RAIO-1117-57422

November 30, 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 Supplemental Response to NRC Request for Additional Information No. 58 (eRAI No. 8835) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 58 (eRAI No. 8835)," dated June 08, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 58 (eRAI No.8835)," dated August 07, 2017

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Questions from NRC eRAI No. 8835:

- 03.09.04-1
- 03.09.04-2
- 03.09.04-4

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

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

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Distribution: Gregory Cranston, NRC, OWFN-8G9A
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8835

Enclosure 1:

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

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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:

An update to FSAR Figure 4.6-2 was discussed in the response to eRAI 8835, Question 03.09.04-41 in RAIO-0817-55329, dated August 7, 2017 (ML17219A749). The updated Figure was inadvertently omitted from the FSAR update included with the eRAI 8835 response. FSAR Figure 4.6-2, included in this supplementary response provides a broader overview of the CRDM components. Additionally, FSAR Sections 1.5, 3.9.4, 4.2, 4.4 and 4.6 and associated Figures 3.9-1, 3.9-2, 4.6.1 and 4.6-2 are revised to consistently refer to the "control rod drive shaft." These changes were discussed during a September 13, 2017 public meeting.

Impact on DCA:

FSAR Sections 1.5, 3.9.4, 4.2, 4.4 and 4.6 and associated Figures 3.9-1, 3.9-2, 4.6.1 and 4.6-2 have been revised as described in the response above and as shown in the markup provided with the response to question 03.09.04-4.

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:

A description of the CRDM support structures was provided in the response to eRAI 8835, Question 03.09.04-4 in RAIO-0817-55329, dated August 7, 2017 (ML17219A749). This information has been added to FSAR Section 3.9.4.1 in response to an NRC request during a September 13, 2017 public meeting.

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 the response to question 03.09.04-4.

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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.: 03.09.04-4

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:

The remote disconnect mechanism (RDM) subsection added to the FSAR discussed in the response to eRAI 8835, Question 03.09.04-4 in RAIO-0817-55329, dated August 7, 2017 (ML17219A749), has been revised to address NRC comments provided during a September 13, 2017 public meeting. Figure 4.6-6 has been updated to more clearly depict operation of the remote disconnect mechanism. Additional detail has been added to Section 3.9.4.1.1 to describe the remote disconnect process, including discussion of RDM component operation and configuration during each step of remote disconnect operation. A similar detailed discussion of the remote engagement operation is also added to Section 3.9.4.1.1. Additionally, text discussing reactor trip operation that was removed by the FSAR change discussed by the response to eRAI 8835 has been restored. This change was also discussed during the August 7, 2017 public meeting.

Impact on DCA:

FSAR Section 1.5.1.7 and Figure 4.6-6 have been revised as described in the response above and as shown in the markup provided in this response.

This testing program is currently underway. Test results will characterize the response of the CRA fingers and rodlets due to single-phase primary flow-induced vibration (e.g., turbulent flow excitation) and will characterize the flow patterns and velocities at planes through CRA guide tube structure openings.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

1.5.1.12 Control Rod Assembly Drop and Control Rod Drive Shaft Alignment Test

The NPM is designed with control rod drive shafts that are longer than conventional PWR designs and have the capability to be remotely disconnected. The~~se~~ control rod drive shafts are aligned using the following multiple-support features:

- control rod drive mechanism nozzles in the reactor vessel head
- integrated steam plenum pressurizer baffle plate
- five upper control rod drive shaft supports in the upper riser section
- a control rod drive shaft alignment cone located at the top of the CRA guide tube

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

The design uses a CRA and fuel-assembly design similar to, but shorter than, traditional operating reactors. The arrangement of a shorter fuel assembly and CRA coupled to a longer control rod drive shaft creates a~~n~~ unique configuration of these components with no operational ~~experience~~ or testing experience. The CRA-drop and control rod drive shaft-alignment test program was developed to confirm the operability of this unique design.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

Testing is being performed at the AREVA Technical Center in Erlangen, Germany, and is divided into two main test configurations: an ambient pressure and temperature test, and an elevated temperature condition test. The ambient test configuration is composed of a full-length control rod drive shaft coupled with the NPM control rod assembly and fuel assembly, as well as the control rod drive shaft support structures and a CRA guide tube assembly. The CRA guide tube assembly and fuel assembly are joined in the water under ambient conditions with no coolant flow. During the test, the coupled CRA and control rod drive shaft assembly drops with variations in the alignment of the control rod drive shaft supports and CRA guide tube assembly and a mid-span deflection of the fuel assembly.

In the elevated-condition test configuration, the test specimen is composed of a partial-length control rod drive shaft, with additional weight added to simulate the total weight of the full-length control rod drive shaft. The CRDA is coupled with a CRA, fuel assembly, and a CRA guide tube assembly contained in a pressure vessel. During the test, the CRA drops with variations of CRA guide tube assembly misalignment configurations under nominal operation pressure and flow conditions.

This testing program is currently underway. Test results are used to confirm the operability of the control rod drive shafts for a range of potential component conditions and distortions.

(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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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 control rod 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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09-04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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 ASME 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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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~~control rod 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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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 control rod assembly (CRA) hub, external electromagnetic coils with cooling loop heat exchangers, the power/control system, and the rod position indication system. ~~It~~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 mechanism) 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. Figure 4.6-1 depicts the CRDM support structure and the CRDM support frame.

RAI 03.09.04-1, RAI 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

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 control rod drive shaft supports are depicted in FSAR Figure 3.9-1, Figure 3.9-2 and Figure 5.1.1. 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

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

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~~control rod 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.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

Control Rod Drive Shaft

The control 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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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 control rod drive shaft from the CRA, as the control rod 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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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 inch 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 control rod drive shaft when the teeth of the latch arms are engaged within the grooves in the upper segment of the control rod drive shaft.

RAI 03.09.04-1, RAI 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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 control rod 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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

Control Rod Remote Disconnect

The CRDM includes a remote disconnect mechanism (RDM) which performs the function of disconnecting the control rod drive shaft from the CRA. The disconnection occurs at the junction between the control rod drive shaft and the

CRA hub (as shown in Figure 4.6-6). The RDM enables the control rod 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 control rod drive shaft.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

The remote disconnect sequence is started when the CRA is inserted to the post SCRAM position where the CRA is fully inserted into the fuel. From this location the CRA is lifted, using the stepping process. At this elevation the remote disconnect gripper is actuated and holds the knob, on the top of the center disconnect rod, keeping it stationary. The stepping process will now move only the outer shaft with the center disconnect rod remaining stationary. The CRA is then lowered (which only moves the outer shaft) sliding the fingers off of the stationary coupling expansion plug on the center disconnect rod. Specifically, the plug on the center disconnect rod is pulled from between the fingers as the control rod drive shaft is lowered. This allows the fingers to spring inward to let the CRA hub separate, and the CRA fall the short distance back to the fully inserted position. The hydraulic snubbers, internal to the fuel assembly, minimize the impact force. After separation the control rod drive shaft is moved up to where the remote disconnect gripper was activated. The remote disconnect gripper is released leaving the plug on the center disconnect rod in the normal locked position. The center and outer shafts will again move as one unit. The control rod drive shaft must be lowered to set on the top of the CRA hub to prevent impact when the power is disconnected. Therefore, the plug on the center disconnect rod must be in the fully inserted (locked) position so the coupling cannot reconnect. This completes the disconnect sequence. A lift verification is performed by observing the difference in lift coil current confirming successful completion of the remote disconnect operation.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.0904-4S1

Control Rod Reconnect

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 (i.e., reconnect) may begin. The control rod drive shaft is reconnected to the CRA hub on the CRA by lowering the outer shaft to retract the plug on the bottom of the center disconnect rod. Utilizing the stepping process, the control rod drive shaft is lowered into the CRA hub (with the plug on the center disconnect rod extracted). The control rod drive shaft is then lowered an extra step to make sure the fingers on the coupling are fully inserted into the CRA hub. The next step is to insert the plug on the center disconnect rod to expand the fingers and lock them in place. To assure the plug on the center disconnect rod completely inserts, the remote disconnect gripper releases the center disconnect rod and lets it fall, along with the spring assist.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

To initiate the reconnect sequence, the control rod shaft is raised above the post-SCRAM position. This is the same position from which the disconnect sequence was started. The disconnect gripper is activated to make the center disconnect rod stationary, to remove the plug on the center disconnect rod from the locked position. The stepping process now only moves the outer shaft. The control rod drive shaft is lowered, which retracts the plug on the center disconnect rod from the fingers on the coupling, and inserts the fingers into the CRA hub. The additional step compresses the spring in the CRA hub slightly, and assures the fingers on the coupling are completely seated. The plug on the bottom of the center disconnect rod is inserted by releasing the remote disconnect gripper. The center disconnect rod then falls, with spring assist, to lock the control rod drive shaft to the CRA hub. A lift verification is then performed by observing the difference in lift coil current confirming successful completion of the remote reconnect operation.

RAI 03.09.04-1, RAI 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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 control rod 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 travel housing plug is broken, and the plug removed for tooling access. The top of the control rod 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.

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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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/control rod drive shaft interface always passes through the in-clear position. This minimizes wear at the latch/control rod drive 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.

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.

RAI 05.02.03-16

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 threaded connections are ~~bolting is~~ designed in accordance with the ASME BPVC Code, Section III~~, as addressed in Section 3.13~~. The threaded connections in the CRDM pressure housing sections use acme threads, and canopy welds as the pressure seals. The CRDM threaded joint configurations are provided in Figure 4.6-4. Additional information on compliance with codes and code cases for the RCPB is provided in Section 5.2.1.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1, 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 control rod 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

above the top of the riser and below the PZR baffle plate, and then flow downward through the annular space outside of the riser and inside of the RPV where the SG helical tube bundles are located.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1, RAI 03.09.05-4, RAI 03.09.05-12

The upper riser assembly includes the upper riser, a series of ~~CRA~~control rod drive shaft and ICI guide tube supports referred to as upper CRDS supports, and the upper riser hanger assembly. The upper riser assembly also accepts and positions the RCS injection piping. The ICI guide tubes, which are supported by the upper riser assembly, extend from their respective penetrations in the RPV top head downward through the PZR space, the upper riser, and the lower riser to their respective fuel assemblies. The portion of the ICI guide tubes extending from the RPV upper head penetrations to the bottom of the upper riser assembly is depicted in Figure 3.9-2. The upper riser assembly hangs from the pressurizer baffle plate. ~~A small vertical clearance is provided between the upper riser and the lower riser to accommodate thermal growth in the vertical direction. In addition, there is a bellows assembly in the lower portion of the upper riser (see Figure 3.9-2) to provide added flexibility in the vertical direction to accommodate circumstances that involve sufficient thermal growth to close the vertical gap between the upper and lower riser assemblies.~~There is a bellows assembly in the lower portion of the upper riser (see Figure 3.9-2). This bellows assembly exerts an initial contact load, in the cold condition, on the lower riser interface, and then allows for the vertical thermal expansion. The RVI materials including base materials and weld filler materials are discussed in Section 4.5.2 and are designed to minimize the number of welds and bolted interfaces within the high neutron flux regions.

During refueling and maintenance outages the upper riser assembly stays attached to the upper section of the NPM (upper CNV, upper RPV and SG) while providing physical access for potential inspection of the feedwater plenums, SG, RPV and control rod drive shaft supports. The lower riser assembly and CSA remain with the lower NPM (lower CNV, lower RPV, core barrel, and core plates) when the module is parted for refueling and maintenance.

The RVI upper riser assembly is supported from the RPV integral steam plenum (e.g., below the bottom of the PZR).

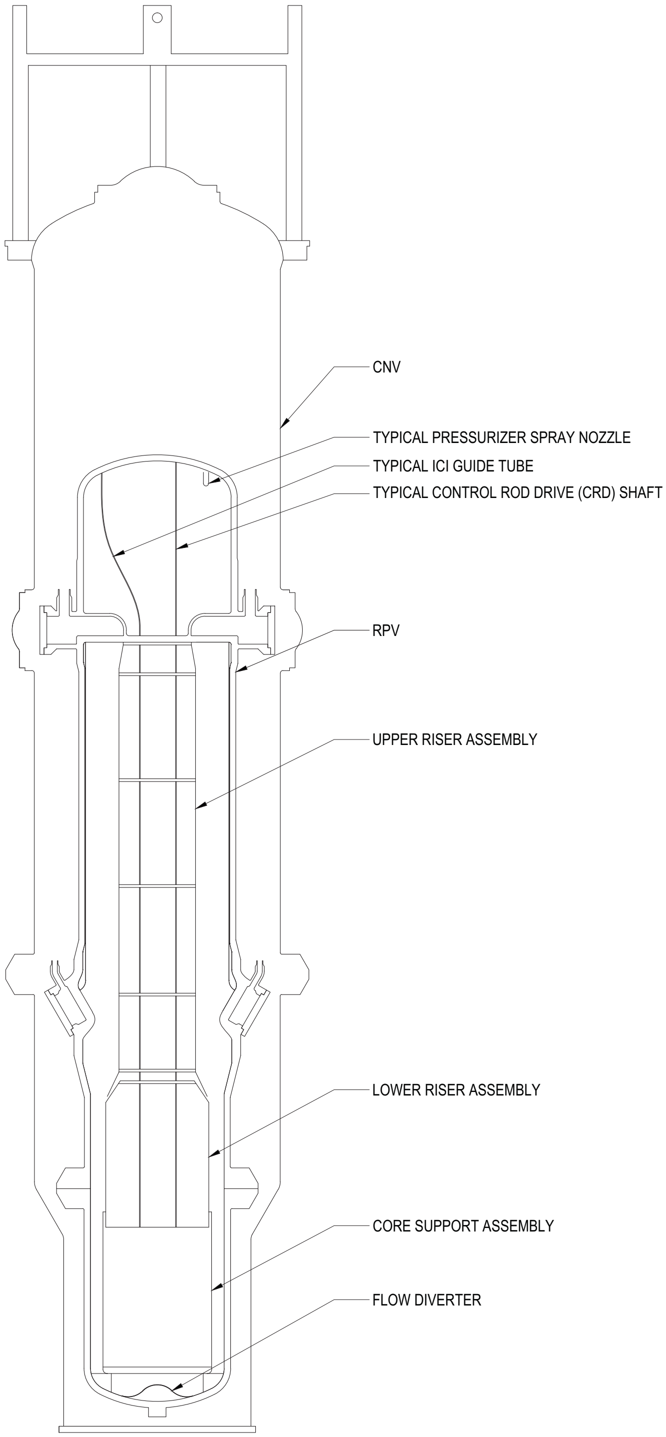
RAI 03.09.05-4, RAI 03.09.05-12

Under normal operation, the reactor core is supported by the core support structures of the CSA (seismic Belleville washers, core support blocks, core barrel, lower core plate and upper core plate) that surround the fuel assemblies. The deadweight and other mechanical and hydraulic loads from the fuel are transferred to the upper and lower core ~~support~~ plates. The motion of the upper and lower core ~~support~~ plates is coupled through the core barrel. Under seismic and other accident conditions, the core barrel transfers lateral loads to the RPV shell through the core support blocks at the bottom of the RPV and the upper support blocks that are attached to the upper portion of the core barrel. The vertical loads are transferred from the core barrel to the RPV head through the seismic Belleville washers and core support blocks.

The fuel is surrounded by a heavy neutron reflector made of reflector blocks stacked on top of each other. The heavy reflector reflects neutrons back into the core to improve

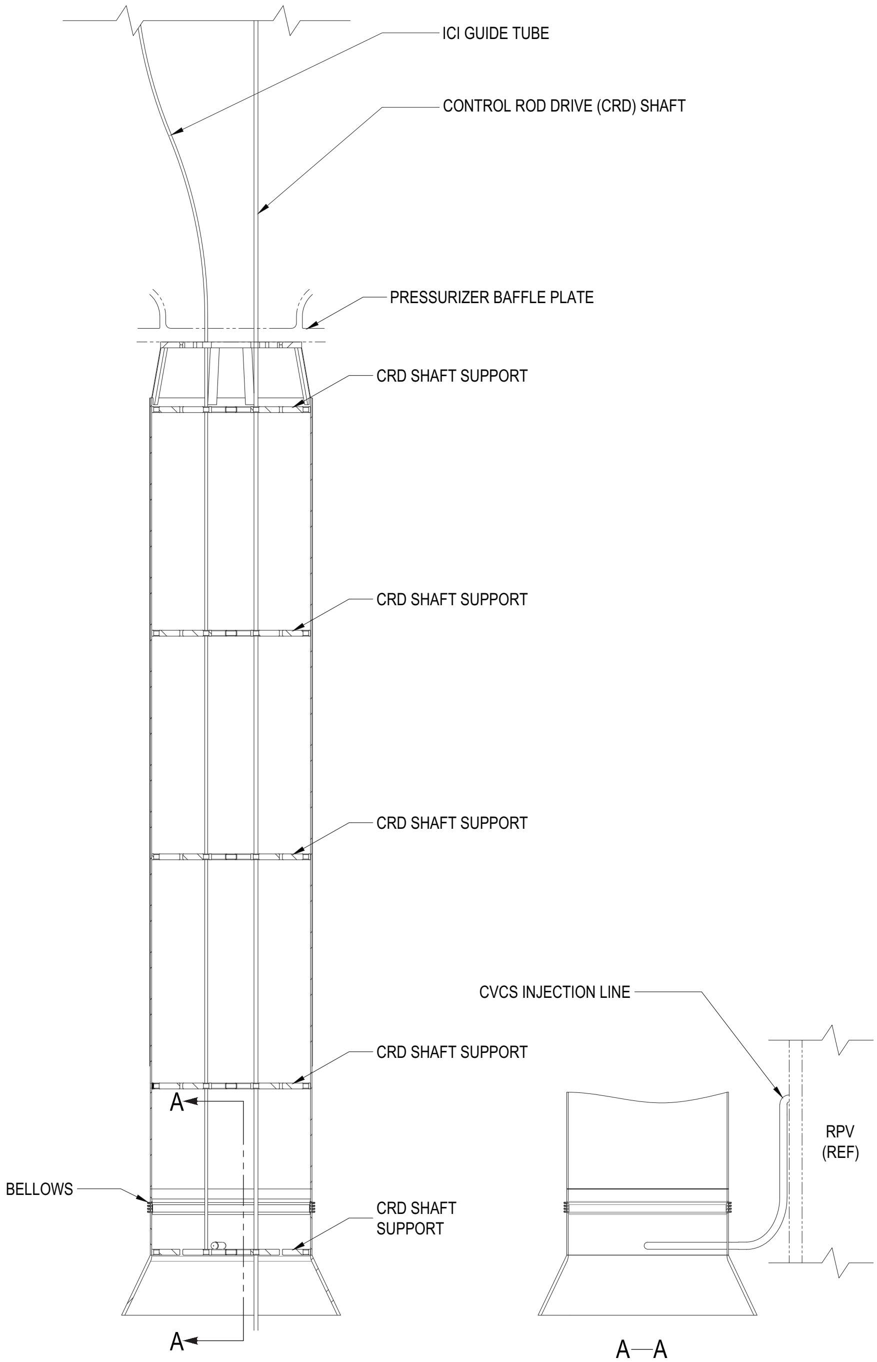
RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

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



RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

Figure 3.9-2: Upper Riser Assembly



around the peak rod, which minimizes energy transfer out of the channel containing the peak rod. This is described in more detail in Reference 4.4-3.

An adjustment to the bounding radial power distribution also includes penalties for $F_{\Delta H}$ measurement and engineering uncertainty on the hot rod. These penalties include the measurement uncertainty on the radial peaking as well as the engineering uncertainty for enthalpy rise. The radial power distribution prior to applying uncertainty penalties retains the conservative peak-to-average for the hot assembly, while the rod in which MCHFR is determined accounts for the applicable uncertainties.

The CHF limiting axial power shape based on core-average axial power is sufficient to be used for most transient analyses. Generally, the core-average axial power shape does not deviate significantly from the spectrum of shapes already considered within the power shapes analysis, and the subchannel limiting axial power shape is held constant during these events. The combination of the core-average axial power shape of initiating power level with the conservative radial power distribution and core hydraulic boundary conditions from NRELAP5 provides a conservative MCHFR calculation. For events where the axial flux shape changes, a specific analysis is performed to determine the axial flux shape that is conservative for the event.

4.4.4.4 Core Thermal Response

The core thermal response during AOOs, IEs, and accidents is presented in Chapter 15.

Low power and shutdown operation is described in Section 19.1.6 and the probabilistic risk assessment for the operation is addressed. The NPM natural circulation design does not require mid-loop operation during shutdown conditions. The core is always submerged in a pool of water so the core is not subjected to mid-loop thermal-hydraulic conditions during refueling operations.

4.4.4.5 Analytical Methods

4.4.4.5.1 Reactor Coolant System Flow Determination

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

The reactor coolant system (RCS) flow loop is comprised of the fuel assembly region, core bypass region, upper core plate, CRA guide tubes, CRA guide tube support plate, riser transition, ~~CRA~~control rod drive shaft support, upper riser turn to annulus, steam generator, downcomer transition, downcomer to lower plenum turn, and the lower core plate. The volumes, flow areas, and volume lengths used to perform the flow calculation are provided in Table 4.4-1. The driving force for flow is the buoyancy arising from the density differences around the RCS flow loop. The primary contributors to pressure loss in the system are the fuel assembly and steam generator regions. These pressure losses are confirmed by testing. The remaining pressure drops are determined analytically. The steady state flow is

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 03.09.04-1S1, RAI 03.09.04-2, RAI 03.09.04-2S1, RAI 03.09.04-4, RAI 03.09.04-4S1, 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~~control rod drive shafts, internal CRDS supports, and CRA guide tubes. The CRDMs 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 control rod 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 control rod 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 which are located directly above the rod travel housing. Rod position indication is facilitated by means of electromagnetic induction in the sensor coils, as the top of the control rod 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~~control rod 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.

4.6.2 Evaluations of the Control Rod Drive System

This section describes how the design of the CRDS conforms to General Design Criteria (GDC) 4, 23, 25, 26, 28, 29 of 10 CFR 50, Appendix A. The design also conforms to Principal Design Criteria (PDC) 27.

RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

GDC 4 is applicable to the CRDS design as it requires the structures, systems, and components important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions during normal plant operation as well as during postulated accidents as a result of equipment failures and external events. The CRDS provides the capability to safely shut down the reactor during normal operations and AOOs and either prevents or mitigates the consequences associated with postulated accident scenarios. The CRDS design features comply with GDC 4 requirement for designing the CRDS to be compatible with the environmental conditions. The CRDS components located inside the containment are protected against dynamic effects as described in Section 3.6. The CRDS structures, systems, and components are located inside the Reactor Building, which is a Seismic Category I structure designed to protect from events and conditions outside the NuScale Power Plant. The CRDS ability to perform the required safety-related functions will not be compromised by adverse environmental conditions. The control rod drive shafts are immersed in 590 degrees F water during normal full power operation. The upper portion of the control rod drive shafts penetrate the pressurizer and are exposed to a steam environment at about 625 degrees F. The control rod drive shafts and latch mechanisms are designed to 650 degrees F and are able to operate without the typical liquid drag forces experienced by CRDMs in the current PWR fleet.

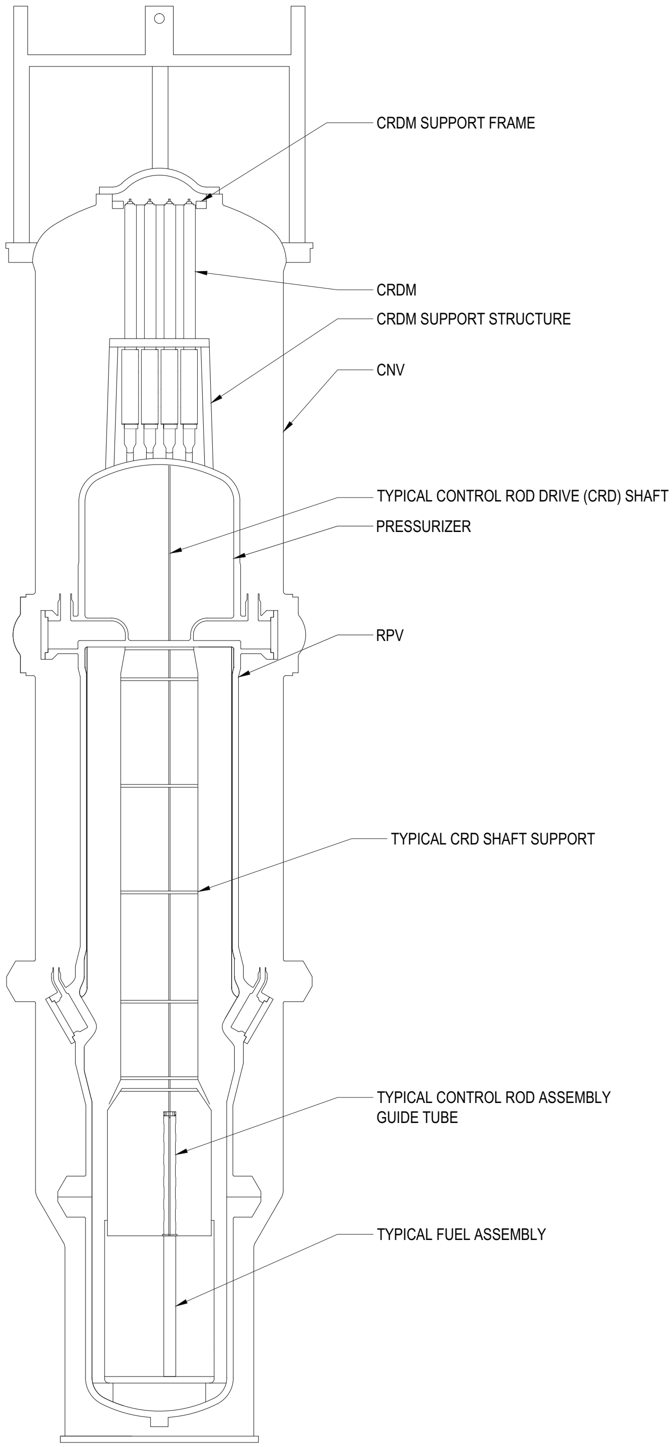
GDC 23 requires that the protection system be designed to fail into a safe state in the event of adverse conditions or environments. The CRDM provides positive core reactivity control through the use of movable CRAs. The movable CRAs provide reactivity control for all modes of operation, including all plant conditions from the cold shutdown condition to the full-load condition. The CRDM, in conjunction with the module protection system, actuate the control rods to perform safety-related functions when necessary to provide core protection during normal operation, AOOs, and accidents. The CRDM is designed to fail in a safe condition, even under adverse conditions, that prevents damage to the fuel cladding and excessive reactivity changes during failure. Loss of electrical power to the reactor trip breaker will initiate a reactor scram, causing rods to drop into the core to shut down the reactor.

GDC 25 requires that the protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems. Chapter 15 safety analyses demonstrate that the CRDS with any assumed credible failure of any single active component is capable of performing a reactor trip when plant parameters exceed the reactor trip setpoint, in accordance with GDC 25.

GDC 26 is applicable to the CRDS design, as the CRDS is one of the independent reactivity control systems. It 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) via gravity-

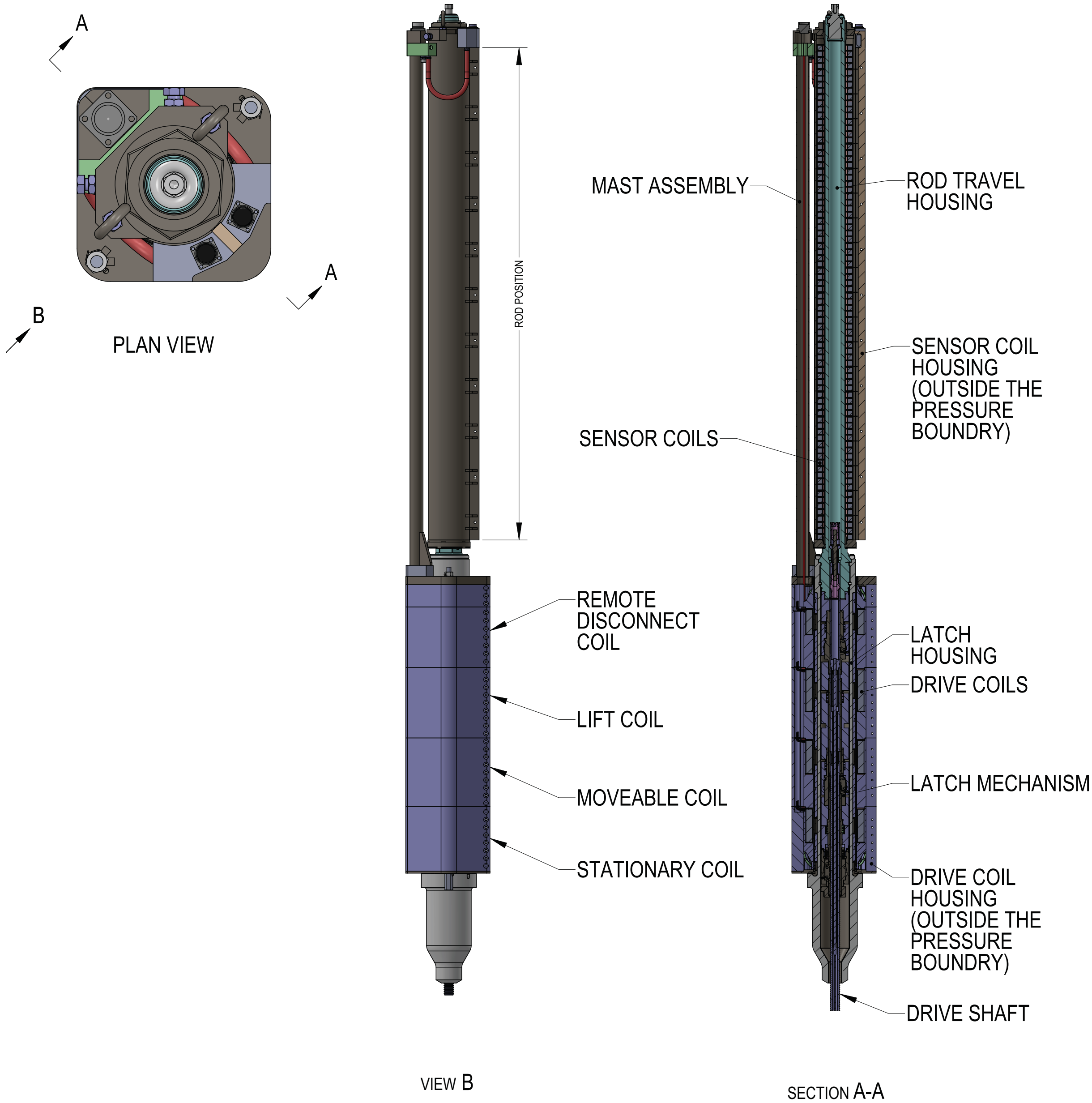
RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

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



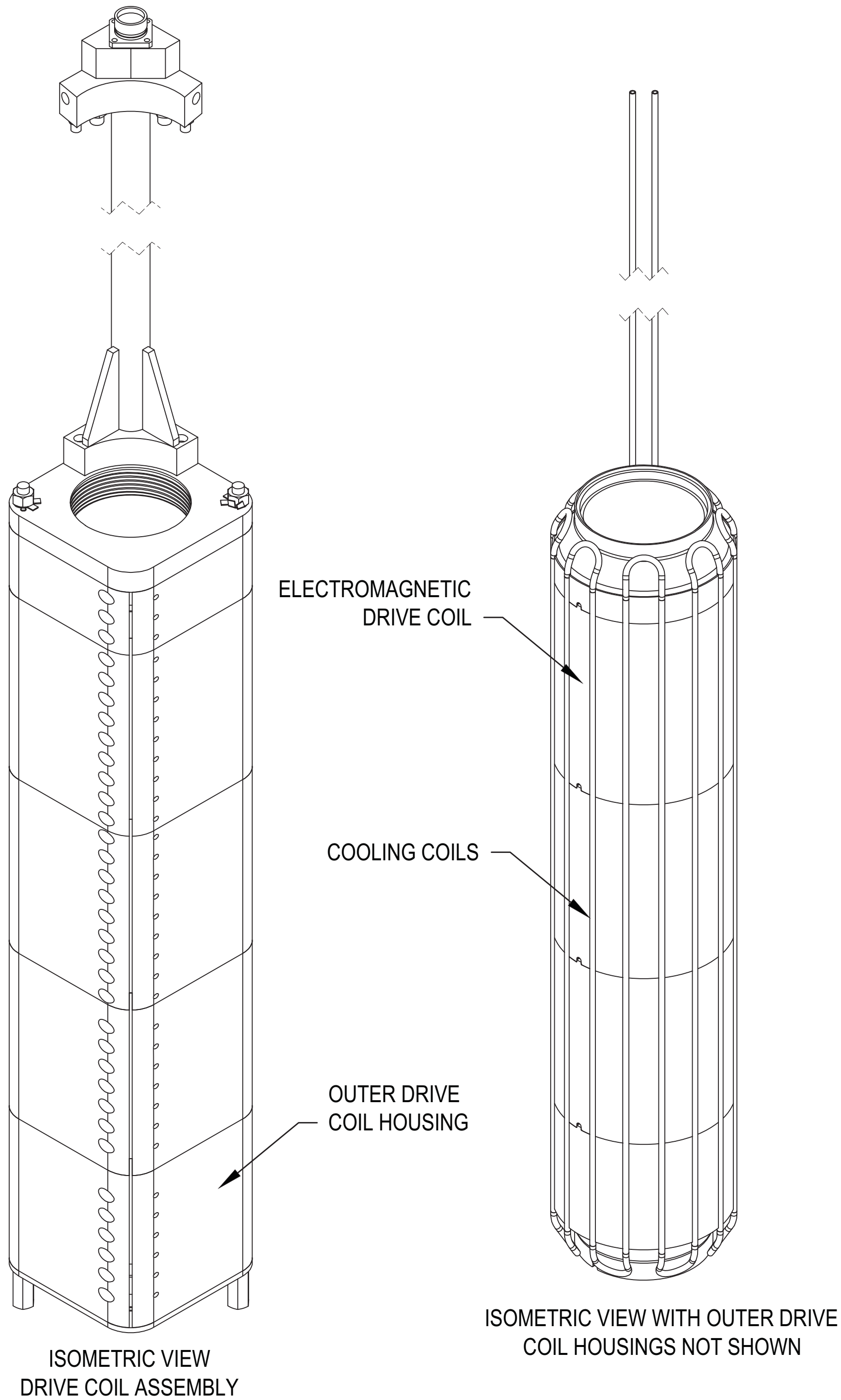
RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

Figure 4.6-2: Control Rod Drive Mechanism Coils and Housings



RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

Figure 4.6-3: Control Rod Drive Mechanism Drive Coil and Cooling Detail



RAI 03.09.04-1S1, RAI 03.09.04-2S1, RAI 03.09.04-4S1

Figure 4.6-6: Control Rod Drive Mechanism Drive Shaft Interface with Control Rod Assembly