June 08, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 421 (eRAI No. 9462) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 421 (eRAI No. 9462)," dated April 12, 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. 9462:

- 04.04-3
- 04.04-4

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
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9462

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NuScale Response to NRC Request for Additional Information eRAI No. 9462

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

NuScale Power™

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

eRAI No.: 9462
Date of RAI Issue: 04/12/2018

NRC Question No.: 04.04-3

GDC 10, *Reactor design*, requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. FSAR, Tier 2, Section 4.4.2.9, states that uncertainties or biases are incorporated into the subchannel methodology to provide conservatism and that these uncertainties establish the design limit for the critical heat flux (CHF) correlation. FSAR, Tier 2, Section 4.4.2.9 further states that the derivation of these penalties or conservative biases are discussed in TR-0915-17564, "Subchannel Analyses Methodology." The methodology used to obtain the penalties and the methodology used to combine the penalties are described in Section 3.12 and Section 3.4 of TR-0915-17564, respectively. Section 1.1 of TR-0915-17564 states that the analysis results presented within the report are for demonstration of the analytical methodology and approval of the results are not sought as part of the report.

The penalties that are used to establish the design limits for the NSP2 and NSP4 CHF correlations supporting the NuScale Design Certification Application are not provided in FSAR Section 4.4. NRC staff needs to establish a finding that the penalties applied to the NSP2 and NSP4 CHF correlations provide suitably conservative safety limits for use in transient and accident analyses. Accordingly, NRC staff requests that NuScale update FSAR, Tier 2, Section 4.4.2.9.2 to provide the penalties and their bases used to set the CHF ratio limits of 1.262 and 1.284 for the NSP2 and NSP4 CHF correlations, respectively.

NuScale Response:

Section 3.12 of the Subchannel Analysis Methodology topical report (TR-0915-17564, Revision 1), provides the methodology for accounting for subchannel analysis uncertainties, including those from analysis method, physical manufacturing design inputs, and operating conditions. Section 4.4.2.9 of the FSAR also discusses this subject. There are two uncertainties that are applied as penalties to the MCHFR design limit for subchannel analyses. The values for the heat flux engineering uncertainty factor (F^E_Q) and the fuel rod and assembly bow uncertainty are provided in the topical report. The 95/95 MCHFR design limit is 1.17 for the NSP2 CHF correlation, which is conservatively increased to 1.19. When the F^E_Q and fuel rod and assembly bow penalties are applied, the CHF analysis limit becomes 1.262. In the case of the NSP4 correlation, the 95/95 MCHFR design limit is 1.21; application of the F^E_Q and the fuel rod and assembly bow penalties increase the CHF analysis limit to 1.284. This uncertainty application to the MCHFR design limit is depicted in Figure 4.4-1 of the FSAR.

Impact on DCA:

FSAR Section 4.4.2.9.2 been revised as described in the response above and as shown in the markup provided in this response.

where the hot channel MCHFR occurs. The CHF analysis limit used for thermal margin evaluations biases the 95/95 MCHFR design limit by the penalty for potential rod or assembly bowing. The penalty for rod bow is 3% as described in Reference 4.4-3. This penalty is a conservative value based on the AREVA methodology for rod bow in Reference 4.4-2, which was demonstrated to be applicable for the NuScale fuel design in Reference 4.4-7.

RAI 04.04-03

The F^E_Q and rod bow penalties are both applied to the MCHFR design limit in accordance with the methodology in Reference 4.4-3. As shown in Figure 4.4-1, the 95/95 MCHFR design limit for the NSP4 CHF correlation is 1.21. The CHF analysis limit becomes 1.284 with the F^E_Q and rod bow penalties applied.

As discussed in Reference 4.4-3, the RCS pressure bias is not consistent across all conditions. As a result, case dependent bias directions on pressure are utilized to ensure a conservative calculation of MCHFR.

Core Inlet Flow Distribution Uncertainty

The core inlet flow distribution is discussed in Section 4.4.2.5. For the subchannel analysis methodology, inlet flow distribution uncertainty is applied to the hot or limiting assembly as shown in the distributions presented in Reference 4.4-3. The open lattice of the NuScale core allows flow redistribution to occur for inlet flow imbalances and the 5 percent reduction to the hot assembly assumed has a minimal effect on MCHFR.

Core Exit Pressure Distribution Uncertainty

The open upper plenum design allows for pressure equilibrium and no core exit pressure distribution uncertainty is necessary for the subchannel analyses.

4.4.2.10 Flux Tilt Considerations

Radial tilt is a condition where the power is not symmetric between azimuthally symmetric fuel assemblies. Azimuthal power tilt is an allowable limit on operation. Once the flux tilt is beyond an allowable threshold, actions are required to remedy the condition.

The design $F_{\Delta H}$ safety limit inherently accounts for the radial tilt, expressed as:

$$F^{TS}_{\Delta H} = F_{\Delta H}(1 + T_q)$$

where,

$F^{TS}_{\Delta H}$ = COLR enthalpy rise design peaking factor,

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NRC Question No.: 04.04-4

GDC 10, *Reactor design*, requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. FSAR, Tier 2, Section 4.4.2.10, states that the enthalpy rise peaking factor specified in technical specifications includes an additional term, $T_{q,}$ to accommodate azimuthal tilt that could increase the enthalpy rise peaking factor above the design limit for core design calculations. Section 3.10.4 of TR-0915-17564, "Subchannel Analysis Methodology," Revision 1 (ML17046A333), provides a similar discussion of flux tilt. NRC staff needs to establish a finding that the methodology for calculating T_q is suitably conservative. Accordingly, NRC staff requests that the applicant update the FSAR to describe the methodology used to determine T_q.

NuScale Response:

A neutronic analysis of xenon perturbations is performed in order to determine the increase in the enthalpy rise design peaking factor ($F_{\Delta H}$) that is a result of azimuthal tilt (T_q). The increase in $F_{\Delta H}$ is captured with the T_q term as described in Section 3.10.4 of the Subchannel Analysis Methodology topical report (TR-0915-17564). The $F_{\Delta H}$ limit in the Core Operating Limits Report that is used in the subchannel analysis is inclusive of the core design $F_{\Delta H}$ and the azimuthal tilt.

Impact on DCA:

FSAR Section 4.4 has been revised as described in the response above and as shown in the markup provided in this response.

$F_{\Delta H}$ = Design limit for core design calculations, and

T_q = azimuthal tilt.

RAI 04.04-4

The design limit is met by accounting for radial tilt due to asymmetric~~caused by xenon transients that disturb symmetric~~ power peaking. Radial tilt is evaluated in core design calculations by inducing xenon oscillations or transients. Xenon transients are triggered by inserting control rod banks or single control rods as discussed in Section 4.3.2.7. The maximum calculated radial peaking factor after the resulting tilt is then compared to the COLR $F_{\Delta H}$ design peaking factor to ensure that it is below the limit.~~Therefore, the subchannel analysis requires no additional methodology to account for radial tilt as described in Reference 4.4-3.~~

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The subchannel analysis methodology requires no additional factor to account for radial tilt as described in Reference 4.4-3. Table 4.4-4 lists the uncertainties used in the subchannel analyses of AOOs. The development of the uncertainties and the values used in the subchannel analysis are provided in Reference 4.4-3.

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

The NPM is a self-contained nuclear steam supply system comprised of a reactor core, a pressurizer, and two SGs integrated within the RPV. The RPV is an approximately cylindrical steel vessel. The upper and lower heads are torispherical. The pressurizer baffle plate is integrated with the steam plenums, and has orifices to allow surges of water into and out of the pressurizer, and acts as a thermal and hydraulic barrier.

Figure 5.1-1 is a diagram of an NPM and shows the RPV within the containment vessel. Figure 5.1-2 provides a simplified diagram of the RCS. Figure 5.1-3 denotes and describes the major RPV loop flow paths during normal, steady-state, and full-power operating conditions.

4.4.3.1 Plant Configuration Data

Table 5.1-2 lists the nominal operating parameters of the RCS at various power levels. Table 4.4-1 provides geometrical information on the key components of the RPV flow path.

4.4.3.1.1 Core Bypass Flow

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The subchannel analysis considers the flow through the heated core and does not consider the flow that effectively bypasses the fuel rods and is not available to remove heat.