February 07, 2019

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. 247 (eRAI No. 9132) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 247 (eRAI No. 9132)," dated September 29, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 247 (eRAI No.9132)," dated November 21, 2017
3. NuScale Power, LLC Response to NRC "Request for Additional Information No. 247 (eRAI No. 9132)," dated May 24, 2018

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 Question from NRC eRAI No. 9132:

- 14.03.03-3

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 Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Cayetano Santos, NRC, OWFN-8H12

Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9132

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9132

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

eRAI No.: 9132

Date of RAI Issue: 09/29/2017

NRC Question No.: 14.03.03-3

The NRC regulations in 10 CFR 52.47(b)(1) require that a design certification application contain the inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. The ITAAC proposed in the NuScale FSAR Tier 1, Section 2.1, "NuScale Power Module," to verify the capability of safety-related valves in the NuScale Power Module (NPM) for the NuScale Power Plant to perform their safety functions are not sufficient to satisfy 10 CFR 52.47(b)(1). The NRC staff understands that the ITAAC for the qualification of safety-related valves in the NuScale Power Plant will be provided in NuScale FSAR Tier 1, Section 2.8, "Equipment Qualification." In light of this understanding, the applicant does not need to address the qualification of the NPM safety-related valves in the ITAAC in NuScale FSAR Tier 1, Section 2.1. However, the NRC staff requests that the NuScale design certification applicant address the following aspects regarding the proposed ITAAC for the NPM safety-related valves:

a. NuScale FSAR Tier 1, Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," specifies in ITAAC #13, 14, 15, 16, 17, and 21 that safety-related valves in the reactor coolant system (RCS), containment isolation system (CNTS), emergency core cooling system (ECCS), and decay heat removal system (DHRS) change position under design differential pressure and/or flow conditions depending on the specific ITAAC. These proposed ITAAC in NuScale FSAR Tier 1, Table 2.1-4 are not sufficient to verify the capability of the NPM safety-related valves during preoperational testing. For example, the Design Commitment should specify that the NPM safety-related valves change position under design-basis temperature, differential pressure, and flow conditions.

Consistent with the staff's draft standardized ITAAC (e.g., see April 8, 2016, transmittal letter providing draft ITAAC, ML16096A121), the inspections, tests, and analysis (ITA) should specify that a diagnostic stroke test will be performed of the NPM safety-related valves under preoperational temperature, differential pressure, and flow conditions. The Acceptance Criteria should specify that each NPM safety-related valve listed in the applicable ITAAC table strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions with sufficient diagnostic data to correlate valve performance to its design-basis capability as established by the type test performed in accordance with the applicable functional qualification ITAAC. As discussed above, the NRC staff requests that the NuScale design certification applicant revise the proposed ITAAC to verify the capability of the NPM safety-related valves during preoperational testing.

b. NuScale FSAR Tier 1, Table 2.1-4 proposes ITAAC #18, 19 and 20 to verify the loss of motive power capability of CNTS, ECCS, and DHRS safety-related valves, respectively. The ITAAC need to verify that each applicable system safety-related valve will perform its function to fail to (or maintain) its safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions sufficient to correlate valve performance to its design-basis capability as established by the type test performed in accordance with the applicable functional qualification ITAAC. The NRC has prepared standardized ITAAC for verification of the loss of motive power capability for safety-related valves in new reactors (e.g., see April 8, 2016, transmittal letter providing draft ITAAC, ADAMS Accession No. ML16096A121). For example, the standardized Design Commitment specifies that the applicable system's safety-related valves will perform their function to fail to (or maintain) their safety-related position on loss of motive power under design-basis temperature, differential pressure, and flow conditions. The standardized ITA specifies that a stroke test will be performed of these safety-related valves under preoperational temperature, differential pressure and flow conditions. The standardized Acceptance Criteria specify that each applicable system safety-related valve listed in the ITAAC table performs its function to fail to (or maintain) its safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions sufficient to correlate valve performance to its design-basis capability as established by the type test performed in accordance with the applicable functional qualification ITAAC. The NRC staff requests that the NuScale design certification applicant discuss the intent of its proposed ITAAC in comparison to the standardized ITAAC or revise the proposed ITAAC

to verify the capability of the NPM safety-related valves during pre-operational testing.

NuScale Response:

NRC and NuScale held a clarification call on November 1, 2018 regarding the proposed Supplement 2 response. The NRC had asked whether correlation of preoperational test performance to design basis conditions was to be performed under QME-1 or an alternative means. NuScale explained that its safety-related valves are committed to qualification under QME-1. NuScale uses qualification testing to meet QV-7400, and at the same time, to determine some of the test parameter values for performance assessment testing. NuScale has committed to use preservice performance assessment testing to meet both the requirements of OM ISTC Code and QME-1 AV-7463 (demonstration of functional capability of production valve assemblies).

NuScale revised Tier 2, Section 3.9.6.3.2 on Valve Testing. This approach eliminates the need for an additional ITAAC to correlate preoperational test performance to design basis conditions. The addition in Section 3.9.6.3.2 to QME-1 and OM ISTC are operational programs that are required to be completed prior to fuel load.

Impact on DCA:

Tier 2, Section 3.9 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16

Valves that operate during normal plant operation at a frequency that satisfies the exercising requirement do not have to have an additional exercise test provided that the observations (and measurements) required of inservice testing are made and recorded at the required frequency specified by ISTC and that fail-safe requirements have been met.

RAI 14.03.03-3S2, RAI 14.03.03-4S2

Power-Operated Valve Skid Mounted Components - Safety-related HOVs are hydraulically powered from a common central hydraulic power unit (CHPU). There are two hydraulic skids per NPM that provide hydraulic power to all safety-related HOVs. These safety-related HOVs include PSCIVs, MSIVs (and bypass), FWIVs, and DHRS actuation valves. Valve actuator subcomponents are located on the CHPUs and are treated as skid mounted components. The components which support the HOV closing function are those in the hydraulic vent path. This includes solenoid valves, dump valves (MSIV and DHR valves only), and a hydraulic relief valve. These subcomponents meet the criteria of ISTC-1200(b) and are tested as part of each valve exercise test. Additional testing and maintenance on pressure relief device subcomponents is performed as part of performance assessment testing of HOVs.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, RAI 03.09.06-16S1, RAI 14.03.03-3S2, RAI 14.03.03.4S2

Power-Operated Valve ~~(POV)~~ Performance Assessment Tests - Lessons learned from operating experience at nuclear power plants were used in developing the NuScale design. The results are a simplified design that relies on passive safety systems and far fewer components than in a typical inservice testing plan. The active safety functions of the highly safety significant valves in the NuScale inservice test plan include containment isolation and emergency core cooling.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, 03.09.06-16S1

High safety significant POV groups for the NuScale Power Plant include ECCS reactor recirculation valves and the reactor vent valves, and certain small actuator containment isolation HOVs. Risk significant components are identified pursuant to Section 19.1, Probability Risk Assessment, which evaluates the NuScale Power Module for full power, low power, and shutdown modes of operation for both internal and external events.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, RAI 03.09.06-16S1

Performance assessment testing to ensure that NuScale power-operated valves perform their intended safety function(s) when called upon shall consider NRC RIS 2000-03 and OM Mandatory Appendix IV (OM-2017). The requirements for OM Mandatory Appendix IV are applied to both AOVs and HOVs. Lessons learned and recommendations from the AOV Joint Owners Group are considered in the development of the specific on-site performance assessment test procedures for all NuScale POVs.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, RAI 03.09.06-16S1

NuScale HOV performance assessment testing will contain the following attributes:

RAI 03.09.06-16S1

- fail safe, exercise test, and stroke time measurement

RAI 03.09.06-16S1

- verifying of the integrity of the nitrogen cylinder via visual inspection
- recording of as-found and as-left nitrogen pressure and temperatures when performing stroke time measurements
- comparing of nitrogen pressure and temperature with the previous valve tests to determine cylinder leakage rate over the test period

RAI 14.03.03-3S2, RAI 14.03.03-4S2

- testing the two redundant, fail-safe hydraulic vent paths on each valve ~~separately~~ to ensure that each vent path is fully functional

RAI 14.03.03-3S2, RAI 14.03.03-4S2

  - flow device downstream of each solenoid valve verifies both safety-related solenoid valves open on valve stroke to safe position

RAI 14.03.03-3S2, RAI 14.03.03-4S2

- periodic inspection and replacement of subcomponent relief devices
  - thermal relief check valve (PSCIV only), Appendix J Type C testing of the inboard PSCIV leak tests the thermal relief check valve
  - gas bottle relief valve
  - actuator housing relief valve
  - hydraulic line relief valve
- measuring and trending obturator torque periodically to verify and monitor valve friction degradation

RAI 03.09.06-16S1

- leakage testing, as required (Appendix J Type B, Type C, DHR boundary)

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16

NuScale ECCS valve performance assessment testing contains the following attributes:

RAI 03.09.06-16S1

- fail safe, exercise test, and stroke time measurement during NuScale Power Module (NPM) shutdown

RAI 03.09.06-16S1

- testing or inspection to ensure minimum flow capacity Cv(min) is confirmed
- testing of the inadvertent block valve function
- testing of any ECCS valve not opened during exercise testing during NPM shutdown to demonstrate that the valve will open on low RCS pressure while the trip valve remains energized (closed)

RAI 03.09.06-16S1

- leakage testing, as required for Appendix J Type B for pilot valves

RAI 03.09.06-16S1

- leak testing, owner specified leakage requirement for main valve and block valve pursuant to Mandatory Appendix I

RAI 14.03.03-3S2, RAI 14.03.03-4S2

Preservice Performance Assessment Testing and QME-1 - Power operated valves that meet the criteria of ISTA-1100 are qualified in accordance with ASME QME-1-2007 as accepted in Regulatory Guide 1.100. Each POV design is qualified to QME-1, subsection QV-7400. Qualification results are used meet the requirements of subsection QV-7463, Demonstration of Functional Capability of Production Valve Assemblies. Physical attributes, application and diagnostic test data from qualification test valves are used to develop performance assessment test parameters for the power operated valves design. Preservice performance assessment testing is utilized to verify the functional capability of the production valve to its qualified valve assembly. This determines that the valve is operating acceptably and baseline test data established, meeting the requirements for QME-1 and OM ISTC for demonstrating the functional capability of production valve assemblies.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, RAI 03.09.06-16S1

COL Item 3.9-8: A COL applicant that references the NuScale Power Plant design certification will develop specific test procedures to allow detection and monitoring of power-operated valve assembly performance sufficient to satisfy periodic verification design basis capability requirements.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16

COL Item 3.9-9: A COL applicant that references the NuScale Power Plant design certification will develop specific test procedures to allow detection and monitoring of emergency core cooling system valve assembly performance sufficient to satisfy periodic verification of design basis capability requirements.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16

(4) Check Valve Tests

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16

Check Valve Exercise Tests - Check valves identified with specific safety-related functions to transfer closed or maintain close are periodically tested.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16

There are no check valves with an open safety function in the NuScale design.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, RAI 03.09.06-16S1

There are four check valves per NPM in the NuScale inservice test plan. They are all normally closed, nozzle check valves in the feedwater line. Valves

REQUIREMENT The ASME OM-2012 Code Edition was used to develop the inservice testing plan for the NuScale Power Plant design certification.

RAI 03.09.06-16S1

ALTERNATIVE Portions of the ASME OM-2017 Code Edition,

RAI 03.09.06-16S1

SCOPE Pursuant to 10 CFR 50.55a(z), ISTA-2000 and Mandatory Appendix IV of the ASME OM-2017 Code Edition are utilized to clarify inservice test requirements for the NuScale design.

RAI 03.09.06-16S1

ISTA-2000 was utilized in developing the cold shutdown definition relief request (subsection 3.9.6.5.1). ISTA 2000 introduces definitions for "cold shutdown outage" and "refueling outage" that clarify the intent of "cold shutdown" and "refueling" as utilized in the OM Code.

RAI 03.09.06-16S1

Mandatory Appendix IV was used as an alternative to OM-2012 to develop inservice performance assessment testing as described in subsection 3.9.6.3.2 and Table 3.9-16. Mandatory Appendix IV is referenced to provide an acceptable level of quality and safety by utilizing established code requirements for testing to demonstrate that valves can perform their safety function at design basis conditions.

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, RAI 03.09.06-16S1

COL Item 3.9-7: Not Used.

3.9.6.5 Augmented Valve Testing Program

RAI 03.09.06-16S1, RAI 14.03.03-3S2, RAI 14.03.03-4S2

Components not required by ASME OM Code, Subsection ISTA-1100, but with augmented quality requirements similar to ISTA-1100 are included in an augmented inservice testing program. These components either provide nonsafety backup to a safety-related function or are nonsafety related valves that provide an augmented quality function. These components will be tested to the intent of the OM Code and applicable addenda, as endorsed by 10 CFR 50.55a(f), or where relief has been granted by the NRC in accordance with 10 CFR 50.55a(f) commensurate with their augmented requirements. The valve augmented test requirements are presented in Table 3.9-17. ~~and include valves in the chemical and volume control system, condensate and feedwater system, and the main steam system.~~

RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-16, RAI 03.09.06-16S1, RAI 06.02.04-5S1, RAI 06.02.04-5S2, RAI 06.02.04-5S3, RAI 09.01.03-1, RAI 14.03.03-3S2, RAI 14.03.03-4S2

Table 3.9-16: Valve Inservice Test Requirements per ASME OM Code

Valve No.	Description	Valve / Actuator[1]	Position	Augmented Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
Chemical and Volume Control System								
CVC-AOV-0089	Demineralized Water Supply to CVC Makeup Upstream Isolation Valve	BALL Remote AO	Closed	Active Boron Dilution Prevention	Class 3 Category B	Position Verification Test/2 Years Exercise Full Stroke/Quarterly Failsafe Test/Quarterly Performance Assessment Test	1	5, 16
CVC-AOV-0090	Demineralized Water Supply to CVC Makeup Downstream Isolation Valve	BALL Remote AO	Closed	Active Boron Dilution Prevention	Class 3 Category B	Position Verification Test/2 Years Exercise Full Stroke/Quarterly Failsafe Test/Quarterly Performance Assessment Test	1	5, 16
Containment System								
CVC-HOV-0324	Pressurizer Spray Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	2	6, 16
CVC-HOV-0325	Pressurizer Spray Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	2	6, 16
CVC-HOV-0330	Chemical and Volume Control System Injection Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	2	6, 16
CVC-HOV-0331	Chemical and Volume Control System Injection Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Reactor Coolant Pressure Boundary Containment Isolation	Class 1 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	2	6, 16

Table 3.9-16: Valve Inservice Test Requirements per ASME OM Code (Continued)

Valve No.	Description	Valve / Actuator[1]	Position	Augmented Function(s)[2]	ASME Class / IST Category	IST Type and Frequency[3]	Valve Group[4]	Notes
CFD-HOV-0022	Containment Flooding & Drain Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/ Quarterly Failsafe Test/ Quarterly Containment Isolation Leak Test Performance Assessment Test	2	6, 16
RCCW-HOV-0184	Reactor Component Cooling Water Inlet Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	2	6, 7, 16
RCCW-HOV-0185	Reactor Component Cooling Water Inlet Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	2	6, 7, 16
RCCW-HOV-0190	Reactor Component Cooling Water Outlet Inboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	2	6, 7, 16
RCCW-HOV-0191	Reactor Component Cooling Water Outlet Outboard Containment Isolation Valve	BALL Remote HO	Closed	Active Containment Isolation	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Containment Isolation Leak Test Performance Assessment Test	2	6, 7, 16
FW-HOV-0137	Feedwater Isolation Valve	BALL Remote HO	Closed	Active Feedwater Isolation Containment Isolation Decay Heat Removal Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	2~~3~~	8, 15, 16
FW-HOV-0237	Feedwater Isolation Valve	BALL Remote HO	Closed	Active Feedwater Isolation Containment Isolation Decay Heat Removal Boundary	Class 2 Category A	Position Verification Test/2 Years Exercise Full Stroke/Cold Shutdown Failsafe Test/Cold Shutdown Leak Test Performance Assessment Test	2~~3~~	8, 15, 16

the valves is to open and remain open when actuated. The closed safety function to support the reactor coolant pressure boundary is passive. The reset pilot is a nonsafety function and is not inservice tested as part of the ASME OM Code IST program. The trip valve is tested during failsafe and exercise testing.

RRVs and RVVs do not have specific leakage criteria. Seat tightness will be in accord with the requirements of the OM Code Mandatory Appendix I. ECCS valve seat leakage will be RCS unidentified leakage and must meet Technical Specification surveillance criteria. The owner's seat tightness criteria should be in accordance with the methods prescribed in OM Mandatory Appendix I, Table I-8220-1. The associated pilot valve bodies form part of the reactor coolant and containment boundaries and are subject to 10 CFR 50 Appendix J Type B testing.

ISTC-5110 Power Operated Relief Valves - RRVs and RVVs have attributes of both power operated valves (ISTC-5100) and relief valves (ISTC-5240). Performance assessment testing per Note 16 includes a functional test of the inadvertent actuation block at normal RCS pressure to confirm that the ECCS valve does not open. Testing also includes an operational test to demonstrate that the valves not exercise tested will open on low RCS pressure even though the trip valves remain energized (closed).

13. Reactor Safety Valves (Section 5.1.3.5): These valves are not exercised for inservice testing; their position indication components are tested by local inspection without valve exercise. RSVs do not have specific leakage criteria. Seat tightness will be in accord with the requirements of the OM Code Mandatory Appendix I. Any RSV seat leakage will be RCS unidentified leakage and must meet Technical Specification surveillance criteria. Owner's as-left seat tightness criteria shall be no observed leakage utilizing the methods prescribed in OM Mandatory Appendix I, Table I-8220-1.
14. Steam Generator System Thermal Relief Valves (Section 5.4.1.2): These thermal relief valves are located inside containment on each SG system feedwater header.
15. All secondary systems containment isolation valves close to complete the decay heat removal system boundary. All of these valves have specific leakage criteria and are tested per NuScale Technical Specification surveillance test (Technical Specification SR 3.7.1.2 and SR 3.7.2.2).
16. These valves are subject to performance assessment testing per the requirements of 10 CFR 50.55a. The test frequencies are to be established in accordance with the intent ASME OM Code - 2017, Mandatory Appendix IV. The approach detailed in Mandatory Appendix IV shall be applied to both AOVs and HOVs.

    OM Mandatory Appendix IV and this Plan address the attributes of a successful POV program as delineated in NRC Regulatory Issue Summary (RIS) 2000-3, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power Operated Valves Under Design Basis Conditions." See subsection 3.9.6.3~~2~~.2 (3) for the factors to be considered in the evaluation of performance assessment testing.
17. Reactor Building Rupture Disks: These passive, redundant, nonreclosing pressure relief devices provide reactor building overpressure protection. 5 year replacement frequency unless historical data indicates a requirement for more frequent replacement (OM I-1360).

RAI 14.03.03-3S2, RAI 14.03.03-4S2

When the PSCIVs are closed, the fluid between the two valves could have heat added from the containment or external environment. Overpressure protection for this condition is provided in the dual valve design. As the fluid heats up and expands, the pressure increase causes a thermal relief device to relieve fluid back to the CNV across the inboard PSCIV. The thermal relief device is an integral component of the inboard seal and is Type C tested along with the inboard CIV to determine leakage. Testing and maintenance of the thermal relief device is described in performance assessment testing of hydraulically operated valves in Section 3.9.6.3.2.

RAI 06.02.04-3

When valve testing is completed, the test equipment can be vented and the valves can be realigned. The tested valve is opened and the second CIV closed. The test alignment for the second CIV is now established. The test equipment can be re-pressurized and the second valve tested. Following completion of Type C testing, the test insert is removed and the inservice insert is installed. The inservice insert is a passive isolation barrier that requires an Appendix J, Type B leakage test (refer to Table 6.2-4).

Isolation valves whose seats may be exposed to the containment atmosphere during a LOCA are tested with air or nitrogen at a pressure not less than Pa.

The leak rates of penetrations and valves subject to Type B and C testing are combined in accordance with 10 CFR 50, Appendix J. The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 La. If repairs are required to meet this limit, the results shall be reported in a separate summary to the NRC, to include the structural conditions of the components which contributed to the failure. As each Type B or C test, or group of tests, is completed the combined total leak rate is revised to reflect the latest results. Thus, a reliable current summary of containment leak tightness is maintained. Leak rate limits and the criteria for the combined leakage results are described in the plant Technical Specifications.

6.2.6.4 Scheduling and Reporting of Periodic Tests

Schedules for performance of the periodic Type B and C leak rate tests are specified in the plant Technical Specifications [Section 5.5.9]. Provisions for reporting test results are described in the containment leakage rate testing program.

Type B and C tests may be conducted at any time that plant conditions permit, provided that the time between tests for any individual penetration or valve does not exceed the maximum allowable interval specified in the containment leakage rate testing program.

Each time a Type B or C test is completed, the overall total leakage rate for all required Type B and C tests is updated to reflect the most recent test results. In addition to the periodic tests, any major modification or replacement of a component that is part of the reactor containment boundary performed after the preoperational leakage rate test is followed by either a Type B or C test (as applicable) for the area affected by the modification.