August 07, 2017

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. 58 (eRAI No. 8835) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 58 (eRAI No. 8835)," dated June 08, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 8835:

- 03.09.04-1
- 03.09.04-2
- 03.09.04-3
- 03.09.04-4
- 03.09.04-5
- 03.09.04-6
- 03.09.04-7
- 03.09.04-8
- 03.09.04-9

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 58 (eRAI No. 8835). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The proprietary enclosures have been deemed to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Marieliz Vera, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8835, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8835, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0817-55333

NuScale Power, LLC
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Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8835, proprietary

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Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8835, nonproprietary

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

NuScale Response:

As discussed during a June 7, 2017 closed call with the staff, FSAR Figure 4.6.2 provides an overview of the CRDM, and more detail is provided by the other Section 4.6 Figures. Therefore, the CRDM details are depicted by all of the Section 4.6 Figure, not by any individual Figure. For clarity, FSAR Section 4.6.1 has been updated to describe each of the Section 4.6 Figures more clearly, and to describe the interrelationship between the Figures.

A detailed description of the latch mechanism, depicted by Figure 4.6-5, and its operation has been added to Section 3.9.4.1.1. Also, Figure 4.6-5 has been updated to define all acronyms contained in the Figure. Figure 4.6-2 has also been updated to provide a broader overview.

Impact on DCA:

FSAR Sections 3.9.4 and 4.6 along withFSAR Figures 4.6-2 and 4.6-5 have been revised as described in the response above and as shown in the markup provided with this response.

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

NuScale Response:

Two support structures are provided for the CRDMs outside of the RPV, the "CRDM Support Structure" on the top of the RPV and "CRDM Support Frame" in the top dome of the CNV head. The design for the CRDM support structure consists of a box around the perimeter of the top of the CRDM latch housings (at mid-height of the mechaism) with adjusting screws to set contact.

This box is supported by a four legged tower. The design for the CRDM support frame structure consists of a box around the perimeter and adjusting screws to remove the space between each CRDM. This support frame provides support for the CRDMs at the top of the CRDM rod travel housings. A description of the location of these support structures, along with references to Figures 4.6-1 and 5.1-1, that depict them, were added to FSAR Section 3.9.4.1.

Internal to the upper riser, control rod drive shaft lateral supports are provided for the CRD shafts that extend down from the drive mechanisms to the control rod assemblies. In addition to these dedicated CRD shaft supports, the pressurizer baffle plate provides a lateral support point for the shafts. The drive shaft supports are depicted in FSAR Figures 3.9-1, 3.9-2 and 5.1.1. These figures have been updated to show a consistent number of drive shaft supports. The design alignment tolerance limits for these supports will be determined as part of the misalignment drop tests.

Impact on DCA:

FSAR Section 3.9.4 along withFSAR Figures 3.9-1, 3.9-2 and 5.1-1 have been revised as described in the response above and as shown in the markup provided with this response.

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

NuScale Response:

NuScale has provided ER-A022-3459, Revision 1, "NuScale CRDM Failure Modes and Effects Analysis (FMEA)" for NRC staff audit. The FMEA provides a table that identifies the probability and operational and safety consequences of postulated passive and active sub-component failures. The table also identifies preventive or corrective actions to mitigate each of these postulated failures.

Impact on DCA:
There are no impacts to the DCA as a result of this response.

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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:

> 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.

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.

NuScale Response:

Additional descriptive information has been added to FSAR Section 3.9.4.1.1 for the latch mechanism assembly (LMA), including a description of the three types of gripper latches; stationary gripper (SG), movable gripper (MG) and remote disconnect gripper (RDG) latches. Additional descriptive information has also been added to Section 3.9.4.1.1 concerning the remote disconnect mechanism (RDM). Section 3.9.4.1.2 has been updated to add a description of the different modes of CRDM operation, including a statement indicating that reactor trip is achievable during any of these modes of operation.

Impact on DCA:

FSAR Section 3.9.4 has been revised as described in the response above and as shown in the markup provided in this response.

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

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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> 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.

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)?

NuScale Response:

The remote disconnect coil is always de-energized during normal operations and remains in this de-energized state during a reactor trip. A detailed description of remote disconnect mechanism (RDM), which allows separation of the CRA drive shaft from the CRA prior to refueling has been added to FSAR Section 3.9.4.1.1. The FSAR update describes RDM engagement and disengagement operation and describes an alternative non-remote means of disengagement of the CRA drive shaft, if remote disconnection is not possible. Also, Section 3.9.4.1.1 was updated to clearly indicate that the reactor trip function occurs whenever power is removed from the CRDM.

Section 1.5.1.7 has been modified. The statement indicating that rod position indication is not a common feature in CRDMs (and is unique to the NuScale design) has been removed. Although the specific implementation of rod position indication (RPI) in the NuScale design has some differences from existing CRDM designs, the presence of and overall design of the rod position indication does not warrant characterization as "unique" or "first-of-a-kind". Minor differences for the NuScale RPI are the level of integration with other CRDM components and the specific environmental operating conditions (which actually apply to the entire CRDM component, not the RPI in particular).

Impact on DCA:

FSAR Sections 1.5.1.7 and 3.9.4 have been revised as described in the response above and as shown in the markup provided with this response.

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

General Design Criterion (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.
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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.

NuScale Response:

The numerical values plotted in Final Safety Analysis Report (FSAR) Figure 4.3-23, Control Rod Trip Position versus Time after Trip, are provided below. As stated in Section 4.3, these values are based on a calculation that is considered to be realistic and yet slightly conservative. A more conservative bounding control rod drop time is used in the Chapter 15 analyses (See Figure 15.0-1).

The trip delay time is discussed in FSAR Section 7.1.1.2.1 and the assumed delay times for safety analysis are provided in Table 7.1-6. These delay times include a standard 1.0-second engineered safety features actuation system signal processing time, and a 1.0-second delay for reactor trip breaker (RTB) response time that includes the control rod drive shaft delatch time. Control rod free fall will begin within 150 milliseconds regardless of the position of the mechanism at the time of power interruption or trip (included in RTB signal delay).

Table
Control Rod Trip Position versus Time after Trip
(Figure 4.3-23)

Distance (inches)	Time (seconds)
0.000	{{
2.000	
4.000	
6.000	
8.000	
10.000	
12.000	
14.000	
16.000	
18.000	
20.000	
22.000	
24.000	

26.000	
28.000	
30.000	
32.000	
34.000	
36.000	
38.000	
40.000	
42.000	
44.000	
46.000	
48.000	
50.000	
50.476	
52.000	
54.000	
54.437	
54.599	
55.200	
56.000	
58.000	
60.000	
62.000	
64.000	
66.000	
68.000	
68.565	
70.000	
72.000	
74.000	
76.000	
76.183	}}2(a), (c), ECI

Note: The total rod travel is 76.183 inches and the total rod drop time is {{ }}2(a),(c),ECI seconds.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

NuScale Power

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

NuScale Response:

The question requests details concerning how the sensor coil is attached to the rod travel housing. The statement that the sensor coil assembly is "attached" is incorrect. The coil assembly slides over the rod travel housing and sits on a ledge at the base of the rod travel housing. This allows for unrestricted thermal motion between the long axes of the two assemblies. FSAR Section 3.9.4.1.1 has been revised accordingly.

Impact on DCA:

FSAR Section 3.9.4 has been revised as described in the response above and as shown in the markup provided with this response.

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

NuScale Response:

FSAR Section 3.9.4 has been updated to indicate that the pressure retaining components of the CRDS are designed and constructed in accordance with ASME Boiler and Pressure Vessel Code (BPVC), Section III Division 1, and are consistent with the requirements of 10 CFR 50.55a. Section 3.9.4.2 has been updated to clearly define the applicable Code, Code edition and subsections for the RCPB pressure retaining portions of the CRDS and the non-RCPB pressure retaining portions of the CRDS.

A major non-pressure retaining CRDM component is the long drive shaft. Since this is a seismic category 1 component and it meets the definition of a Subsection NG, internal structure, Subsection NG is applied. Application of Subsection NG includes design, material, fabrication and inspection. Other non-pressure boundary parts will meet the applicable ASME/ASTM standards. The CRDM ASME Design Specification has been updated to include this requirement.

A drive shaft/CRA drop test program, to determine deflection limits for proper scram times and stress levels, is scheduled for completion the second quarter of 2018. Therefore, design margins cannot be defined at the present.

Impact on DCA:

FSAR Section 3.9.4 has been revised as described in the response above and as shown in the markup provided with this response.

The pressure boundary materials are in accordance with the requirements of ASME BPVC, Section II. These pressure boundary materials are described in Section 5.2.3. The non-pressure boundary materials of the CRDS are described in Section 4.5.1.

The CRDM, which is considered part of the reactor coolant pressure boundary (RCPB), is designed in accordance with 10 CFR 50.55a. The pressure boundary components are designed to meet the stress limits and design and transient conditions specified in Table 3.9-6. The preservice and inservice inspection requirements of ASME Code, Section XI (Reference 3.9-2) are applicable to the CRDM. Welding is performed in accordance with the ASME BPVC Code, Section III, Division I, Subsection NB. The requirements to prevent brittle fracture presented in ASME BPVC Code, Section III, Division I, Subsection NB are also applicable to the CRDM. The CRDM bolting is designed in accordance with the ASME BPVC Code, Section III, as addressed in Section 3.13. Additional information on compliance with codes and code cases for the RCPB is provided in Section 5.2.1.

RAI 03.09.04-8

The design, fabrication, inspection and testing of non-pressure retaining components typically do not come under the jurisdiction of the ASME Code. For those materials which do not have established stress limits, the limits are based in the material specification mechanical property requirements. A major non-pressure retaining CRDM component is the long drive shaft. Since this is a Seismic Category I component that meets the definition of an ASME Section III, Subsection NG, internal structure, ASME BPCV, Section III, Division 1, Subsection NG Code requirements are applied for design, material fabrication and inspection.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The CRDM internal design and normal operating conditions are listed below:

- design pressure (RCS) - 2,100 psia
- normal operating pressure (RCS) - 1,850 psia
- design temperature (RCS) - 650 degrees Fahrenheit
- normal operating temperature (RCS) - 625 degrees Fahrenheit

The CRDMs are designed for the loading combinations and loading values specified in Section 3.9.3.

The worth of the 16 CRA in conjunction with the CRDS trip function is sufficient to overcome a stuck rod event. In addition, design requirements have been established for clearances during seismic, thermal expansion and dynamic events.

3.9.4.4 Control Rod Drive System Operability Assurance Program

The ability of the CRDS pressure housing components within the CRDMs to perform throughout the operating design life of 60 years is confirmed by the design report required by the ASME BPVC, Section III (Reference 3.9-1).

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

eRAI No.: 8835
Date of RAI Issue: 06/08/2017

NRC Question No.: 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.

NuScale Response:

FSAR Section 3.2.2 indicates that Quality Group A applies to pressure-retaining components that form part of the reactor coolant pressure boundary (RCPB) and that these components meet the requirements for Class 1 components of the ASME BPVC. Section 3.9.4.2 has been revised to clarify the text to indicate that the RCPB portions of the CRDS are in accordance with ASME BPVC, 2013 Edition, Section III, Division 1, Subsection NB. The CRDS cooling water piping is not safety related or risk significant, however, it is classified Quality Group B.

FSAR Section 3.9.4 indicated "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." This statement has been clarified for consistency with language of SRP 3.9.4, Section 3.9.4 now indicates "a positive means of insertion of the control rods is always maintained and, combined with the design of the CRDS, provides a margin for malfunctions such as a stuck rod (Refer to Section 4.3.1.5)." Section 4.3.1.5 describes how insertion of the control rod assemblies provide sufficient shutdown margin to accommodate the reactivity affect of a stuck rod.

The compliance statement for GDC 1 has been clarified. The statement in FSAR Page 3.9-33 now indicates "GDC 1 (as further specified in 10 CFR 50.55a), as they relate to the CRDS being designed to quality standards commensurate with the importance to safety functions being performed."

The CRDM pressure boundary is maintained during and after an SSE. FSAR Section 3.9.4 has been revised accordingly.

Rod position indication (RPI), CRDM control, and CRDM electrical power components are classified as non-safety related, non-risk significant (as shown in Table 3.2-1). RPI is a PAM Type B variable. Neutron flux is utilized to demonstrate successful rod insertion. The CRDMs delatch and the CRAs are inserted into the core upon any loss (or failure) of electrical power. Therefore, the sentence in Section 3.9.4 indicating CRDS electrical and I&C components can fully operate during and after a seismic event was removed.

Impact on DCA:

FSAR Section 3.9.4 has been revised as described in the response above and as shown in the markup provided with this response.

PIM. The NRELAP5 code, which is based on a commercial version of RELAP5-3D, is a thermal-hydraulic analysis code developed at NuScale for use in the thermal-hydraulic design, safety analysis, and licensing of the NPM. The code incorporates models and correlations specific to the unique design features of the NPM. The PIM code is a NuScale-developed proprietary code used to assess the stability characteristics of the NPM during operation.

The NIST-1 is a scaled representation of the NPM reactor, containment, and reactor pool systems. It is constructed of stainless steel and has a maximum operating pressure of 1650 psia (11.4 MPa) and temperature of 630 degrees F (605 degrees K). Being a scaled facility, the volumes, lengths, and areas for the NIST-1 geometry are obtained by multiplying the respective NPM design volumes, lengths, active heat transfer areas, and flow areas by the applicable scale factors determined through a detailed scaling analysis. This process ensures the NIST-1 properly captures the important thermal-hydraulic phenomena and processes that would occur in the plant.

A series of tests have been completed at the NIST-1, located on the Oregon State University campus in Corvallis, OR, in support of NuScale's Design Certification Application. These tests include:

- facility characterization tests used to develop the NRELAP5 model of the NIST-1.
- loss-of-coolant accident (LOCA) tests used to validate NRELAP5 for LOCA analyses.
- flow-stability tests used to validate PIM for reactor stability analyses.
- non-LOCA (anticipated operational occurrence) tests used to validate NRELAP5 for non-LOCA analyses.
- long-term cooling tests used to validate NRELAP5 for long term cooling analyses.

Data obtained from the NIST-1 tests identified above have been used to successfully validate the NRELAP5 and PIM codes for LOCA, non-LOCA, flow stability, and long term cooling applications. This test program was inspected by the NRC in accordance with Inspection Procedures IP 35034, 35017, and 36100.

1.5.1.7 Control Rod Drive Mechanism Proof Test

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-7, RAI 03.09.04-9

The control rod drive mechanism for the NPM contains ~~two new~~ features that are not common in conventional control rod drive mechanisms: a remote disconnect mechanism and ~~rod position indication~~a long control rod drive shaft. A proof-of-concept testing program was conducted to demonstrate the validity of these new designs with regard to performance, reliability, and repeatability of each system. Additional testing to determine misalignment limits is described in Section 1.5.1.12.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-7, RAI 03.09.04-9

Testing was completed at the Curtiss Wright facilities in Cheswick, PA, for both ~~normal~~remote connect and remote disconnect operation of the coils. The test setup included a functional drive rod assembly, a prototypic remote disconnect gripper coil, a prototypic remote disconnect gripper latch, a prototypic lift coil, and weights to simulate the control rod assembly (CRA) with a prototypic CRA hub socket.

The remote disconnect mechanism was found to provide a reliable and repeatable method to engage and disengage the CRA within the reactor pressure vessel. This is consistent with the results of the remote operation, lift verification, and manual disengagement testing that was performed.

The tests provided a demonstration of hardware performance, which has been extended to aid in the design of drive rod position detection circuitry. Information gained from this testing has been used as a development tool to improve the design and does not create a design basis for the final control rod drive mechanism.

1.5.1.8 Steam Generator Flow Restrictor Hydraulic Test

To maintain secondary-side (within-tube) flow stability during operation of the NPM helical SG, a flow restriction device is used at the inlet to each SG tube. Performance of the flow restriction device varies as a function of power level (flow rate), and manufacturing and installation tolerances. For some potential flow restrictor designs, an uncertainty exists in the analytically-determined performance characteristics. Therefore, testing of potential flow restriction device designs, including variants to account for installation and manufacturing tolerances, was conducted to confirm the actual performance is acceptable in comparison to the expected design performance. Specifically, testing was needed to confirm flow-dependent pressure drop performance and vibration behavior.

The three objectives of the flow restrictor hydraulic test were to: (1) experimentally determine performance (K-factor) for flow-restrictor design concepts to verify design performance analyses, (2) assess the vibration behavior of the flow restrictors under different operating conditions, and (3) determine sensitivity of flow restrictor design to manufacturing and installation tolerances. Flow testing was performed for three different flow restriction devices (configured in a near-prototypical SG tube inlet) using water at a range of flow conditions representative of Reynolds numbers prototypic of NPM operating conditions between1 percent and 110 percent of full flow.

Three types of flow restrictors (annular, threaded, and center flow) were fabricated and tested at the Alden Research Laboratory hydraulic test facilities in Holden, MA. Multiple-instrumented prototypical SG tubes with shortened lengths were used. Water temperature, pressures, flow rates, and vibration were recorded for each flow restrictor under varying conditions. All restrictors were able to be tested at conditions such that excessive vibration did not occur and no tube contact was observed. The tests provided a demonstration of hardware performance, which has informed the design of the actual flow restrictor device.

1.5.1.9 Steam Generator Flow-Induced Vibration

In accordance with Regulatory Guide 1.20, a comprehensive vibration assessment program (CVAP) for reactor internals must be completed prior to commercial operation. The CVAP must verify the structural integrity of the reactor internals for flow-induced vibrations by analytical, testing, in-service measurement, and inspection

The design of the pressure relief valves uses the guidance of ASME BPVC Code Section III, Appendix O, "Rules for the Design of Safety Valve Installations," with respect to calculation of reaction loads. The reaction forces and moments are based on a static analysis with a dynamic load factor of 2.0 unless a justification is provided to use a lower dynamic load factor. A dynamic structural analysis may also be performed to calculate these forces and moments. The safety or relief valves that discharge directly to the atmosphere or containment are considered open-discharge configurations. The analysis requirements for these devices are addressed in Section 3.12.

3.9.3.3 Pump and Valve Operability Assurance

The NuScale Power Plant does not rely on pumps to perform any safety-related functions. A listing of the active safety related valves is provided in Section 3.9.6.

Active valves are subject to factory tests to demonstrate operability prior to installation. These tests are followed by post-installation testing in the plant. The factory- and post-installation tests performed are described in the inservice testing (IST) program. The IST requirements for ASME Class 1, Class 2, and Class 3 components are contained in the ASME Operation and Maintenance (OM) Code (Reference 3.9-3).

A description of the functional and operability design and qualification provisions and IST programs for safety-related valves is provided in Section 3.9.6. Environmental qualification of safety-related valves is discussed in Section 3.11. The seismic qualification of safety-related valves is performed in accordance with ASME QME-1 (Reference 3.9-4) as endorsed by RG 1.100, Revision 3 and as discussed in Section 3.10.

The stress limits are discussed in Section 3.9.3.1.

3.9.3.4 Component Supports

Section 3.9.3.1 provides the load combinations, system operating transients, and stress limits for component supports.

As described in Section 3.9.3.3, the functionality assurance, environmental and seismic qualification programs that are applied to components are also applied to the associated supports.

3.9.4 Control Rod Drive System

The control rod drive system (CRDS) consists of the control rod drive mechanisms (CRDMs), and related mechanical components that provide the means for control rod assembly (CRA) insertion into the core as described in Section 4.6, as well as the rod position indication to the module control system. The CRDM control cabinets, rod position indication cabinets and associated cables, plus the CRDS cooling water piping inside containment are also part of the CRDS. The CRDM is an electro-magnetic device which moves the CRA in and out of the nuclear reactor core and is connected to two independent rod position indication trains. The CRDS provides one of the independent reactivity control systems as discussed in GDC 26 and NuScale Principal Design Criteria (PDC)-27.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The control rods and their drive mechanisms are capable of reliably controlling reactivity under conditions of normal operation, including AOOs, or under postulated accident conditions. The CRDM internals, consisting of the latch mechanism and drive shaft are, therefore, safety related. 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~~provides a margin for malfunctions such as a stuck rod~~s~~ (refer to Section 4.3.1.5).

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The CRDM internals that ensure positive CRA insertion consist of the latch mechanism and control rod drive shaft and are classified as safety related and risk significant. Portions of the CRDS are a part of the RCPB (specifically the pressure housings of the CRDMs) and are safety related. The system is designed, fabricated, and tested to quality standards commensurate with the safety-related functions to be performed. The design, fabrication, and construction complies with the codes ~~and standards~~ in accordance with 10 CFR 50.55a (refer to Section 3.9.4.2). This provides assurance the CRDS is capable of performing its safety-related functions by withstanding the effects of AOOs, postulated accidents, and natural phenomena, such as earthquakes, as discussed in GDC 1, 2, 14, 26, 29 and PDC-27.

The structural materials of construction for the CRDS are discussed in detail in Section 4.5.1. Materials for the pressure boundary portions of the CRDM are discussed in Section 5.2.3.

The NuScale Power Plant design complies with the relevant requirements of the following General Design Criteria (GDC) of 10 CFR 50, Appendix A and NuScale Principal Design Criteria (PDC):

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

- GDC 1 ~~and 10 CFR 50.55a~~(as further specified in 10 CFR 50.55a), as they relate to the CRDS being designed to quality standards commensurate with the importance of the safety functions to be performed. The NuScale quality assurance program satisfies the requirements of 10 CFR 50 Appendix B and ASME NQA-1 "Quality Assurance Requirements for Nuclear Facility Applications." As such the NuScale QA program provides confidence that the SSC, including CRDS that are required to perform safety functions, will perform the functions satisfactorily.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

- GDC 2, as it relates to the CRDS being designed to withstand the effects of an earthquake without loss of capability to perform its safety-related functions. See Section 3.2 for the seismic classification of the CRDS in accordance with RG 1.29. The seismic analysis is performed for the CRDM to ensure that the components can withstand the effects of natural phenomena without loss of capability to perform their safety functions. Dynamic analysis of the CRDM is performed for the SSE event to ensure that pressure integrity is maintained during and after the SSE and the capability to lower the CRA connect to the CRDM drive shaft is not compromised.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

~~Seismic qualification is performed for the CRDS electrical and instrumentation and controls components to ensure that the CRDM electrical and instrumentation and controls equipment can fully operate after the seismic event. Additional p~~Protection against the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and tsunamis, is provided by locating the CRDS components inside the Reactor Building, which is a Seismic Category I building.

- GDC 14, as it relates to the RCPB portion of the CRDS being designed, constructed, and tested for the extremely low probability of leakage or gross rupture. The pressure-retaining components are seismically and environmentally qualified, ensuring components RCBP is maintained.
- GDC 26, as it relates to the CRDS being 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 AOOs. The CRDS facilitates reliable operator control by performing a safe shutdown (i.e., reactor scram) by gravity dropping of the CRA on a reactor trip signal or loss of power. The CRDS is designed such that core reactivity can be safely controlled and that sufficient negative reactivity exists to maintain the core subcritical under cold conditions.
- PDC-27, as it relates to the CRDS being designed with appropriate margin for reliably controlling reactivity under postulated accident conditions. The ECCS does not perform core cooling by adding any fluid mass. Therefore, a poison addition safety function is not required to compensate for the addition of otherwise nonborated fluid. As discussed in Section 3.1.3, the CRDS and the CVCS, along with the boron addition system, have the combined capability to reliably control reactivity changes and maintain the core cooling capability under postulated accident conditions with appropriate margin for a stuck rod.
- GDC 29, as it relates to the CRDS, in conjunction with reactor protection systems, being designed to assure an extremely high probability of accomplishing its safety-related functions in the event of AOOs. The CRDS fulfills its safety-related functions to control the reactor within fuel and plant limits during AOOs despite a single failure of the system. The CRDS accomplishes safe shutdown (i.e., reactor shutdown via gravity-dropping of the control rod assemblies) on a reactor trip signal or loss of power. The CRDS pressure housing is an ASME Class 1 pressure boundary.

3.9.4.1 Descriptive Information of Control Rod Drive System

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The CRDS is composed of a pressure-retaining housing enclosing the working mechanism, a control rod drive shaft with a coupling for attaching to the CRA hub, external electromagnetic coils with cooling loop heat exchangers, the power/control system, and the rod position indication system. ~~It~~The control rod drive shafts are laterally supported by both the reactor vessel (at the pressurizer baffle plate) as depicted by Figure 5.1-1 and by the CRA drive shaft support structures that are part of the reactor vessel internals (RVI) components in the upper riser as depicted by Figure 3.9-2. The CRDS provides the rod control, reactor scram, and control rod position indication necessary for operation of the reactor module. The CRDS includes the CRDM, the control and indication cabinets and cables, and supporting SSCs as described below and in Section 4.6. Information regarding the CRA and its interface with the fuel system design is in Section 4.2.

The CRDS functional testing program is discussed in Section 3.9.4.4.

3.9.4.1.1 Control Rod Drive Mechanism

The CRDM assembly is a hermetically sealed electro-mechanical device, which moves the CRA in and out of the nuclear reactor core, or may hold the CRA at elevations within the range of CRA travel. If electrical power is interrupted to the CRDM, the CRA (connected to the CRDM drive shaft) is released and inserted by gravity into the core. Figures 4.6-1 through 4.6-6 depict the CRDM assemblies mounted above the pressurizer steam space on the reactor pressure vessel (RPV). The structural materials of construction for the non-pressure boundary portions of the CRDM are discussed in Section 4.5.1. Materials for the pressure boundary portions of the CRDM are discussed in Section 5.2.3. The materials for the CRA are provided in Section 4.2.2.9. Additional characteristics of the CRDMs are provided in Section 4.1.

The reactor core is controlled using 16 CRDMs. One CRDM consists of two pressure housings (including the lower portion called latch housing, and the upper portion called rod travel housing), a latch mechanism assembly internal to the lower pressure housing operated by an outside drive coil assembly, one control rod drive shaft, a rod position indication coil assembly, and the associated wiring and water cooling connections which are described in further detail below. The rods are moved in a controlled manner to maintain control of the power level and power distribution in the core. The CRDM is connected to the CRA at the bottom end of the control rod drive shaft.

The CRDMs insert (scram) the control rod drive shaft and the attached CRA by force of gravity following a power interruption or a reactor trip. The CRDM is capable of a continuous full-height withdrawal and insertion and holding a position during normal operating conditions.

The CRDM components in contact with the primary coolant are designed to operate for a 60 year design life. The CRDM are designed to be replaceable and freely interchangeable without limitations in function and connections.

Control Rod Drive Shaft

The rod drive shaft is the link and the method of transferring force between the CRDM and the CRA. The control rod drive shaft must pass through the upper region of the reactor vessel to allow the CRDM to raise, lower, or hold the CRA. The control rod drive shaft must also interact with the rod position indication sensor coils that communicate the elevation of the control rods. The control rod drive shaft allows for the release of the CRA for refueling purposes.

Drive Coil Assembly

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The drive coil assembly has four main coils: the lift coil, the movable gripper coil, the stationary gripper coil, and the remote disconnect coil. The direct current generated by the control cabinets is sent through a coil which generates a magnetic field; this magnetic field engages the flat-face plunger magnet, which moves the latch arm to engage the control rod drive shaft. The rate at which the movable gripper coil, the stationary gripper coil, and the lift coil are energized determines the speed of the control rod drive shaft. The power from the direct current electrical and alternating current distribution system to the CRDM control cabinet ~~can be~~is interrupted ~~if~~when the reactor trip breakers open, causing the control rods to be inserted via gravity. The CRDS safety function of rapid insertion of the control rods is accomplished when power is removed from the CRDM. Rod movement logic tracks the speed of the control rods, which utilizes direct rod position indication. The rod movement logic has a latching function for providing extra current to the coil(s) during initial movement (startup) to ensure the latch assembly is engaged positively to the control rod drive shaft. 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.

Pressure Housings

The pressure housings include all components of the CRDM that form the pressure boundary for the reactor coolant. The pressure housings are ASME BPVC Section III, Subsection NB components. The pressure housings consist of the latch housing (welded to the reactor vessel head nozzle) ~~and~~, the rod travel housing, and the rod travel housing plug. The rod travel housing is threaded into and seal welded to the top of the latch housing.

Latch Mechanism Assembly

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

~~The latch mechanism assembly consists of three separate latch assemblies that have the ability to grab and release the drive shaft in order to lift and lower the drive shaft in three-eighths-inch incremental steps and support operation of the remote disconnect mechanism. These motions are produced by electromagnetic forces generated by the drive coils. The latch mechanism assembly releases the control rod drive shaft during loss of power. The latch mechanism assembly is shown in Figure 4.6-5.~~The basic functions of the Latch Mechanism Assembly (LMA) are to grab (engage, hold), release, lift, and lower the CRA. The lifting and lowering functions are also referred to as "stepping," and these steps are in 0.375 in. increments. The LMA contains three different latches. From bottom to top, they are the Stationary Gripper (SG) latch, the Movable Gripper (MG) latch, and Remote Disconnect Gripper (RDG) latch, as shown in Figure 4.6-5. The latches grip (or hold) the drive shaft when the teeth of the latch arms are engaged within the grooves in the upper segment of the drive shaft.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The SG and MG latches are used during normal stepping operations, while the RDG latch is used relatively infrequently during maintenance, repair, and refueling operations when the drive shaft is decoupled from the CRA. Since the SG and MG latches both participate in normal stepping, they have similar requirements in terms of loads and cycles, and thus have many similar features. The RDG latch is used only during RDM operation. It is never used during normal stepping or holding operations. The MG latch is used only during stepping. The SG latch is used during stepping and holding. In comparison, the RDG latches have much lower loads and cycles than the "stepping" latches (SG & MG), and are reduced slightly in size and complexity.

The latch assembly attaches to the bottom of the rod travel housing and is inserted into the latch housing.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Remote Disconnect Mechanism

The CRDM includes a remote disconnect mechanism (RDM) which performs the function of disconnecting the CRDM drive shaft from the CRA. The disconnection occurs at the junction between the CRD shaft and the CRA hub (as shown in Figure 4.6-6). The RDM enables the CRDS drive shaft to remain with the upper section of the RPV as the upper RPV is separated from the lower section of the RPV. The CRA is retained with the fuel assembly prior to refueling the reactor. The remote disconnect action at the disconnect point is the result of CRDM gripper actions which are transmitted via mechanical components within the hollow CRD shaft.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-7, RAI 03.09.04-9

The remote disconnect process is facilitated by the combined engagement of stationary grippers (SG) and remote disconnect gripper (RDG) on top of the mechanism (Figure 4.6-5) during the refueling sequence. To begin the disengagement process, the drive shaft with CRA is inserted to the post-SCRAM position and the SG coil and MG coil are energized, subsequent energizing of the RD coil lifts the top of the drive shaft by about seven inches. The first approximate four inches of this lift compresses the RD upper spring solid, the following 1½ inches lifts the drive shaft and extracts the RD coupling from the CRA hub which causes the CRA to drop. The load reduction is indicated by a current response in the RDG coil. After the next approximate 3/8 inch the drive shaft reaches its maximum elevation, and the remaining travel locates the RD coupling 1¼ inch above the elevation at which it last made contact with the CRA hub, at which point the RDG is released and the spring-loaded lower end of the drive shaft expands (Figure 4.6-6). This completes the disengagement process, the system rests now in the post-RD configuration.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

After the reactor has been refueled and the plant is restored to the state it was in at the completion of the RDM disengagement process, the remote engagement process may begin. The engagement process that follows is performed to couple all 16 pairs of CRDS drive shafts and CRAs, and the reverse of above stated process. Once complete, the CRDMs can be operated normally.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

In the event that the CRA drive shaft cannot be remotely disconnected from the CRA remotely, an alternate non-remote method is provided to disengage the CRA through the top of the rod travel housing (Figure 4.6-4). Since operation of the RDM requires the entire CRDM to be operational, there are a number of reasons that could prevent an inadvertent remote disconnect. This includes, but is not limited to, the inability of any of the SG or RDG latches to properly engage, either due to a mechanical failure of the latches, failure of the drive coils, or a failure of the disconnect verification. In the event that RDM operation is not available, the pressure boundary seal weld around the rod housing plug is broken, and the plug removed for tooling access. The top of the drive shaft contains a locking feature that allows for manual lift of the remote disconnect rod and unlock the CRA (Figure 4.6-6).

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-7, RAI 03.09.04-9

Drive Coil Assembly

The drive coil assembly slides over the latch housing and sets on a ledge at the base of the latch housing. The drive coil assembly is depicted by Figure 4.6-3.

Sensor Coil Assembly

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The sensor coil assembly contains the rod position indication coils ~~and is attached to, and supported by the rod travel housing~~the coil assembly slides over the rod travel housing and sets on a ledge at the base of the rod travel housing. The sensor coil assembly is shown in Figure 4.6-4.

3.9.4.1.2 Operation of the Control Rod Drive Mechanisms

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The basic CRDM mechanical and operational requirements are discussed in Section 4.6. The following describes the different modes of CRDM operation. Reactor trip, consisting of full insertion of the CRAs into the core at design conditions, is achievable during any part of the CRDM operating modes described below.

~~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 the signal is given to lift the control rod drive shafts, the CRDM drive coils are energized in the sequence to provide lifting of the control rods step-by-step starting from the rest position. Sequential rod control is necessary to control reactivity addition rates automatically and to control rod programming for the desired flux level. Rod control includes manual mode, automatic mode, and insertion-only automatic mode. Rod selection while in sequential rod control is consistent with rod programming requirements.~~

~~When a reactor trip signal occurs, the operating coils are de-energized. This causes the latch mechanism assembly magnets to drop, retracting the latches from the drive shaft grooves and allowing the drive shaft and the CRA to drop into the reactor core under gravity.~~

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Control Rod Insertion

The control rod insertion sequence begins with only the stationary coil energized and the stationary gripper supporting the control rod. The lift coil is energized and the lift armature and movable gripper are raised 0.375 inches by the magnetic force acting on the armature. The movable coil is energized and the moveable gripper engages the control rod. The stationary coil is de-energized and the load of the control rod is transferred to the movable gripper by the force of gravity. The lift coil is de-energized and the lift armature, movable gripper and the control rod assembly move down 0.375 inches under the force of gravity. The stationary coil is re-energized and the stationary gripper engages the control rod. The movable coil is de-energized and the load is transferred to the stationary gripper by the force of gravity. The insertion sequence is complete. The sequence is repeated for additional insertion steps.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Control Rod Withdrawal

The control rod withdrawal sequence begins with only the stationary coil energized and the stationary gripper supporting the control rod. The movable coil is energized and the moveable gripper engages the control rod. The stationary coil is de-energized and the load of the control rod is transferred to the movable gripper by the force of gravity. The lift coil is energized and the control rod assembly is lifted 0.375 inches by the magnetic force acting on the lift armature. The stationary coil is re-energized and the stationary gripper engages the control rod. The movable coil is de-energized and the load is transferred to the stationary gripper by the force of gravity. The lift coil is de-energized. Only the stationary coil remains energized. The withdraw sequence is complete. The sequence is repeated for additional withdrawal steps.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Control Rod Holding

During most of the plant operating time, the CRDMs hold the CRAs withdrawn from the core in a static position, i.e. holding position. The latches of the LMA grip the drive rod when the teeth of the latch arms are engaged within the grooves in the drive rod. The three latch positions are referred to as "in-contact" (engaged and loaded, holding, closed), "in-clear" (engaged and unloaded, closed), and "out" (disengaged, open).

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

In normal steady state operation, in which stepping is not occurring, and the CRA is being maintained at a particular elevation (i.e., holding position), the stationary gripper (SG) latches are in the in-contact position, and the movable gripper (MG) and remote disconnect gripper (RDG) latches are out.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Control Rod Stepping

During normal stepping operations, the interface between the latch arms and drive rod alternates between three distinct positions. The in-contact position is the position in which the rod and CRA weight are being supported by the latch arms. In the normal stepping sequence, the SG and MG latches cycle through the three positions, but the latches never move in or out when supporting the drive rod. When changing from in-contact to out, or vice versa, the latch/shaft interface always passes through the in-clear position. This minimizes wear at the latch/shaft interface. Whenever the SG or MG latch moves into or out of the in-clear position, the weight of the drive rod is being supported by the other latch.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The main control of the stepping cycle is the voltage profile that is imposed on the three drive coils (SG, MG, and lift). The maximum allowed duration for each one way step (either up or down) is 1.5 seconds. This is derived by dividing the 0.375 inch step by the maximum required velocity of 15 in/min.

3.9.4.2 Applicable Control Rod Drive System Design Specifications

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

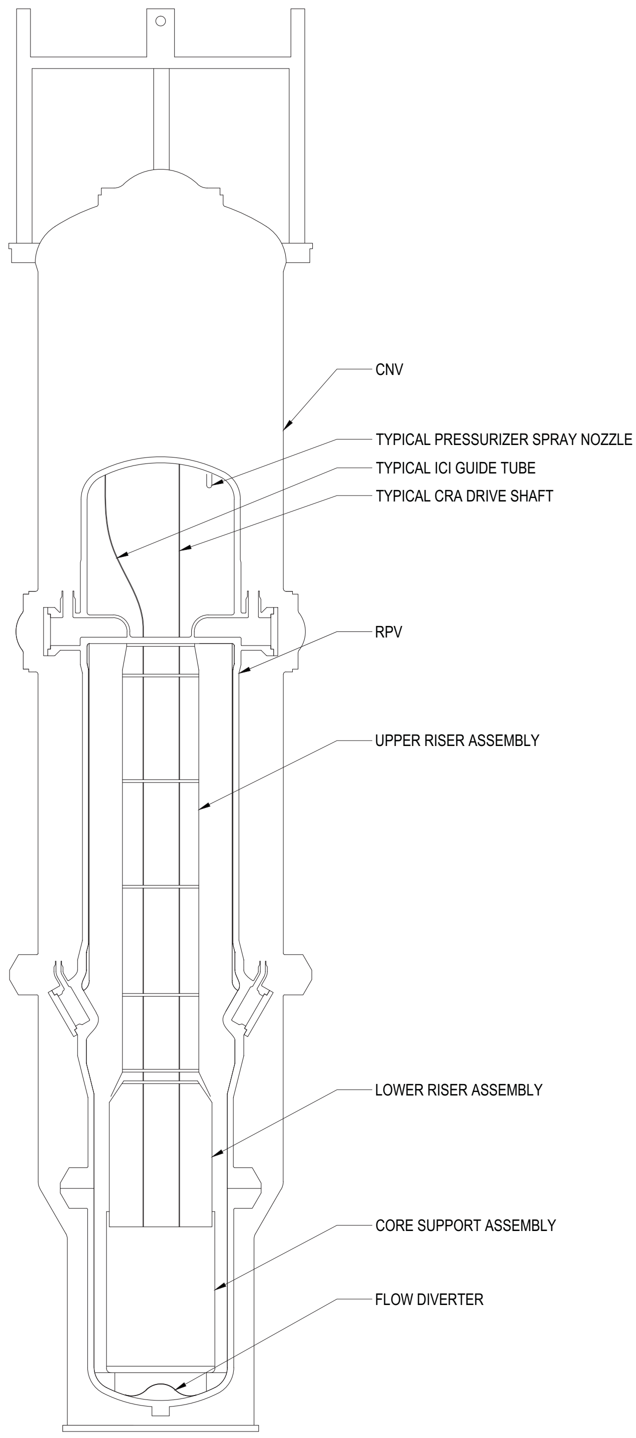
The design, fabrication, construction, examination, testing, inspection, and documentation of the RCPB pressure boundary parts of the CRDS are in accordance with the requirements of ASME BPVC, ~~Code~~2013 Edition, Section III (Reference 3.9-1), Division I, Subsection NB. Classification of the pressure retaining portions of the CRDS is addressed in Section 3.2.2.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The design, fabrication, examination, testing, inspection and documentation for the CRDM coil heat exchangers, cooling tubes and cooling water connectors are in accordance with the requirements of ASME BPVC, 2013 Edition, Section III (Reference 3.9-1), Division 1, Subsection NC. These components are conservatively classified Quality Group B to minimize the potential for fluid leakage inside containment, as discussed by Section 4.5.1. The pressure retaining components of the CRDS are designed, fabricated, constructed, and tested in accordance with ASME BPVC, 2013 Edition, Section III Division 1 and are consistent with the requirements of 10 CFR 50.55a.

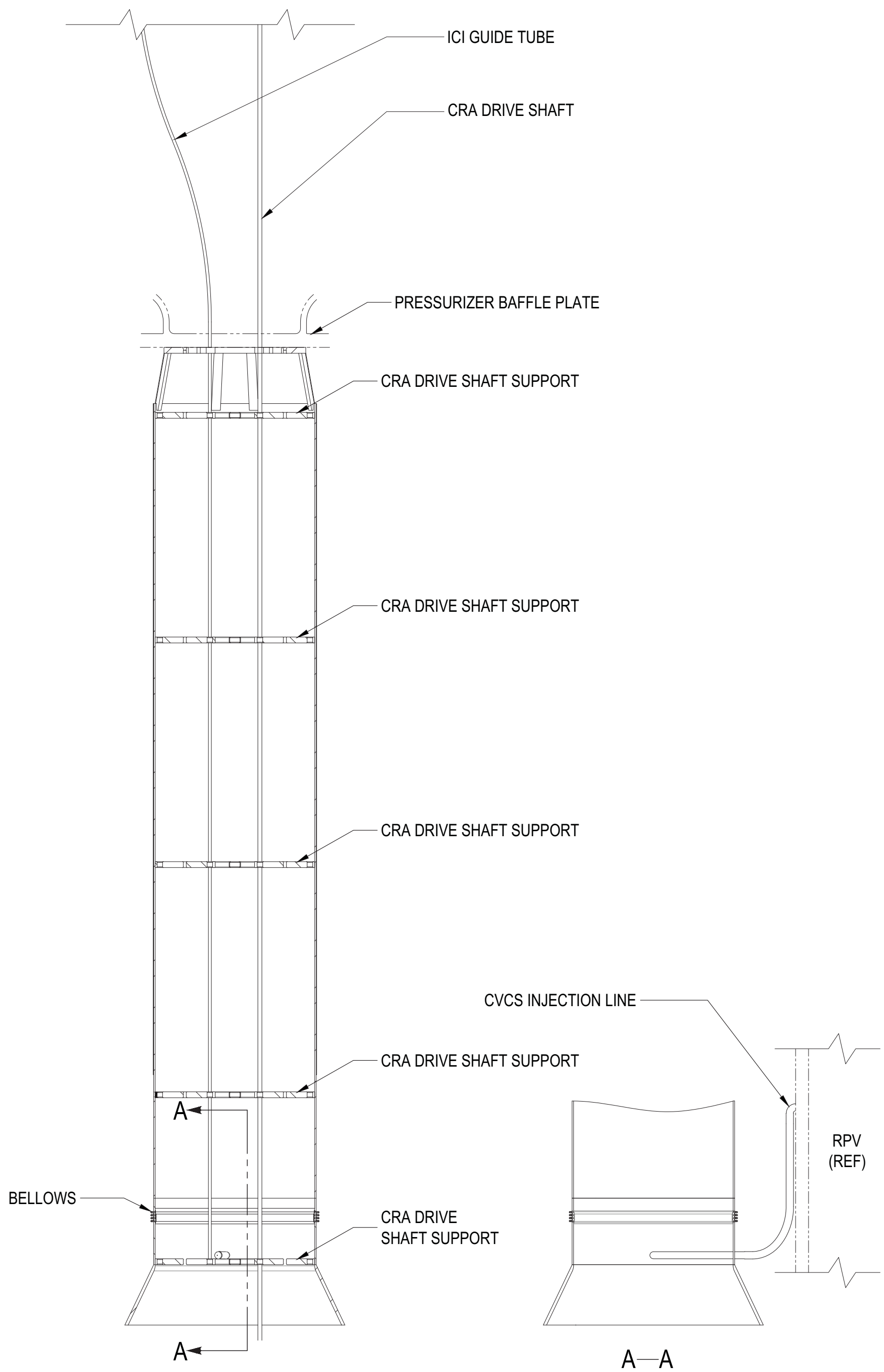
RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Figure 3.9-1: Reactor Module Showing Reactor Vessel Internals Component Assemblies



RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Figure 3.9-2: Upper Riser Assembly



4.6 Functional Design of Control Rod Drive System

The design of the control rod drive system (CRDS) and its supporting structures, systems, and components provides the functional capability to achieve safe shutdown and maintain the fuel cladding acceptance criteria during anticipated operational occurrences (AOOs), infrequent events and accidents.

The CRDS performs the following safety-related functions:

- releases the control rod assemblies during a reactor trip
- maintains the pressure boundary of the reactor pressure vessel

The CRDS performs the following non safety-related functions:

- latching, holding, and maneuvering the CRAs during reactor startup, power operation, and shutdown
- provides rod position indication
- protects fuel integrity during reactor disassembly and reassembly prior to and after refueling

4.6.1 Description of the Control Rod Drive System

The CRDS includes the control rod drive mechanisms (CRDMs) and all electrical and instrumentation and controls components, including rod position indicators, to operate the CRDMs. The CRDM includes the control rod drive shaft, which extends to the coupling interface with the control rod assemblies (CRAs) in the reactor pressure vessel. The CRDS supports the CRA by latching, holding, and maneuvering the CRA during reactor startup, power operation, and shutdown in response to signals from the control rod drive power converter and controller assembly, and in releasing the CRA during a scram. The CRDS also includes the rod position indicator cabinets and cables, CRDM power cables, and cooling water supply and return piping inside containment. The mechanical design of the CRDM is described in Section 3.9.4 and the design of the CRA is described in Section 4.2.2. The instrumentation and controls for the CRDS are described in Section 7.0.4.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Figure 4.6-1 through Figure 4.6-5 illustrate the principal features of the CRDS. Figure 4.6-1 is a simplified drawing showing an overview of the location of the various components of the CRDS relative to the reactor pressure vessel (RPV) and the containment vessel (CNV). It includes the CRDMs and supports, CRA drive shafts, internal CRDS supports, and CRA guide tubes. The mechanisms are located on top of the RPV and laterally constrained at two elevations above in order to limit relative lateral seismic motion, yet allow for unrestricted axial expansion. The long CRA drive shafts are located inside the RPV, and aligned laterally by CRDS support structures that are part of the reactor vessel internals (RVI). Further details are provided in Section 3.9.4.1. The electromagnetic load transfer across the primary pressure boundary is facilitated by electromagnetic coils on the outside (Figure 4.6-3) that engage a set of magnetic poles connected to latches on the inside (Figure 4.6-5), in order to move the CRA drive shaft in a predetermined stepping sequence (refer to Section 3.9.4.1.2. Figure 4.6-2 provides an illustration of the CRDM electromagnetic coils and housings, including the pressure housings. The major components of the CRDM are annotated, and detailed in the subsequent figures. The power and cooling water connectors are located on top of the mast assembly and sensor coil for ease of access through the removable cover on top of the CNV (Figure 4.6-1). Figure 4.6-3 illustrates the CRDM drive coil and embedded cooling coils shown on the right view without the coil stack housings and mast assembly. The electrical connector on top of the left view is located above the cooling water fittings for separation purposes. Figure 4.6-4 shows the layout of the rod position indicator sensor coil assemblies in close contact with the rod travel housing. Rod position indication is facilitated by means of electromagnetic induction in the sensor coils, as the top of the CRA drive shaft travels upwards or downwards within the pressure boundary. Figure 4.6-5 provides an overview of the latch mechanism assembly (LMA), with the remote disconnect latch shown separately for better illustration. The three magnetic poles, latches and grippers on the left represent an industry-standard LMA design that performs the rod withdrawal/insertion/SCRAM functions, whereas the remote disconnect grippers (RDG) are relied upon during the remote disconnection/re-connection for NPM refueling only. Figure 4.6-6 illustrates the remote disconnection of the CRDM drive shaft from the CRA that is not available in the operating NPM location, in order to preclude inadvertent CRA disengagement.

The CRDM assembly is a hermetically sealed electro-mechanical device, which moves the CRA in and out of the reactor core, and holds the CRA at any elevation within the range of CRA travel. If electrical power is interrupted to the CRDM, the control rod drive shaft is released, and the attached CRA drops into the reactor core.

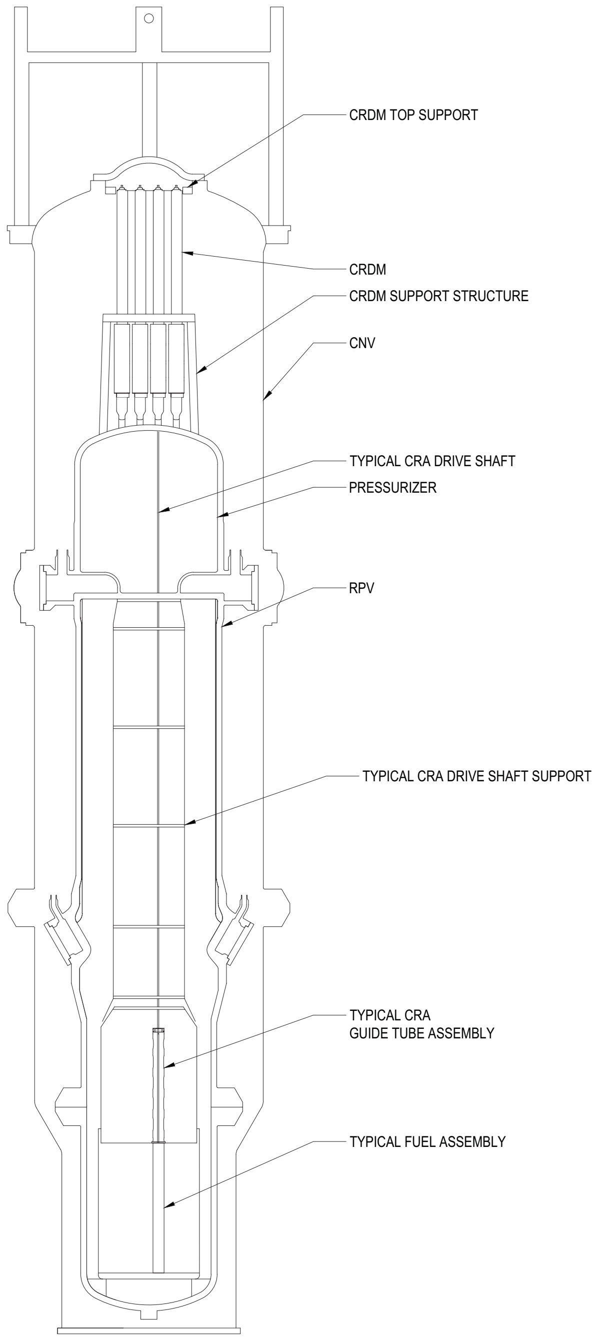
The CRDMs are mounted on the RPV head, and the CRDM pressure housings are safety-related American Society of Mechanical Engineers (ASME) Class 1 pressure boundaries. The CRDS components internal to the reactor coolant pressure boundary are designed to function in borated primary coolant with up to 2000 ppm boron at primary coolant pressures and temperatures ranging from ambient conditions to 650 degrees F design temperature and 2,100 psia RPV design pressure. During normal operating conditions the upper portion of the RPV and the CRDM pressure housing are in contact with saturated steam on the inside at 625 degrees F and 1850 psia. The lower portion of the drive rod is submerged in the primary coolant at hot leg temperature flowing upward through the upper riser and CRA guide tubes. The electric coil operating conditions require active cooling by water through a CRDS cooling water distribution header to cooling tubes in the drive coils of each CRDM as shown in Figure 4.6-3. The cooling requirements for the CRDMs are provided by the reactor component cooling water system (RCCWS) in Section 9.2.2.

The CRDS cooling line is branched into supply lines inside the containment vessel to each individual CRDM. After passing through the CRDM cooling tubes, the flexible return lines rejoin into a single return header leaving containment. A thermal relief valve is provided on the return header to provide overpressure protection for the CRDS cooling piping during a containment isolation event.

The structural materials of construction for the CRDS are discussed in detail in Section 4.5.1.

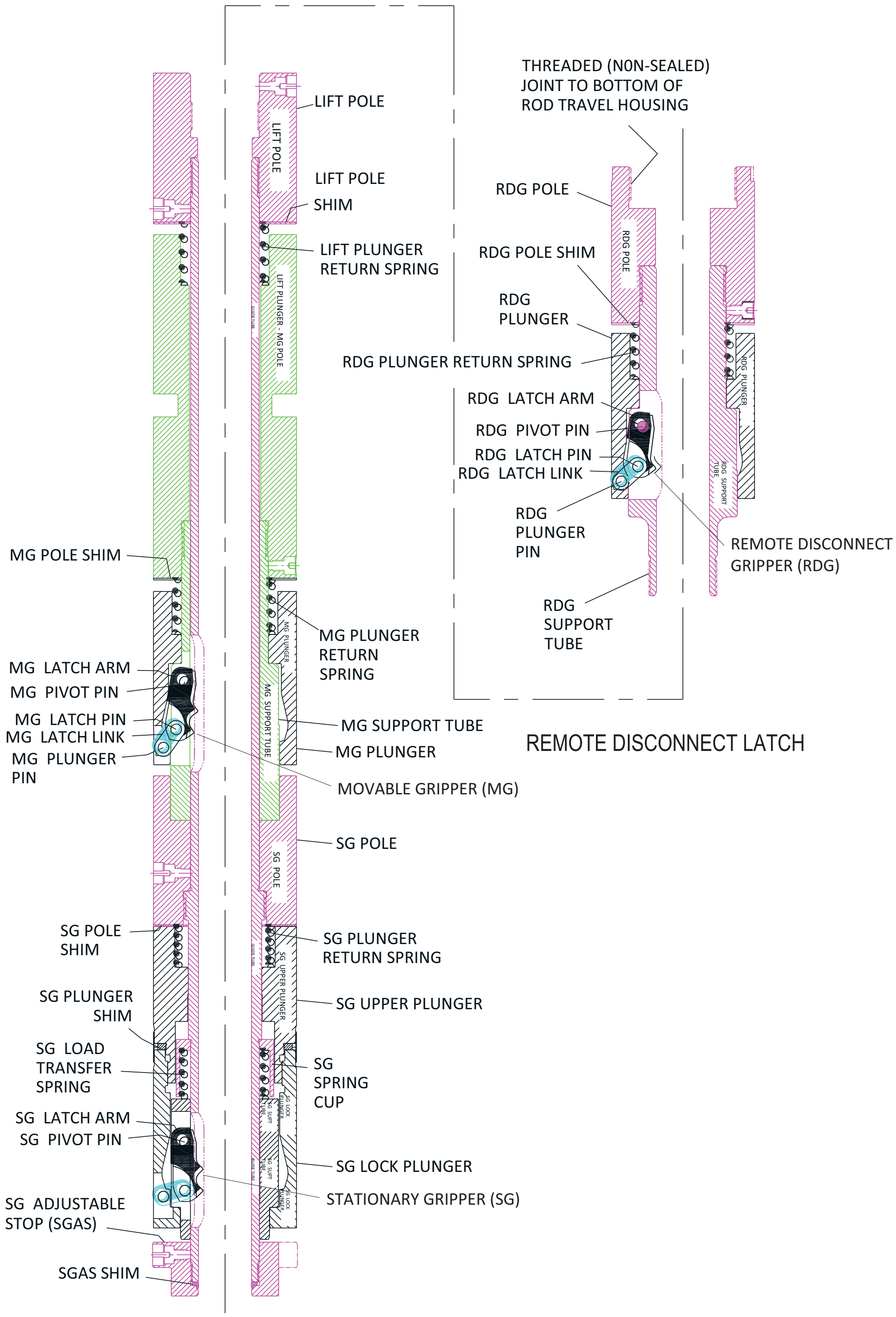
RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Figure 4.6-1: Overview of Control Rod Drive Mechanism Locations in Relation to the Reactor Pressure Vessel and Containment Vessel



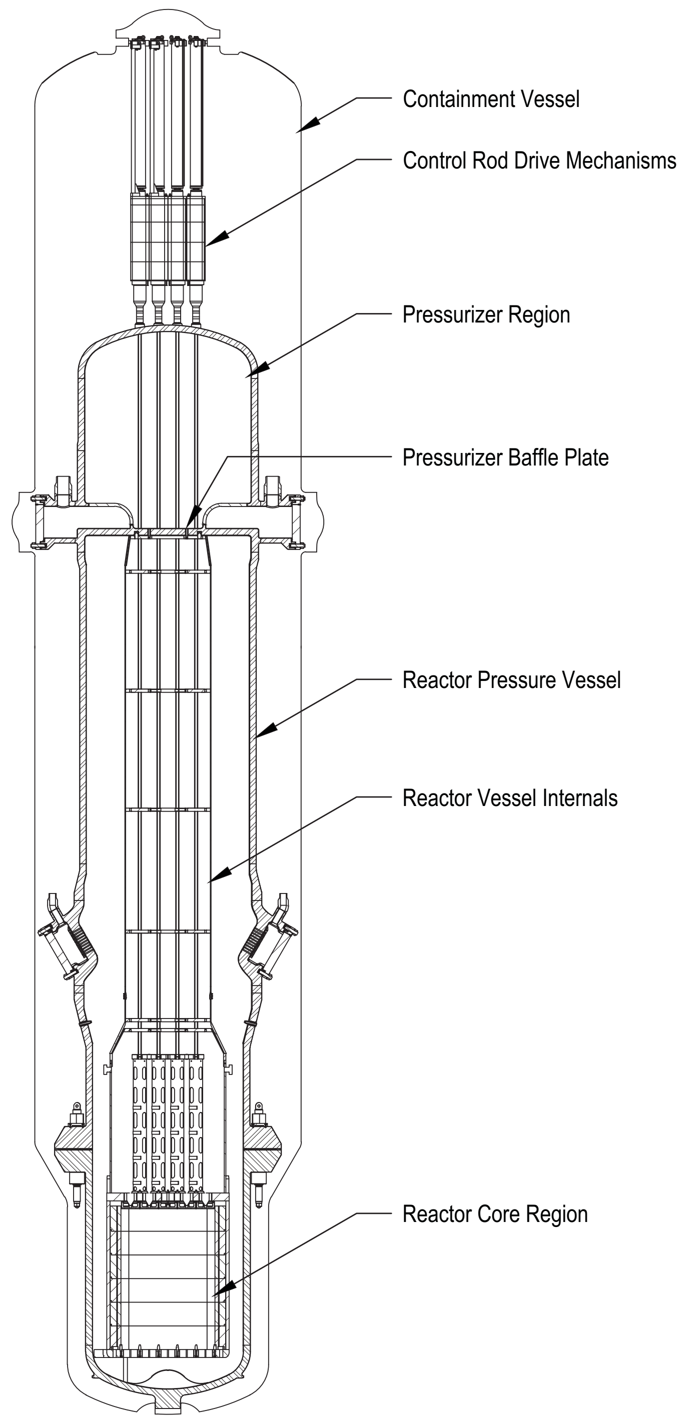
RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Figure 4.6-5: Overview of Latch Mechanism Assembly



RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Figure 5.1-1: NuScale Power Module Major Components



Enclosure 3:

Affidavit of Zackary W. Rad, AF-0817-55333

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.
2. I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:
   a. The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.
   b. The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.
   c. Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
   d. The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.
   e. The information requested to be withheld consists of patentable ideas.
3. Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects of NuScale reactivity control systems.

   NuScale has performed significant research and evaluation to develop a basis for this components and has invested significant resources, including the expenditure of a considerable sum of money.

   The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

   If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

4. The information sought to be withheld is in the enclosed Request for Additional Information No. 58, eRAI 8835. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
    a. The information sought to be withheld is owned and has been held in confidence by NuScale.
    b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
    c. The information is being transmitted to and received by the NRC in confidence.
    d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
    e. Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 8/7/2017.

Zackary W. Rad