RAIO-0518-60117

May 21, 2018

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. 395 (eRAI No. 9416) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 395 (eRAI No. 9416)," dated March 21, 2018

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. 9416:

- 15.02.06-1
- 15.02.06-2
- 15.02.06-3

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 Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9416

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9416

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

eRAI No.: 9416
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.06-1

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the general design criteria (GDC) in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A. GDC 15 requires that the reactor coolant system (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Design-Specific Review Standard (DSRS) Section 15.2.6, "Loss of Nonemergency AC Power to the Station Auxiliaries," provides guidance for meeting the requirements of several GDC, including GDC 15. In doing so, it guides the reviewer to evaluate the values of system parameters and initial core and system conditions as input to the model.

Some of the bias conditions listed in FSAR Tier 2, Table 15.2-18, "Input Parameters Loss of Non-Emergency AC Power - Limiting Cases," do not appear to be consistent with the limiting bias conditions in engineering calculation (EC)-0000-2908, Revision 1, "Loss of Non-Emergency AC Power to the Station Auxiliaries Analysis," which supports the FSAR Section 15.2.6 analysis. In particular, the initial RCS temperature and initial SG pressure for the RCS overpressure case are not consistent between the FSAR and the EC. Please confirm whether the bias conditions in FSAR Tier 2, Table 15.2-18 are accurate. If so, justify the difference relative to the EC. If not, update the FSAR as appropriate.

NuScale Response:

Table 15.2-18 has been corrected to show the low reactor coolant system (RCS) temperature bias and high steam pressure bias consistent with the loss of AC power limiting RCS pressure scenario.

Impact on DCA:

FSAR Table 15.2-18 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.02.06-1

Table 15.2-18: Input Parameters Loss of Non-Emergency AC Power -Limiting Cases

Parameter	RCS Overpressure	SG Overpressure	MCHFR
Initial RCS pressure	1920 psia	1920 psia	1780 psia
Initial RCS temperature	~~555~~535°F	555°F	555°F
Initial PZR level	68%	68%	68%
Initial Feedwater temperature	310°F	310°F	310°F
Initial SG Pressure	~~500~~535 psia	535 psia	500 psia
Drift on RSV setpoint	2137 psia (+3%)	2137 psia (+3%)	2137 psia (+3%)
Moderator and Doppler coefficients of reactivity	0.0/-1.40pcm/°F	0.0/-1.40pcm/°F	0.0/-1.40pcm/°F
RCS Flowrate	1179 lbm/s	1179 lbm/s	1179 lbm/s
Pool Temperature	200°F	200°F	200°F
SG Tube Heat Transfer	Nominal	+30%	+30%
DC Power Available	EDNS, EDSS	EDNS, EDSS	EDNS, EDSS

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eRAI No.: 9416
Date of RAI Issue: 03/21/2018

NRC Question No.: 15.02.06-2

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.6 provides guidance for meeting the requirements of GDC 10, 13, 15, and 26. To meet these requirements, DSRS Section 15.2.6 states that the most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

FSAR Section 15.2.6.2 states that no single failure caused more severe consequences for the RCS pressure and MCHFR cases but does not provide justification for the statement. Justify why no single failure is more limiting for RCS pressure and MCHFR, and update the FSAR as appropriate.

NuScale Response:

For the loss of AC power event, the failure of either a feedwater isolation valve (FWIV) or a main steam line isolation valve (MSIV) to close has no impact on reactor coolant system (RCS) pressure given the loss of secondary flow. This is confirmed by sensitivity cases documented in Table 1 which evaluate the failure of a FWIV or MSIV to close for the limiting peak RCS pressure case presented in FSAR Section 15.2.6.

Table 1. Single Failures for Peak RCS Pressure Case

Description	Peak RCS Pressure (psia)	Time of Peak RCS Pressure (s)	Time FWIV/MSIVs Begin to Close (s)
Limiting RCS Pressure Case	2162	10	9
w/ Failure of One FWIV	2162	10	9
w/ Failure of One MSIV	2162	10	9

Given the event sequence for the limiting minimum critical heat flux ratio (MCHFR) case, the

failure of either a FWIV or MSIV to close has no impact on MCHFR as the limiting value occurs while the valves remain open prior to secondary isolation per Figure 15.2-26 of the FSAR.

Section 15.2.6.2 of the FSAR is updated to clarify why neither the failure of a FWIV nor a failure MSIV to close causes more severe peak RCS pressure or a more limiting MCHFR for the loss of non-emergency AC power to station auxiliaries.

Impact on DCA:

FSAR Section 15.2.6.2 has been revised as described in the response above and as shown in the markup provided in this response.

A loss of AC power event is expected to occur one or more times in the life of the NPM, so it is classified as an AOO. The categorization of the NuScale design basis events are discussed in Section 15.0.0, Table 15.0-1.

15.2.6.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of a loss of AC power event assumes the plant control systems and MPS perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a loss of AC power event.

The loss of AC power event for the NuScale design assesses the loss of the low voltage AC power distribution system (ELVS), which supplies power to plant motors, heaters, packaged equipment and battery chargers. The severity of the loss of AC power event is dictated by the timing of the reactor trip and the loss of power. For all limiting cases, the transient is more severe if there is a loss of AC power with both the EDSS and EDNS DC power supplies still available. Loss of DC power initiates a reactor trip and ESF actuations earlier in the transient and results in less limiting conditions.

In the limiting RCS pressure case, power is lost to pressurizer heaters; condensate, feed, and CVCS pumps; and the turbine trips immediately. The loss of cooling in the secondary causes a heatup and increase in pressure in the RCS which initiates a reactor trip and actuates DHRS on the high pressurizer pressure MPS signal. Immediately following closure of the MSIVs and FWIVs, the peak RCS pressure occurs coincident with the lifting of an RSV, after which system pressure quickly decreases. Secondary pressure increases until the DHRS valves are fully open and natural circulation is established.

RAI 15.02.06-2

The limiting single failure for a loss of AC power is the failure of a FWIV to close, which results in the maximum steam generator pressure. ~~No single failures increased peak RCS pressure or reduced the MCHFR.~~No single failure of a FWIV or MSIV to close increased peak RCS pressure as secondary flow is lost at event initiation. No single failure reduced the MCHFR as the limiting value occurs prior to secondary isolation signal being generated.

Three event sequences for the loss of AC power event are provided in Table 15.2-15, Table 15.2-16, and Table 15.2-17 for the limiting scenarios considering biased boundary conditions.

15.2.6.3 Thermal Hydraulic and Subchannel Analyses

15.2.6.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a loss of AC power event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

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NRC Question No.: 15.02.06-3

The transient and accident analyses in FSAR Tier 2, Chapter 15 serve, in part, to demonstrate compliance with the GDC. DSRS Section 15.2.6 provides guidance for meeting the requirements of GDC 10, 13, 15, and 26. To meet these requirements, DSRS Section 15.2.6 states that the most limiting plant system single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems."

FSAR Tier 2, Section 15.2.6, "Loss of Non-emergency Power to the Station Auxiliaries," discusses that the failure of a feedwater isolation valve (FWIV) to close is limiting for steam generator (SG) pressure. However, FSAR Tier 2, Table 15.2-16, "Loss of Non-Emergency AC Power - SG Peak Pressure - Sequence of Events," does not appear to reflect the failure of a FWIV to close, as the table indicates that FWIVs are fully closed at 14 seconds. Therefore, it is unclear whether the results do, in fact, consider the most limiting plant system single failure. Please address this discrepancy, and update the FSAR as appropriate.

NuScale Response:

The results presented in FSAR Table 15.2-16 were unclear regarding the most limiting plant system single failure. This table has been revised to reflect the failure to close one of the feedwater isolation valves (FWIV) on feedwater (FW) line 1 and the time of the feedwater regulating valve (FWRV) to close.

Impact on DCA:

FSAR Table 15.2-16 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.02.06-3

Table 15.2-16: Loss of Non-Emergency AC Power -SG Peak Pressure - Sequence of Events

Event	Time [s]
Loss of AC power occurs	0
Turbine trip occurs	0
Feedwater pump trips.	0
CVCS pump trips (approximated as CVCS isolation)	0
High pressurizer pressure is reached (2000 psia).	5
RTS actuation on high pressurizer pressure signal.	7
DHRS actuation on the high pressurizer pressure signal. DHRS actuation valves begin to open.	7
FWIVs and MSIVs begin to close.	9
RSV1 opens	10
Peak RPV pressure is reached (2156 psia)	10
MSIVs are fully closed.	14
FWIV~~s~~ is~~are~~ fully closed (train 1 fails to close).	14
DHRS actuation valves are fully open.	37
FWRVs are fully closed.	39
Peak steam generator pressure is reached.	76