

## NuScaleDCRaisPEm Resource

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**Sent:** Tuesday, November 14, 2017 4:04 PM  
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**Subject:** Request for Additional Information No. 281 RAI No. 9082 (5.4.3)  
**Attachments:** Request for Additional Information No. 281 (eRAI No. 9082).pdf

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

Please submit your technically correct and complete response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

Gregory Cranston, Senior Project Manager  
Licensing Branch 1 (NuScale)  
Division of New Reactor Licensing  
Office of New Reactors  
U.S. Nuclear Regulatory Commission  
301-415-0546

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## Request for Additional Information No. 281 (eRAI No. 9082)

Issue Date: 11/14/2017

Application Title: NuScale Standard Design Certification - 52-048

Operating Company: NuScale Power, LLC

Docket No. 52-048

Review Section: 05.04.07 - Residual Heat Removal (RHR) System

Application Section: 5.4.3

### QUESTIONS

05.04.07-4

10 CFR Part 50, Appendix A, GDC 34 requires in part that a system have the capability to transfer decay heat and other residual heat from the reactor such that fuel and pressure boundary design limits are not exceeded. NuScale has proposed a PDC that is functionally equivalent to GDC 34 with the exception of the requirements associated with electrical power. For the NuScale design, the decay heat removal system (DHRS) serves the decay heat removal function.

In reviewing the NuScale detailed design documentation during an audit, staff discovered that the initial reactor pool conditions for the cases referred to in the FSAR (Figures 5.4-11 through 5.4-15) were 100F, rather than 140F (the maximum allowed reactor pool value in technical specifications). In FSAR section 5.4.3.3.4, these analyses are referred to as "assuming limiting off-normal conditions". This statement aligns with the staff's understanding that the single-train analyses documented are intended to represent bounding analyses for the DHRS performance. Although staff recognizes that NuScale has performed sensitivity studies that demonstrate the DHRS performance for higher reactor pool temperatures, these cases are currently not reflected in the licensing basis documentation. Staff requests that NuScale clarify the description of the limiting DHRS performance in the FSAR by either revising the calculations presented there or adding additional discussion related to the sensitivity cases and characterizing the effect of any conditions that were non-limiting in the cases presented in the FSAR.

05.04.07-5

10 CFR Part 50, Appendix A, GDC 34 requires in part that a system have the capability to transfer decay heat and other residual heat from the reactor such that fuel and pressure boundary design limits are not exceeded, with suitable redundancy in components and features, to assure that the system safety function can be accomplished, assuming a single failure. NuScale has proposed a PDC that is functionally equivalent to GDC 34 with the exception of the requirements associated with electrical power. For the NuScale design, the decay heat removal system (DHRS) serves the decay heat removal function.

FSAR Section 5.4.3.2 states that "the DHRS function is dependent on the closure of the associated safety relief MSIVs and FWIVs." Staff's understanding of this statement is that any single failure in the DHRS can render only one train of DHRS inoperable (while secondary isolation valves exist as stated in Section 5.4.3.2, they are not safety related and therefore are not relied on in the application of single failure when evaluating the system

itself). Based on the information available, a single train appears to be sufficient for all events where DHRS is credited save for a steam generator tube rupture, which could result in the initiating event (the tube rupture) rendering one train of DHRS inoperable while a single failure of an isolation valve renders the other train inoperable. Clarify, in the FSAR, the description in Section 5.4.3.2 as applicable to the steam generator tube rupture event and clearly state which, if any, non-safety related valves are being credited for isolation.