## NuScaleDCRaisPEm Resource

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**Subject:** Request for Additional Information No. 55, RAI 8835
**Attachments:** Request for Additional Information No. 58 (eRAI No. 8835).pdf

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

Please submit your 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
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# Request for Additional Information No. 58 (eRAI No. 8835)

Issue Date: 06/08/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 03.09.04 - Control Rod Drive Systems
Application Section:

QUESTIONS

03.09.04-1

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The control rod drive system (CRDS) is one such SSC.

General Design Criterion (GDC) 1, "Quality standards and records", in 10 CFR Part 50, Appendix A, (as further specified in 10 CFR 50.55a), requires that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

GDC 2, "Design bases for protection against natural phenomena," in 10 CFR Part 50, Appendix A, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

GDC 14, "Reactor coolant pressure boundary," in 10 CFR Part 50, Appendix A, requires that the reactor coolant pressure boundary (RCPB) portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

Very little technical information in Figure 4.6-2 is legible. Tier 2 Section 3.9.4.1.1 of the NuScale DCD states:

> The remote disconnect mechanism coil and latches are capable of remotely connecting and disconnecting the drive shaft from the CRA, as the drive shafts are not accessible during reactor module disassembly, as customary for the current fleet of PWRs.

Update Figure 4.6-5 to include a detailed presentation of the configuration of the latch mechanism assembly. Also, provide definitions for acronyms contained in Figure 4.6-5. Provide legible and detailed drawings (including component identification, class breaks, and dimensions) for all drawings related to SRP Section 3.9.4 to better describe the design of this system and allow staff to make a safety finding for GDC 1, 2, and 14.

03.09.04-2

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The control rod drive system (CRDS) is one such SSC.

General Design Criterion (GDC) 1, "Quality standards and records", in 10 CFR Part 50, Appendix A, (as further specified in 10 CFR 50.55a), requires that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

GDC 2, "Design bases for protection against natural phenomena," in 10 CFR Part 50, Appendix A, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

CRDM support structures are shown in Figure 4.6-1, and DCD Tier 2 Section 3.9.3.1.2 briefly mentions the CRDM seismic supports located on both the RPV and CNV head as ASME Code Class 1, Seismic Category I component supports. However, Figure 5.1-1 also illustrates the CRDM support structures, showing a different number of support structures than Figure 4.6-1. Additionally, DCD Tier 2 Section 3.9.4 does not discuss these support structures or any other means in which the CRDS is supported, despite discussion in the DCD regarding the very long length of the control rod drive shafts when compared to traditional large light water reactors.  Provide an explanation of the support configuration in order for staff to make a safety finding for the review area of GDC 1 and 2.

## 03.09.04-3

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.  The control rod drive system (CRDS) is one such SSC.

General Design Criterion (GDC)  2, "Design bases for protection against natural phenomena," in 10 CFR Part 50, Appendix A, requires that the CRDS be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions.

GDC 14, "Reactor coolant pressure boundary," in 10 CFR Part 50, Appendix A, requires that the reactor coolant pressure boundary (RCPB) portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

GDC 26, "Reactivity control system redundancy and capability," in 10 CFR Part 50, Appendix A, requires that the CRDS be one of the independent reactivity control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation, including anticipated operational occurrences.

GDC 27, "Combined reactivity control systems capability," in 10 CFR Part 50, Appendix A, requires that the CRDS be designed with appropriate margin, and in conjunction with the emergency core cooling system, be capable of controlling reactivity and cooling the core under postulated accident conditions.  The NuScale Design Certification applicant (NuScale or the applicant) has proposed an exemption from this criterion and proposes a principal design criterion (PDC) 27, which states:

> The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.
>
> Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods, provided the probability for a return to power assuming a stuck rod is sufficiently small and specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.

GDC 29, "Combined reactivity control systems capability," in 10 CFR Part 50, Appendix A, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.

DCD Tier 2 Section 4.6.2 mentions that a failure modes and effects analysis has evaluated failures of the CRDM, but results of this analysis are not discussed.  Provide this analysis or describe the postulated failures (both mechanical and electrical) in order to allow staff to make a safety finding for GDC 2, 14, 26, 27, and 29.

## 03.09.04-4

The NRC regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.  The control rod drive system (CRDS) is one such SSC.

General Design Criteria (GDC) 26, "Reactivity control system redundancy and capability," in 10 CFR Part 50, Appendix A, requires that the CRDS be one of the independent reactivity control systems that is designed with appropriate margin to assure its reactivity control function under conditions of normal operation, including anticipated operational occurrences.

GDC 27, "Combined reactivity control systems capability," in 10 CFR Part 50, Appendix A, requires that the CRDS be designed with appropriate margin, and in conjunction with the emergency core cooling system, be capable of controlling reactivity and cooling the core under postulated accident conditions.  The NuScale Design Certification applicant (NuScale or the applicant) has proposed an exemption from this criterion and proposes a principal design criterion (PDC) 27, which states:

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GDC 29, “Combined reactivity control systems capability,” in 10 CFR Part 50, Appendix A, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.

Additional detail on the method of operations is required to make a safety finding for GDC 26, 27, and 29. Statements like “coils are energized in the sequence,” when describing the stepping process provide an insufficient level of detail to make a determination that the operation sequence does not place the system in a non-fail-safe configuration. Provide additional detail on the configuration of the latching mechanism (e.g. how many latches per mechanism, redundancies present in function, etc.). Include specific language to indicate that the CRA drops fully into the core and that the reactor trip function is achievable during any part of the insertion/withdrawal sequence under all design conditions in the discussion of the reactor trip function. Examples of more detailed discussion methods of operation may be found in the DCDs for other design centers, such as AP1000 or EPR.

03.09.04-5

The NRC regulations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The control rod drive system (CRDS) is one such SSC.

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GDC 29, “Combined reactivity control systems capability,” in 10 CFR Part 50, Appendix A, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.

Guidance in the SRP states that “of particular interest are any new and unique features that have not been used in the past.” DCD Tier 2 Section 1.5.1.7 states two new features – a remote disconnect mechanism and rod position indication. DCD Tier 2 Section 3.9.4.4 provides a list of unique features as a longer control rod drive shaft and a remote disconnect mechanism. The remote disconnect coil is considered one of the four main coils in the drive coil assembly and is used to remotely connect and disconnect the drive shaft from the CRA, as described in DCD Tier 2 Section 3.9.4.1.1. DCD Tier 2 Section 3.9.4.1.2 states:

> During operation, the CRA in each control bank are held in place by the control rod drive shafts when the drive coils are energized….. When a reactor trip signal occurs, the operating coils are de-energized.

Provide additional information about the remote disconnect mechanism and other new and unique features of the NuScale CRDM design to support a staff safety finding for GDC 26, 27, 29 (e.g. whether the CRA connection to the CRDM drive shaft is maintained by an energized coil). Is the remote disconnect coil de-energized on a reactor trip? What are the effects of this de-energization (e.g. a decoupled CRA from the CRDM drive shaft)?

03.09.04-6

The NRC regulations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The control rod drive system (CRDS) is one such SSC.

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> Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods, provided the probability for a return to power assuming a stuck rod is sufficiently small and specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.

GDC 29, “Combined reactivity control systems capability,” in 10 CFR Part 50, Appendix A, requires that the CRDS, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of anticipated operational occurrences.

DCD Tier 2 Section 3.9.4.4 describes pre-operational testing to verify that design requirements are met for insertion, withdrawal, and drop times. DCD Tier 2 Section 4.2.4.2.3 states that the drop time is provided in Figure 4.3-23. This figure shows a plot of position versus time, requiring visual estimation to determine the drop time indicated which is insufficient for staff to make a safety finding. Provide a numerical drop time as well as numerical values for other important operational parameters (e.g. trip delay). This information is necessary to support a staff safety finding for GDC 26, 27, and 29.

03.09.04-7

The NRC regulations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The control rod drive system (CRDS) is one such SSC.

General Design Criterion (GDC) 1, “Quality standards and records”, in 10 CFR Part 50, Appendix A, (as further specified in 10 CFR 50.55a), requires that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

GDC 14, “Reactor coolant pressure boundary,” in 10 CFR Part 50, Appendix A, requires that the reactor coolant pressure boundary (RCPB) portion of the CRDS be designed, constructed, and tested for the extremely low probability of leakage or gross rupture.

DCD Tier 2 Section 3.9.4.1.1 states that the sensor coil assembly is attached to the rod travel housing (a portion of the RCPB), but it is unclear how this attachment is made. Please specify the means of attachment for the sensor coil and drive coil assemblies to the rest of the CRDM system. This information supports a staff safety finding for GDC 1 and 14.

03.09.04-8

The NRC regulations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50 specify principal design criteria to establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The control rod drive system (CRDS) is one such SSC.

General Design Criterion (GDC) 1, “Quality standards and records”, in 10 CFR Part 50, Appendix A, (as further specified in 10 CFR 50.55a), requires that the CRDS be designed to quality standards commensurate with the importance of the safety functions to be performed.

DCD Section 3.9.4 discusses meeting design requirements in accordance with 10 CFR 50.55a, but not construction requirements. Confirm, within the DCD, that construction will be in accordance with the same codes used for the design of the components. Specify the codes and standards used to ensure the satisfaction of 10 CFR 50.55a requirements. For non-pressurized components, provide a discussion of the codes and standards used for design and construction and the design margins achieved, allowable stress and deformation limits utilized, how fatigue is considered, and how these are comparable to other similar designs. This information is necessary for staff to make a safety finding for GDC 1.

03.09.04-9

DCD Tier 2 Section 3.9.4 contains several instances where specific wording should be revised to enhance clarity and provide direct indication of regulatory compliance. Several of these instances are noted below:

-DCD Tier 2, Section 3.9.4.2 specifies that pressure boundary parts of the CRDS are in accordance with ASME BPVC Section III, NB. The NRC believes this to be applicable to the RCPB portions of the CRDS. However, the CRDS cooling water piping, a pressure boundary listed in Table 3.2-1 as part of the CRDS, is a Quality Group B component. Please clarify whether the CRDM cooling system is relied on to perform any safety-related functions and if it is in accordance with ASME Code Section III, NB, or if the term “pressure boundary parts of the CRDS” should be clarified to read “RCPB portions of the CRDS.”

-DCD Tier 2, Section 3.9.4 states that a positive means of insertion of the control rods is always maintained and, combined with the design of the CRDS, a margin of safety is provided that accommodates postulated malfunctions such as stuck rods. Please elaborate on how the design of the CRDS specifically provides a margin of safety.

-Page 3.9-33 states that “the NuScale Power Plant design complies with the relevant requirements of the following General Design Criteria;” however, 10 CFR 50.55a is separate from the General Design Criteria. Please correct this regulatory reference to indicate such as “GDC 1 (as further specified in 10 CFR 50.55a).”

-The applicant’s statement on Page 3.9-33 regarding dynamic analysis of the CRDM for the SSE event to comply with GDC 2 requirements only indicates pressure integrity is maintained during an SSE. Please clarify if this capability will also be maintained after the SSE.

-The applicant’s statement regarding seismic qualification of CRDS electrical and I&C components indicates that they can fully operate after the seismic event. Please clarify if this qualification will show that they can fully operate during the seismic event.