NuScaleDCRaisPEm Resource

From: Cranston, Gregory
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To: Request for Additional Information
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Subject: Request for Additional Information No. 520 eRAI No. 9642 (16)
Attachments: Request for Additional Information No. 520 (eRAI No. 9642).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Design Certification Application.

Please submit your technically correct and complete response by May 15, 2019, to the RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

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Request for Additional Information No. 520 (eRAI No. 9642)

Issue Date: 03/20/2019
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 16 - Technical Specifications
Application Section: Part 2, FSAR Section 16; Part 4, GTS and Bases

QUESTIONS

16-61

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

Generic TS Subsection 3.4.3 Action D states:

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

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

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

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

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

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

> CNV level instrumentation. The CNV is flooded approximately to the elevation of the RPV pressurizer baffle plate. Automatic action shuts off **the operating CFDS pumps** and closes the NuScale Power Module CES isolation valve when the preset water level in the CNV is reached, as determined by CNV level instrumentation. At completion of flooding operation, the CFDS containment isolation valves and the CFDS module isolation valve for the NPM being flooded are closed.

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

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

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

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

1. Regarding the above quotation from FSAR Section 9.3.6.2.2:

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

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

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

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

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

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

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

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

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

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

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

16-62

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

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

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

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

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

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

...The SFM logic unit performs calculations to determine the rate of change and compares the result to a setpoint. The trip provides protection against core damage and protects the reactor coolant pressure boundary (RCPB) during the following events:

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

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

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

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

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

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

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

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

16-63

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

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

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

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

MODE 2, ~~and~~ MODE 2 to MODE 3 and not PASSIVELY COOLED, and not PASSIVELY COOLED to PASSIVELY COOLED.

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

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

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

16-64

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

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

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

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

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

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

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

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

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

Background Discussion:

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

- W-STS Subsection 3.6.11, "Iodine Cleanup System (ICS)"
  - SR 3.6.11.3 *[-----------------------------NOTE-----------------------------*

    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

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

- W-STS Subsection 3.7.10, "Control Room Emergency Filtration System (CREFS)"
  - SR 3.7.10.3 *[-----------------------------NOTE-----------------------------*

    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

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

- W-STS Subsection 3.7.12, "ECCS Penetration Room Exhaust Air Cleanup System (PREACS)"
  - SR 3.7.12.3 *[-----------------------------NOTE-----------------------------*

    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

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

- W-STS Subsection 3.7.13, "Fuel Building Air Cleanup System (FBACS)"
  - SR 3.7.13.3 *[-----------------------------NOTE-----------------------------*

    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

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

- W-STS Subsection 3.7.14, "Penetration Room Exhaust Air Cleanup System (PREACS)"
  - SR 3.7.14.3 *[-----------------------------NOTE-----------------------------*

    Not required to be met for dampers and valves locked, sealed or otherwise secured in the actuated position.
    ---]

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

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

- CE-STS Subsection 3.7.10, “Essential Chilled Water (ECW)”
  - SR 3.7.10.2 *[-----------------------------NOTE------------------------------*

    Not required to be met for valves locked, sealed or otherwise secured in the actuated position.
    ---]

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

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

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

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

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

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

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

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

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

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

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

In Revision 0 of DCA Part 4, the Bases for SR 3.6.2.3 contain the first sentence of the passage quoted below. This statement also appears similar to the intent of the traveler’s bracketed surveillance column

Note (emphasis added). In Revisions 1 and 2 of DCA Part 4, the Bases for SR 3.6.2.4 (as renumbered beginning in DCA Revision 1) also includes the second sentence of this passage, regarding the added exception, that states the following:

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

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

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

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

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

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

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

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

> NuScale has adopted and incorporated the conceptual basis of TSTF traveler 541 into the proposed GTS *independent of the outcome of the TSTF-NRC traveler review and approval activities*.
>
> The conceptual basis and underlying issue is that as described in Section SR 3.0, if a surveillance requirement cannot be met, then the associated LCO is not being met and the associated Condition must be entered. However in some circumstances *the*

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

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

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

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

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

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

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

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

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

Although not directly applicable, the Intent of the traveler was adopted in the NuScale GTS. NuScale safety-related reactor trip system and ECCS components are credited with a single safety-related position, each of which is achieved by the component being de-energized.

The implementation of this traveler is under additional review and consideration as requested by the NRC staff at the time this technical report was developed. See RAI [197-9051, Question] 16-28.

In RCDR Revision 1, Table C-1 listed the affected GTS Specifications influenced by TSTF-541, draft Revision 1. These Specifications, along with the affected SRs, are listed below. The staff notes that Specification 3.3.1, "MPS Instrumentation," should also be included in the list because the supplemental response (ML18355Annn) to RAI 196-9050, Question 16-17, revised Subsection 3.3.1 of Revision 2 of DCA Part 4 so that SR 3.3.1.5 specifies performing a Channel Calibration of Class 1E isolation devices for MPS instrumentation components. The supplemental response to Question 16-17 similarly revised SR 3.3.2.3 and SR 3.3.3.3; the revised SRs are quoted below.

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

3.1.9, Boron Dilution Control

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

3.3.1, MPS Instrumentation

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

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

3.3.2, Reactor Trip System Logic and Actuation

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

Perform ACTUATION LOGIC TEST. | IAW SFCP

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

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

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

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

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

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

3.3.3, Engineered Safety Feature Actuation System Logic and Actuation

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

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

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

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

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

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

3.4.6, Chemical and Volume Control System Isolation Valves

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

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

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

3.4.10, Low Temperature Overpressure Protection Valves

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

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

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

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

3.5.1, Emergency Core Cooling System

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

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

SR 3.5.1.2 -----------------------------NOTE----------------------------------
Not required to be met for valves that are open.

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

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

3.5.2, Decay Heat Removal System

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

3.6.2, Containment Isolation Valves

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

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

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

3.7.1, Main Steam Isolation Valves

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

3.7.2, Feedwater Isolation

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

16-65

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

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

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

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

The Bases for SR 3.3.1.3 state:

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

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

Observations on SR 3.3.1.3 and associated Bases:

1. The word "required" is unnecessary in the Surveillance statement.
2. In the Bases phrase, "channel actuation response time," the word "actuation" is unnecessary and inconsistent with SR 3.3.1.3, which uses the phrase "channel response time."
3. The phrase "accident analysis" is used in the Bases for SR 3.3.1.3, but the phrase "safety analysis" is used in the corresponding similar sentences in the Bases for SR 3.3.2.2 and SR 3.3.3.2. This appears to be inconsistent.
4. In the Bases, consider modifying the reference "FSAR Chapter 7" to say "FSAR Section 7.2 (Ref. 1)."
5. The "channel response time" verified by SR 3.3.1.3 appears to span the channel's process sensor to the channel's output from the analog to digital converter, and excludes the comparison of the digital signal with the channel trip setpoint in the SFM. SER Section 7.2 gives the staff's evaluation of the "digital response time" verification testing.
6. When "channel response time" is meant, the Bases should use the full phrase for clarity, not just "response time," which is more general. Consider discussing the overlapping component response times in an MPS instrument channel (e.g., "sensor response time" is already called out).
7. Regarding allocated MPS instrument channel component response times, the last sentence of the definitions of the W-AP1000-STS defined terms RTS Response Time and ESF Response Time states:

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

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

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

The Bases for SR 3.3.2.2 state:

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

... The maximum digital time response is described in the FSAR. This SR encompasses the response time of the RTS division from the output of the equipment interface modules until the RTBs are open. ...

Observations on SR 3.3.2.2 and associated Bases:

1. The word "required" is unnecessary in the Surveillance statement.
2. The phrase "accident analysis" is used in the Bases for SR 3.3.1.3, but the phrase "safety analysis" is used in the corresponding similar sentences in the Bases for SR 3.3.2.2 and SR 3.3.3.2. This appears to be inconsistent.
3. In the Bases, consider modifying the reference to "FSAR" to say "FSAR Section 7.2 (Ref. 1)."
4. Consider whether it would be more accurate to say "total division measurements" in place of "total channel measurements."
5. The "RTS division response time" verified by SR 3.3.2.2, appears to span the analog output of the RTS EIM to the division's two RTBs, and excludes verification of the "digital time response," which appears to span the components from receipt of the digital process signal, to the setpoint comparison in the SFM, through the SVM, and through the priority logic of the RTS EIM. SER Section 7.2 gives the staff's evaluation of the "digital response time" verification testing.
6. Consider discussing in the Bases the overlapping digital component response times in an RTS division and how "maximum digital time response" is verified.

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

The Bases for SR 3.3.3.2 state:

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

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

Observations on SR 3.3.3.2 and associated Bases:

1. The word "required" is unnecessary in the Surveillance statement.
2. The phrase "accident analysis" is used in the Bases for SR 3.3.1.3, but the phrase "safety analysis" is used in the corresponding similar sentences in the Bases for SR 3.3.2.2 and SR 3.3.3.2. This appears to be inconsistent.
3. In the Bases, consider modifying the reference to "FSAR" to say "FSAR Section 7.2 (Ref. 1)."
4. Consider whether it would be more accurate to say "total division measurements" in place of "total channel measurements."
5. The "ESFAS division response time" verified by SR 3.3.3.2, appears to span the analog output of the pressurizer heater breaker EIM to the division's two pressurizer heater breakers, and excludes verification of the "digital time response," which appears to span the components from receipt of the digital process signal, to the setpoint comparison in the SFM, through the SVM, and through the priority logic of the pressurizer heater breaker EIM. Also excluded is the digital portion of the ESFAS division for the other ESF Logic and Actuation functions. See SER Section 7.2 for the staff's evaluation of the "digital response time" verification testing.
6. Consider discussing in the Bases the overlapping digital component response times in an ESFAS division and how "maximum digital time response" is verified.
7. Consider clarifying in the Bases for the following SRs for Inservice Testing Program ESFAS valve actuations (The Frequency of "In accordance with the Inservice Testing Program" is taken to mean 24 months for these SRs.) that the valve "isolation (or 'closure') time" or "open actuation time" (the time to stroke closed or stroke open, respectively) is included in the ESF Function's overall response time. Also, the word "required" is not needed because it is redundant to "within limits.":

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

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

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

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

BACKGROUND DISCUSSION

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME

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

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

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

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

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

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

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

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

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

16-66

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.

This question is a followup of RAI 472-9445 (ML18130A984), Question 16-43, to which the applicant has responded ((ML18163A417).

NuScale’s methodology to set the AXIAL OFFSET (AO) (LCO 3.2.2) and Power Dependent Insertion Limits (PDILs) (LCO 3.1.6) is dependent upon TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model” and TR-0716-50350, “Rod Ejection Accident Methodology,” in additional to the other methodologies listed. Also, CHF is used as acceptance criteria in LOCA and rod-ejection analyses. Accordingly, the staff believes the following changes need to be made to paragraph b of GTS 5.6.3:

The applicant is requested to update References 1, 3, and 4 to include TR-0516-49422, “Loss-of-Coolant Accident Evaluation,” and TR-0716-50350, “Rod Ejection Accident Methodology.”