RAIO-0418-59518

April 13, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 374 (eRAI No. 9389) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 374 (eRAI No. 9389)," dated February 28, 2018

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

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9389:

- 03.09.04-12

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,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9389

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9389

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

eRAI No.: 9389
Date of RAI Issue: 02/28/2018

NRC Question No.: 03.09.04-12

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 shaft 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 this SSC be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed

In previous RAIs 8835, Question 03.09.04-8 and 9181, Question 03.09.04-10, the applicant has established that the control rod drive shaft will be considered an ASME BPV Code, Subsection NG internal structure. As internal structures have no requirement beyond the Certificate Holder certifying that the internal structure shall not adversely affect the integrity of the core support structure, unless specifically stipulated, the applicant specified the following:

- CRD shaft scram loads and control rod drive shaft deflection limits will be established by testing. CRD shaft scram and SEE loads are added to FSAR Table 3.9-6 for Service Level A and D loading combinations for control rod drive shafts.
- The control rod drive shafts are evaluated against the limits of NG-3222.1 and NG-3222.2 for normal operating (Service Level A) conditions. Service Level A loads for the control rod drive shafts are the deadweight of the control rod assembly and scram loading.
- The control rod drive shafts are evaluated against 110% of the limits of NG-3222.1 and NG- 3222.2 for Service Level D loads. Consideration of cyclic loading is not required.
- Martensitic stainless steel materials used in the control rod drive shafts shall be Cv tested in accordance with NG-2331.

The above information should be incorporated into the DCD, as well as a pointer provided to the additional requirements for control rod drive system materials located in Section 4.5.1 of the DCD. This information supports GDC 1 as mentioned above

Also, in the response to 03.09.04-8, "BPCV" should be "BPVC," and in the response to

03.09.04-10, “SEE” should be “SSE” in the first bullet point.

Finally, in the response to 03.09.04-10, the addition to FSAR Table 3.9-6 (SCRAM) requires a corresponding addition to FSAR Table 3.9-2, as the term SCRAM is currently undefined.

NuScale Response:

FSAR Section 3.9.4 has been updated to include the control rod drive shaft design requirements identified in the response to eRAI 9181, Question 03.09.04-10, transmitted in NuScale letter RAIO-1217-57607, dated December 11, 2017 (ML17345A943). In addition, Table 3.9-2 has been updated to define SCRAM loads. Finally, Section 3.9.4.1.1 is updated to correct the acronym for the ASME Boiler and Pressure Vessel Code: "BPVC" versus "BPCV". Note: since the acronym for safe shutdown earthquake, SSE is correctly used in Section 3.9.4, and Section 3.9.4 refers to Section 4.5.1 for CRDS materials, no changes are required for these items.

Impact on DCA:

FSAR Section 3.9.4 and Table 3.9-2 have been revised as described in the response above and as shown in the markup provided in this response.

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.

RAI 03.09.04-12

Table 3.9-16 includes control rod drive shaft scram and SSE loads for Service Level A and D loading combinations. The control rod drive shafts are evaluated against the limits of NG-3222.1 and NG-3222.2 for Service Level A conditions. The Service Level A loads that are evaluated are the deadweight of the control rod assembly and scram loading. The control rod drive shafts are evaluated against 110 percent of the limits of NG-3222.1 and NG-3222.2 for Service Level D loads. Consideration of cyclic loading is not required.

Martensitic stainless steel materials used in the control rod drive shafts are Cv tested in accordance with NG-2331.

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 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, 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 designed in accordance with the ASME BPVC Code, Section III. 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, RAI 03.09.04-12

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~~BPVC, 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:

RAI 03.09.03-1, RAI 03.09.04-12

Table 3.9-2: Pressure, Mechanical, and Thermal Loads

Load	Description
P	Operating pressure[1]
P_{des}	Design pressure[2]
PD	Operating pressure difference[3]
PD_{des}	Design pressure difference
DW	Deadweight
B	Buoyancy
TH	Transient loads[4]
R	Steam generator tube failure
REA	Rod ejection accident
EXT	Mechanical loads other than piping such as RPV and CNV support reactions, RVI and CNV interface loads, scram loads, fuel assembly weights, and nozzle loads
M	Piping mechanical and thermal loads
MSPB	Main steam pipe break
FWPB	Feedwater pipe break
DBPB[5]	Design basis pipe break other than FWPB and MSPB
RSV	Reactor safety valve actuation
ECCS	Emergency core cooling system actuation
SSE	Safe shutdown earthquake
OBE	Operating basis earthquake
L	Lifting and handling
LL	Live load
LT	Load test
TR	Transportation
H	Hydrostatic test
P_{g1}	Hydrogen detonation
P_{g2}	Hydrogen detonation with deflagration-to-detonation transition
SCRAM	Reactor trip

Notes:

1. Operating pressure, "P," is the highest pressure during an applicable transient.
2. As used for ASME Code stress analysis, design pressure is specified as a gage pressure in accordance with NB-3112.1(b) giving consideration for operation of the RPV with a vacuum on the CNV or pressure testing of the CNV conservatively assuming a vacuum internal to the RPV.
3. Operating pressure difference, "DP," is the highest pressure difference during an applicable transient and may be internal or external.
4. Transient loads include transient thermal loads, as well as other transient loads, such as rapid pressure fluctuations.
5. DBPB includes CVCS pipe break and spurious valve actuation of the RVV, RRV and RSV CVCS pipe break includes DBPB for RPV high point degasification, PRZ spray, RCS discharge and RCS injection piping inside of containment.