensors must indicate ≥ 200°F, or
- One or more RCS temperature sensors must indicate ≥ 200°F.

c. The applicant is also requested to revise (1) the Applicability statement to clarify the intended meaning, and (2) the Applicability section of Subsection B 3.1.3 as needed to conform the Bases discussion to the revised MODE 3 applicability statement.

d. The applicant is further requested to (1) revise Required Action B.1 ("Be in MODE 3 with reactor coolant temperature < 200°F.") to be consistent with exiting the Applicability (as clarified) of LCO 3.1.3; (2) clarify whether all RCS coolant temperature sensors must indicate < 200°F, or just certain ones, to complete the action; and (3) revise the Actions section of Subsection B 3.1.3 as needed to conform the Bases discussion to the revised Applicability and Action statements.

NuScale Response:

The Applicability of 3.1.3 has been revised to better align with the Technical Specifications Writer's Guide by separating the MODE 3 applicability to a distinct line. Additionally, the temperature restriction has been clarified to 'any' RCS temperature greater than or equal to 200 degrees Fahrenheit. Corresponding changes have been incorporated into balance of the LCO and Bases.

Impact on DCA:

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

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 MTC shall be within limits specified in the COLR.

APPLICABILITY: MODE 1 for upper MTC limit,
MODES 1 ~~, 2, and MODE 3 with reactor coolant temperature ≥ 200°F for lower MTC limit~~and 2 for lower MTC limit,
MODE 3 with any RCS temperature ≥ 200°F for lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. MTC not within limits.	A.1	Be in MODE 2.	6 hours
B. MTC not within lower limit.	B.1	Be in MODE 3 with ~~reactor coolant temperature~~all RCS temperatures < 200°F.	48 hours

BASES

APPLICABLE SAFETY ANALYSES (continued)

one, which is assumed withdrawn. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations assuming steady state conditions at core beginning of cycle (BOC) and EOC. A measurement is conducted two-thirds of the core operating cycle; when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload design predictions.

MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

LCO 3.1.3 requires the MTC to be within specified limits of the COLR to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The limit on a least negative MTC ensures that core overheating accidents will not violate the accident analysis assumptions. The most negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be easily controlled once the core design is fixed. During operation, therefore, the LCO can only be ensured through measurement. The surveillance checks of MTC at BOC and near two-thirds of core burnup provide confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the upper limit on the MTC must be maintained to ensure that any accident will not violate the design assumptions of the accident analysis. The limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled CRA withdrawal, will not violate the assumptions of the accident analysis.

The lower MTC limit must be maintained in MODES 1 and 2 and MODE 3 with ~~reactor coolant~~any RCS temperature ≥ 200~~-~~°F,~~ 1,~~ to ensure that cooldown accidents will not violate the assumptions of the accident analysis.

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

eRAI No.: 9034
Date of RAI Issue: 09/14/2017

NRC Question No.: 16-34

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 model standard technical specifications (STS) in the following documents provide NRC guidance on format and content of TS as acceptable means to meet 10 CFR 50.36 requirements. These documents may be accessed using the Agencywide Documents Access and Management Systems (ADAMS) by their accession numbers.

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- NUREG-2194, "STS Westinghouse Advanced Passive 1000 (AP1000) Plants," Revision 0
  (ADAMS Accession No. ML16111A132)

The NRC staff needs to evaluate technical differences in the proposed generic TS (GTS) from applicable provisions in these documents, which are referenced by the DC applicant in Design Control Document (DCD) Tier 2, Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the GTS to ensure adequate protection of public health and safety, and the completeness and accuracy of the GTS Bases.

SR 3.1.4.1 requires verification of individual control rod assembly (CRA) alignment. The Frequency is modified by a surveillance column Note. The Surveillance and its Note, state (with a staff suggested grammatical enhancement added):

SR 3.1.4.1	------------------------------NOTE------------------------------ Not required to be performed for rods associated with an inoperable rod position indicator. -- Verify position of individual CRAs within alignment limit.

The third paragraph of the Bases for SR 3.1.4.1 states,

> The SR is modified by a Note that permits it not to be performed for rods associated with an inoperable rod position indicator. The alignment limit is based on rod position indicator which is not available if the indicator is inoperable. LCO 3.1.7, "Rod Position Indication," provides Actions to verify the rods are in alignment when one or more rod position indicators are inoperable.

a. The applicant is requested to describe whether other means of determining rod position are available to perform SR 3.1.4.1 (such as provided in the Actions of LCO 3.1.7). If other means are available, discuss whether they should be allowed for determining rod alignment. If they are allowed, then the staff suggests removing the Note, to preclude an interpretation that the Note would allow unit operation with the rod alignment undetermined.

b. The staff notes that the second and third paragraphs of the Applicable Safety Analyses section of Subsection B 3.1.4 need correction for consistency with the Chapter 2.0 SLs, and editing for improved clarity, as follows:

> Accident and transient ~~analysis~~ analyses associated with CRA misalignment, static and dynamic, ~~are analyzed~~ accounting for misalignment of 6 steps at the initiation of the event ~~of 6 steps~~. The results of the CRA misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, ~~fuel centerline temperature, or RCS~~ or the SLs on CHFR, peak Linear Heat Rate, and pressurizer pressure occur.
>
> CRA alignment limits and OPERABILITY requirements satisfy ~~Criteria~~ Criterion 2 of 10 CFR 50.36(c)(2)(ii).

The applicant is requested to make the indicated changes after verifying their technical accuracy.

NuScale Response:

The grammatical change has been incorporated into the Note applicable to SR 3.1.4.1.

As described in the Bases of 3.1.4.1, the means of determining rod position to verify compliance with the limits is by means of the rod position indication requirements specified in LCO 3.1.7. Therefore the means described in LCO 3.1.7 are adequate to verify that rod alignment is within limits. A description of the control rod drive system is provided in FSAR Section 4.6.

Additional grammatical changes to the Bases have been incorporated. Note that the proposed SL have been modified and the Bases description aligns with those limits.

Impact on DCA:

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	-----------------------------NOTE------------------------------ Not required to be performed for rods associated with an inoperable rod position indicator. --- Verify position of individual CRAs within alignment limit.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 4 steps in either direction.	In accordance with the Surveillance Frequency Control Program
SR 3.1.4.3	Verify each CRA drop time ≤ 2.2 seconds.	Prior to reactor criticality after each removal of the upper reactor pressure vessel section

BASES

APPLICABLE SAFETY ANALYSES

CRA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CRA misoperation as any event with the single failure of a safety-related component and multiple failures of non-safety related controls. The acceptance criteria for addressing CRA inoperability or misalignment are that:

a. With the most reactive CRA stuck out of the core there will be no violations of either:

   1. Specified acceptable fuel design limits (SAFDLs); or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after design basis events with all CRAs fully inserted.

Accident and transient analys~~i~~es associated with CRA misalignment, static and dynamic, ~~are analyzed~~ accounti~~ng~~ for misalignment of 6 steps at the initiation of the event ~~of 6 steps~~. The results of the CRA misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, ~~fuel centerline temperature, or RCS~~or the SLs on critical heat flux ratio, fuel centerline temperature, or pressurizer pressure occur.

CRA alignment limits and OPERABILITY requirements satisfy Criterion~~a~~ 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown and regulating CRA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on CRA OPERABILITY ensure that upon reactor trip, the CRAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The CRA OPERABILITY requirements (i.e., trippability) are separate from alignment requirements which ensure that the CRA groups maintain the correct power distribution and CRA alignment. The CRA OPERABILITY requirement is satisfied provided the CRA will fully insert in the required CRA drop time assumed in the safety analysis. CRA control malfunctions that result in the inability to move a CRA (e.g., CRA rod lift coil failures), but do not impact trippability, do not result in CRA inoperability.

The requirement is to maintain the CRA alignment to within 6 steps between any CRA and its group position. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors, or unacceptable SDMs, both of which may constitute initial

NUSCALE POWER™

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

eRAI No.: 9034
Date of RAI Issue: 09/14/2017

NRC Question No.: 16-35

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 model standard technical specifications (STS) in the following documents provide NRC guidance on format and content of TS as acceptable means to meet 10 CFR 50.36 requirements. These documents may be accessed using the Agencywide Documents Access and Management Systems (ADAMS) by their accession numbers.

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The NRC staff needs to evaluate technical differences in the proposed generic TS (GTS) from applicable provisions in these documents, which are referenced by the DC applicant in Design Control Document (DCD) Tier 2, Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the GTS to ensure adequate protection of public health and safety, and the completeness and accuracy of the GTS Bases.

The applicant is requested to consider revising the following Bases Subsections by inserting “(SAFDLs)” after the first use of the phrase “specified acceptable fuel design limits.” Subsequent uses of the phrase within the Subsection should be replaced by “SAFDLs.” Note that Subsection B 3.1.1 uses the phrase twice, and Subsection B 3.1.4 uses ‘SAFDLs” without having defined it on first use.

B 2.1.1, Background section,

- Paragraph 1, Sentence 1

B 3.1.1, Applicable Safety Analyses (ASA) section,

- Paragraph 1, Sentence 2

- Paragraph 2

B 3.1.4, ASA section,

- Paragraph 1, list item a.1

- Paragraph 2, Sentence 2

B 3.1.5, ASA section, Paragraph 2, list item a.1

B 3.3.1, Background section, Paragraph 1, Sentence 1

NuScale Response:

NuScale has inserted the requested "(SAFDLs)" after the first use of the phrase "specified acceptable fuel design limits" in the Bases locations where it is used more than once. These are the Bases for 3.1.1, and 3.1.4. After the initial use and definition of the acronym in the respective Bases now use the acronym "SAFDLs"

Other locations were not modified when the phrase only occurs one time in the respective Bases and the acronym is not otherwise utilized.

Impact on DCA:

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

According to GDC 26 (Ref. 1) the reactivity control systems must be redundant and capable of holding the reactor core subcritical when shutdown under cold conditions. Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel.

SDM requirements provide sufficient reactivity margin to assure that specified acceptable fuel design limits (SAFDLs) will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion or scram of all shutdown and regulating group control rod assemblies (CRAs), assuming that the single CRA of highest reactivity worth is fully withdrawn.

Additionally SDM requirements provide sufficient reactivity margin to ensure that the reactor will remain shutdown at all temperatures with all control rods inserted.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of movable CRAs and soluble boric acid in the Reactor Coolant System (RCS). The CRA System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, following all AOOs~~S~~ and postulated accidents, assuming that the CRA of highest reactivity worth remains withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes and maintain the reactor subcritical under cold conditions.

During power operation, SDM control is ensured by operating with the shutdown group fully withdrawn and the regulating group within the limits of LCO 3.1.6, “Regulating Group Insertion Limits.”

When the unit~~MODULE~~ is in MODES 2, 3, 4 or 5, the SDM requirements are met by means of adjustments to the RCS boron concentration and the boron requirements for the pool, LCO 3.5.3, "Ultimate Heat Sink" and CRA controls.

BASES

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analyses. The safety analyses (Ref. 2) establish a SDM that ensures that ~~specified acceptable fuel design limits~~SAFDLs are not exceeded for normal operation and AOOs, with the assumption of the highest worth CRA stuck out on scram. For MODES 2 and 3, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that ~~specified acceptable fuel design limits~~SAFDLs are maintained. This is done by ensuring that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SDM requirement also protects against:

a. Inadvertent boron dilution;

b. An uncontrolled CRA withdrawal from subcritical or low power condition; and

c. CRA ejection.

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the RCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life, when critical boron concentrations are highest.

~~The uncontrolled CRA withdrawal transient is terminated by a high power, high power rate, high hot temperature, and high pressurizer pressure trips. Power level, RCS pressure, linear heat rate, and the CHFR do not exceed allowable limits.~~

BASES

APPLICABLE SAFETY ANALYSES

CRA misalignment accidents are analyzed in the safety analysis (Ref. 3). The accident analysis defines CRA misoperation as any event with the single failure of a safety-related component and multiple failures of non-safety related controls. The acceptance criteria for addressing CRA inoperability or misalignment are that:

a. With the most reactive CRA stuck out of the core there will be no violations of either:

   1. Specified acceptable fuel design limits (SAFDLs); or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after design basis events with all CRAs fully inserted.

Accident and transient analys~~i~~es associated with CRA misalignment, static and dynamic, ~~are analyzed~~ accounti~~ng~~ for misalignment of 6 steps at the initiation of the event ~~of 6 steps~~. The results of the CRA misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, ~~fuel centerline temperature, or RCS~~or the SLs on critical heat flux ratio, fuel centerline temperature, or pressurizer pressure occur.

CRA alignment limits and OPERABILITY requirements satisfy Criterion~~a~~ 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The limits on shutdown and regulating CRA alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on CRA OPERABILITY ensure that upon reactor trip, the CRAs will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The CRA OPERABILITY requirements (i.e., trippability) are separate from alignment requirements which ensure that the CRA groups maintain the correct power distribution and CRA alignment. The CRA OPERABILITY requirement is satisfied provided the CRA will fully insert in the required CRA drop time assumed in the safety analysis. CRA control malfunctions that result in the inability to move a CRA (e.g., CRA rod lift coil failures), but do not impact trippability, do not result in CRA inoperability.

The requirement is to maintain the CRA alignment to within 6 steps between any CRA and its group position. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors, or unacceptable SDMs, both of which may constitute initial

Response to Request for Additional Information
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NRC Question No.: 16-36

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 model standard technical specifications (STS) in the following documents provide NRC guidance on format and content of TS as acceptable means to meet 10 CFR 50.36 requirements. These documents may be accessed using the Agencywide Documents Access and Management Systems (ADAMS) by their accession numbers.

- NUREG-1431, "STS Westinghouse Plants," Revision 4
  (ADAMS Accession Nos. ML12100A222 and ML12100A228)

- NUREG-1432, "STS Combustion Engineering Plants," Revision 4
  (ADAMS Accession Nos. ML12102A165 and ML12102A169)

- NUREG-2194, "STS Westinghouse Advanced Passive 1000 (AP1000) Plants," Revision 0
  (ADAMS Accession No. ML16111A132)

The NRC staff needs to evaluate technical differences in the proposed generic TS (GTS) from applicable provisions in these documents, which are referenced by the DC applicant in Design Control Document (DCD) Tier 2, Section 16.1, and the docketed rationale for each difference because conformance to STS provisions is used in the safety review as the initial point of guidance for evaluating the adequacy of the GTS to ensure adequate protection of public health and safety, and the completeness and accuracy of the GTS Bases.

The AP1000 STS Subsection 3.1.8, "PHYSICS TESTS Exceptions," specifies a CHANNEL OPERATIONAL TEST (COT) in SR 3.1.8.1 for the power range neutron flux and intermediate range neutron flux channels per SR 3.3.1.6, SR 3.3.1.7, and SR 3.3.3.2, with a Frequency of "Once prior to initiation of PHYSICS TESTS." The applicant is requested to justify the omission of an equivalent COT in GTS Subsection 3.1.8.

NuScale Response:

The NuScale instrumentation technical specifications in Section 3.3, Instrumentation, specify the OPERABILITY requirements applicable to the nuclear instrumentation and actuation functions in Tables 3.3.1-1, and 3.3.3-1, and 3.3.4-1, and in LCO 3.3.2. Table 3.3.1-1 specifies the MODE applicability for the various nuclear instrumentation functions. LCO 3.3.2, and LCO 3.3.3 specify the MODES of applicability requirements for the automatic actuations that result from the nuclear instrumentation. The requirements in technical specifications section 3.3 assure the OPERABILITY of instruments, and assure that required surveillance testing is completed in accordance with SR 3.0 Surveillance Requirements Applicability. Since the applicability in NuScale section 3.3 for nuclear instrumentation corresponds with the applicability of NuScale 3.1.8, PHYSICS TESTS Exceptions, surveillance testing is already specified and an additional surveillance testing requirement is not needed.

Impact on DCA:

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