Response to SDAA Audit Question

Question Number: A-5.4.3.3.4-1

Receipt Date: 03/27/2023

Question:

Section 5.4.3.3.4 states, “SECY 94-084 states that the heat removal system must have sufficient capacity to reduce the RCS temperature to 420 degrees F (safe shutdown condition) within 36 hours and that cooling to 420 degrees F must still be possible in the event of a single active failure. Therefore, a longer cooling period is acceptable in the case of a pipe break (passive failure) that removes the functionality of an entire train. Cases are evaluated for single-train and two-train operation at nominal initial conditions, both of which show that the DHRS is capable of bringing the NPM to a passively-cooled safe shutdown condition.” This is not accurate. SECY-94-084 does not state “single active failure.” It uses the term “single failure” consistent with 10 CFR 50, Appendix A. Appendix A specifies both passive and active failures are to be considered in the design. Additionally, pipe breaks are not the only type of passive failures. NuScale should (1) delete the word “active” and the second sentence from the above quote, and (2) ensure the design of the DHRS and Table 5.4-8 adequately addresses both active and passive failures consistent with 10 CFR 50, Appendix A.

Additionally, Table 5.4-8, “Failure Modes and Effects Analysis - Decay Heat Removal System,” Note 1 states, “Pipe ruptures outside of the CNV are not postulated as the DHRS piping is specified to meet additional stress criteria specified in Branch Technical Position 3-4.” This approach is only applicable for GDC 4 compliance regarding exclusion of dynamic effects, and is not sufficient basis for excluding consideration of pipe breaks within the design basis Chapter 15 safety analysis. NuScale should (1) remove the above quote from the application, and (2) provide analysis of a DHRS pipe break, or provide justification that the existing Chapter 15 analysis addresses or bounds DHRS piping failures.

Response:

Part B of SECY 94-084 addresses the definition of passive failure. In the discussion, reference is made to 10 CFR Part 50, Appendix A, as indicated in this audit question. However, the discussion in Part B of SECY 94-084 also references SECY 77-439 and includes the following excerpt taken from SECY 77-439, "...it has been judged in most instances that the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in the application of single failure criterion to assure safety of a nuclear power plant." SECY 94-084 then states that in licensing reviews, only on a long-term basis does the staff consider passive failures in fluid systems as potential accident initiators in addition to initiating events, and that the staff finds no reason to alter this regulatory practice for passive advanced light water reactor (ALWR) designs except for check valves. In this manner, SECY 94-084 distinguishes between failures that are event initiators, such as a passive pipe break, and failures that occur subsequent to event initiation. For the latter type of failures, SECY 94-084 identifies that passive single failures in fluid systems, unlike active single failures, only need to be addressed in the long-term.

Part C of SECY 94-084 addresses safe shutdown requirements. This section includes the specification of the performance requirement for passive decay heat removal systems to have sufficient capacity to reduce reactor coolant temperature to 420 degrees F within 36 hours of shutdown. The specification goes on to state that upon a single failure, a safety-grade decay heat removal from the reactor coolant system (RCS) shall be possible from full RCS operating pressure and temperatures to a passively cooled, safe shutdown condition. This second specification, which includes the discussion of the single failure, does not explicitly identify the timing or temperature of the safe, stable condition. Based on the background provided in Part B of SECY 94-084, a single passive failure is only considered in the long-term basis.

The intent of the SECY 94-084 discussion in Section 5.4.3.3.4 was to distinguish between the timing of the different single failure types (active versus passive) based on Part B of SECY 94-084. However, NuScale understands that the use of the word “active” in the reference to SECY 94-084 is potentially confusing given that it does not directly appear in the Part C discussion. Therefore, the discussion of SECY 94-084 in Section 5.4.3.3.4 is clarified as indicated in the markups associated with this response.

The thermal-hydraulic performance of the decay heat removal system (DHRS) evaluated in Section 5.4.3.3.4 is performed assuming a single failure, regardless of whether it is passive or active. Even with a loss of one train, DHRS is capable of bringing the NuScale Power Module (NPM) to a passively cooled safe shutdown condition, as defined by Technical Specifications.

For example, Figure 5.4-9 and Figure 5.4-10 show the performance with one and two trains of DHRS available, respectively.

With respect to Table 5.4-8, “Failure Modes and Effects Analysis - Decay Heat Removal System,” the purpose of the table is to demonstrate that the DHRS is capable of performing its design function. The table includes consideration of both active and passive failures. For passive failures of the DHRS loop pressure boundary, a footnote to the table is included. Note 1, first sentence, states, “Pipe ruptures outside of the CNV are not postulated as the DHRS piping is specified to meet additional stress criteria specified in Branch Technical Position 3-4.” This footnote is consistent with the following statement in Section 3.6.2.1.2:

> *Breaks are not postulated in the DHRS piping outside containment in accordance with BTP 3-4, B.1.(ii). Subject to certain design provisions, NRC guidance allows breaks associated with high-energy fluid systems piping in containment penetration areas to be excluded from the design basis. Though the DHRS piping extends beyond what would traditionally be considered a containment penetration area, this approach is chosen because the DHRS cannot be isolated from the CNV as there are no isolation valves.*
>
> *Breaks are not postulated in this segment of piping because it meets the design criteria for break exclusion in a containment penetration area (Section 3.6.2.1.2). Although the DHRS condenser design uses piping products, it is considered a major component and not a piping system; thus, breaks are not postulated.*

The footnote in Table 5.4-8 and the discussion in Section 3.6.2.1.2 for the Standard Design Approval Application (SDAA or US460) Final Safety Analysis Report (FSAR) are identical to what was included in the Design Certification Approval (DCA or US600) FSAR, which was reviewed and approved by the NRC. In Section 3.6.2.4.1.1 of the NuScale US600 final safety evaluation report (FSER), the NRC states (emphasis added):

> *In DCA Part 2, Tier 2, Section 3.6.2.1.2, the applicant stated that break exclusion criteria are applied to the ASME BPV Code Class 1 piping (i.e., the four CVCS RCS lines) from the CNV head to the first isolation valve and to the ASME BPV Code Class 2 MS and FW piping from containment to the first isolation valve, **as well as the DHRS piping outside containment**. The applicant also stated that the remaining piping under the bioshield, including the refueling pipe spools, is designed to comply with BTP 3-4, Revision 2, Section B, Item A(iii), to preclude breaks at intermediate locations, by limiting stresses calculated by the sum of Equations (9) and (10) in NC/ND-3653 of Section III of the*

ASME BPV Code to not exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

Then, the NRC notes (emphasis added):

Based on its review of DCA Part 2, Tier 2, Section 3.6.2.1.2, the staff determined that the applicant had not applied the break exclusion in the containment penetration areas as envisioned in BTP 3-4, Section B, Item A(ii). ***Specifically, the applicant applied the break exclusion criteria to the areas beyond the scope of BTP 3-4 for the containment penetration areas.*** *Also, the applicant did not consider the welds between the CNV vessel wall and the CNV safe-end for the CIVs to be within the containment penetration area and did not include these welds within the BTP 3-4 break exclusion boundary...*

And later the NRC concludes:

Based on the review of the information provided in DCA Part 2, Tier 2, Sections 3.6.2.1.2, 3.6.2.1.2.2, 3.6.2.1.2.3, Figure 3.6-33, Table 6.2-3, Table 6.6-1, and Appendix A to TR-0818-61384-P, Revision 2, as described above, the staff finds that the applicant has adequately demonstrated its design provisions and specified a 100-percent volumetric inservice examination for all the pipe welds within the break exclusion areas. This meets the applicable BTP 3-4 break exclusion criteria in the NRC's guidelines, and therefore, NuScale's application of the break exclusion areas is acceptable.

Chapter 5 of the US600 FSER identifies that the NRC staff reviewed the assessment of the failure modes and effects analysis in DCA Table 5.4-8.

The request to remove the first sentence of the footnote on Table 5.4-8 is not appropriate because it would eliminate the link to Section 3.6.2.1.2 for DHRS expanded break exclusion, as approved in the US600 FSER. Also, the footnote remains unchanged from the DCA FSAR, which was accepted in the US600 FSER.

The NRC noted in this audit question that the language of the footnote is not clear in that it should only apply to break exclusion in the mechanical design perspective and should not exclude the DHRS piping from consideration in the safety analyses in Chapter 15. As previously discussed, SECY 94-084 distinguishes between passive failures subsequent to event initiation and those passive failures that may be the event initiators. Chapter 15 addresses the evaluation of initiating events.

A postulated passive failure of the DHRS pressure boundary piping may be considered as follows, using Figure 5.4-7 as reference. On the steam side of DHRS, the DHRS piping connects to the main steam lines outside of containment. The DHRS actuation valves are closed during normal operation, consistent with the initial conditions for a Chapter 15 initiating event. The plant response to a break in the DHRS steam piping between the connection to the main steam line and the DHRS actuation valves would be similar to the plant response to a break in the main steam line between the primary main steam isolation valve (MSIV) and secondary MSIV, where the primary MSIV is assumed to be failed. In both cases, the break is unisolable, total blowdown of the affected steam generator (SG) occurs, and one DHRS train is unavailable for decay heat removal. A break in the main steam line between the primary MSIV and secondary MSIV, where the primary MSIV is assumed to be failed, is explicitly analyzed as part of the analyses in Section 15.1.5, “Steam Piping Failures Inside and Outside of Containment.” The Section 15.1.5 analyses consider a double-ended guillotine break as well as a range of split breaks for this break location. The range of break sizes considered is sufficient to ensure limiting results for the figures of merit (i.e., RCS pressure, SG pressure, and minimum critical heat flux ratio (MCHFR)). The effective break sizes analyzed in Section 15.1.5 are compared to the DHRS steam piping size in Table 1. As shown in Table 1, a postulated break in the DHRS steam piping is bounded by the range of piping breaks already considered in Section 15.1.5.

During normal operation, the DHRS steam piping downstream of the closed DHRS actuation valves, the DHRS steam header, the DHRS condenser tubes, the DHRS condensate header, and the DHRS condensate piping up to the connection to the feedwater piping are water filled. Part of the DHRS condensate piping is inside containment while the rest of the components are outside. The plant response to a break in the DHRS condensate piping inside containment would be similar to the plant response to a feedwater line break inside containment. In both cases, the break depressurizes the impacted SG, drains the DHRS piping and condenser in the affected loop (making that train of DHRS unavailable for decay heat removal), and increases containment pressure. A break in the feedwater piping inside containment is explicitly analyzed as part of the analyses in Section 15.2.8, “Feedwater System Pipe Breaks Inside and Outside of Containment.” The Section 15.2.8 analyses consider a double-ended guillotine break as well as a range of split breaks for this break location. The range of break sizes considered is sufficient to ensure limiting results for the figures of merit (i.e., RCS pressure, SG pressure, MCHFR, and containment response). The effective break sizes analyzed in Section 15.2.8 are compared to the DHRS condensate piping size in Table 1. As shown in the table, a postulated break in the DHRS condensate piping inside containment is bounded by the range of piping breaks already considered in Section 15.2.8.

The plant response to a break in the DHRS piping downstream of the closed DHRS actuation valves, the DHRS condenser, or the DHRS condensate piping outside of containment would be the same as the plant response to a break in the DHRS condensate piping inside of containment except for the impact on containment conditions. For this reason, these other DHRS break locations are also compared to the analyses in Section 15.2.8 in Table 1. As described in Section 15.2.1, smaller feedwater piping breaks inside containment result in other module protection system (MPS) signals prior to containment-related MPS signals. In addition, the DHRS piping breaks outside of containment could also be detected by the high under-the-bioshield temperature MPS signal which is not credited in the analyses in Section 15.2.8. Therefore, the lack of a containment response for the DHRS piping breaks outside containment does not significantly impact the comparison to the spectrum of analyses in Section 15.2.8. As shown in Table 1, a postulated break in the DHRS steam piping downstream of the closed DHRS actuation valves, the DHRS steam header, the DHRS condenser tubes, the DHRS condensate header, and the DHRS condensate piping outside containment is bounded by the range of piping breaks already considered in Section 15.2.8.

The analyses already performed in Section 15.1.5 and Section 15.2.8 collectively bound a postulated passive failure of the DHRS pressure boundary as an initiating event. Instead of eliminating the first sentence of the footnote to Table 5.4-8, the footnote will be clarified to reference Chapter 15 for consideration of DHRS failures as initiating events. In addition, Section 15.1.5 and Section 15.2.8 will be revised to identify that the analyses bound potential DHRS breaks.

Table 1
Comparison of Main Steam Piping, Feedwater Piping, and Decay Heat Removal System Piping and Condenser Dimensions

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}}2(a),(c)

Markups of the affected changes, as described in the response, are provided below:

Audit Question A-5.4.3.3.4-1

Table 1.9-7: Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and Associated Staff Requirements Memoranda)

Doc ID	Title	Conformance Status	Comments	Section
SECY-89-013	Design Requirements Related to the Evolutionary Advanced Light Water Reactors	Conforms	Addressed through SECY-90-016 and SECY-93-087. Table 1.9-8 contains further information.	-
SECY-90-016	Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements	Partially Conforms	This SECY was directed towards evolutionary advanced light water reactor (ALWR) designs. The applicability of certain SECY-90-016 issues to passive plants was later established in SECY-93-087 and SECY-94-084. Conformance with SECY-93-087 and SECY-94-084 is addressed in Table 1.9-7 and Table 1.9-8. ~~As a passive ALWR design, the NuScale design conforms to the passive plant guidance of SECY-93-087 and SECY-94-084, rather than that of SECY-90-016. Table 1.9-8 contains further information.~~	19.1 19.2
SECY-90-241	Level of Detail Required for Design Certification under Part 52	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-90-377	Requirements for Design Certification under 10 CFR Part 52	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-91-074	Prototype Decisions for Advanced Reactor Designs	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-91-078	Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light WaterReactor (LWR) Certification Issues	Not Applicable	SECY-91-078 pertains to evolutionary ALWR designs and is not directly applicable to passive plant designs.	Not Applicable
SECY-91-178	ITAAC for Design Certifications and Combined Licenses	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	14.3
SECY-91-210	ITAAC Requirements for Design Review and Issuance of FDA	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-91-229	Severe Accident Mitigation Design Alternatives for Certified Standard Designs	Conforms	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	19.2
SECY-91-262	Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs	Conforms	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	-
SECY-92-053	Use of Design Acceptance Criteria During the 10 CFR Part 52 Design Certification Reviews	Conforms	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	14.3

Table 1.9-7: Conformance with Advanced and Evolutionary Light Water Reactor Design Issues (SECYs and Associated Staff Requirements Memoranda) (Continued)

Doc ID	Title	Conformance Status	Comments	Section
SECY-92-092	The Containment Performance Goal, External Events Sequences, and the Definition of Containment Failure for Advanced LWRs	Conforms	Incorporated into NRC Orders, regulatory guidance, and pending rulemaking.	-
SECY-93-087	Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs	Partially Conforms	Table 1.9-8 provides further information.	1.9
SECY-94-084	Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Design (RTNSS)	Partially Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents. The NuScale fire protection system does not contain any RTNSS equipment. The design does not align with the EPRI position on safe shutdown. SECY-94-084 states, "The staff believes that other plant conditions may constitute a safe shutdown state as long as reactor subcriticality, decay heat removal, and radioactive materials containment are properly maintained for the long term." The US460 standard design meets this statement for a safe shutdown state. ~~However, Section C, Safe Shutdown Requirements, of the SECY discusses the stable shutdown condition for passive ALWR, which is applicable to the NPP.~~	5.4 8.1 8.2 8.3 8.4 9.2.5 Appendix 9A 15.0.4 19.3
SECY-94-302	Source-Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	-
SECY-95-132	Policy and Technical Issues Associated with Regulatory Treatment of Non-Safety Systems in Passive Plant Designs	Conforms	Incorporated into 10 CFR 52 and implementing NRC guidance documents.	8.1 8.2 8.3 8.4 19.3
SECY-96-128	Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design	Partially Conforms	Section IV of this SECY applies.	19.3
SECY-14-038	Performance-Based Framework for Nuclear Power Plant Emergency Preparedness Oversight	Not Applicable	None.	13.3

conditions. Off-nominal transients are bounding evaluations of the combined impact of several factors on DHRS heat removal capability.

The factors impacting DHRS heat removal evaluated in the off-nominal cases include: core power uncertainty, reactor pool temperature, valve actuation delays, valve stroke times, noncondensible gas volume, system leakage, SG level, SG fouling, SG tube plugging, DHRS condenser fouling, and number of operational DHRS trains. If applicable, these parameters are biased to produce either a high or low DHRS loop inventory.

Decay Heat Removal System Performance Analysis

The analysis evaluates the DHRS capability of removing heat over a range of DHRS loop inventories with the steady state model. Sensitivity cases indicate that the DHRS is insensitive to valve coefficient and orifice loss coefficient. However, DHRS performance is sensitive to DHRS inventory. Low inventory greatly reduces the heat transfer rate. Similarly, with a high inventory there is also a decline in performance.

Fouling of the heat transfer surfaces and SG tube plugging has a moderate effect on DHRS performance, decreasing the peak heat removal capability, and the presence of noncondensible gas has an impact on DHRS performance.

The presence of non-condensible gas has a small effect on total system performance. The decrease in heat removal due to noncondensible gas increases as the DHRS pressure decreases, because of the same mass of noncondensible gas fills a larger fraction of the gas space.

Assessment of the likelihood of noncondensible gas accumulating down to the level sensors in the DHRS steam piping during the operating cycle concludes that reaching the noncondensible gas limit in the DHRS steam piping is unlikely, based on the allowed normal ranges and action levels specified in the secondary water chemistry control program.

Consideration of steam leakage through the closed MSIVs and water leakage through the closed FWIVs informs a bounding low inventory case because both types of leakage affect system performance in the same manner. In the high inventory case, any leakage from the DHRS loop improves system performance. Additionally, the loss of loop inventory mitigates secondary over-pressurization situations that would otherwise occur. Therefore, omission of loop leakage from high inventory cases creates the most conservative limiting heat transfer case.

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SECY 94-084 provides general performance recommendations for passive decay heat removal systems to~~states that the heat removal system must~~ have sufficient capacity to reduce the RCS temperature to 420 degrees F (safe shutdown condition) within 36 hours and that reaching a safe stable condition~~cooling to 420 degrees F must still be~~ is possible in the event of a single~~ active~~ failure. ~~Therefore, a longer cooling period is acceptable in the case of a pipe break (passive failure)~~SECY 94-084 was considered in the development of the DHRS design capacity. For the US460 design, the technical specifications define the conditions for passive cooling, which include single-train operation of DHRS to account for a failure that removes the functionality of an entire train. Cases are evaluated for single-train and two-train operation at nominal initial conditions, both of which show that the DHRS is capable of bringing the NPM to a passively-cooled safe shutdown condition. The consideration of potential DHRS failures as initiating events is addressed in Chapter 15.

Decay Heat Removal System Performance Results

The system performance analysis indicates the DHRS removes appreciable amounts of heat over a wide range of initial conditions.

Figure 5.4-8 shows RCS cooldown for 36 hours from full power conditions with one DHRS train in operation assuming nominal system conditions. Initially, the decay heat exceeds the combined heat removal of the DHRS. The decay heat power drops off quickly as the transient progresses, and the DHRS begins to remove more heat than is added. This imbalance cools the RCS. This case also demonstrates that a single train of DHRS can provide sufficient cooling of RCS using nominal system conditions.

Figure 5.4-9 shows RCS cooldown for 36 hours from full power conditions with two DHRS trains in operation assuming nominal system conditions. For this nominal two DHRS train case, RCS average temperature stabilizes below 300 degrees F within 36 hours.

Figure 5.4-10 shows an off-nominal DHRS actuation with high DHRS inventory and low DHRS heat transfer. The heat removal bias is lower because of the high fouling, high tube plugging, and a high volume of non-condensible gas. This case assumes 102 percent reactor power. For this off-nominal two DHRS train case, RCS average temperature stabilizes below 420 degrees F within 36 hours.

Figure 5.4-11 shows an off-nominal DHRS actuation with low DHRS inventory and low DHRS heat transfer. The heat removal bias is lower because of the high fouling, high tube plugging, and a high volume of non-condensible gas. This case also uses the presence of loop leakage to further bias results. This case assumes 102 percent reactor power. This event also considers the presence of decreasing inventory due to loop leakage. For this off-nominal two DHRS train case, RCS average temperature stabilizes below 420 degrees F within 36 hours.

The final results show that the DHRS is capable of removing appreciable amounts of heat over a relatively wide range of inventories. The analyses further show the ability to accommodate fouling, SG tube plugging, and the presence of noncondensible gas, thus precluding the need for high-point vent

Audit Question A-5.4.3.3.4-1

Table 5.4-8: Failure Modes and Effects Analysis - Decay Heat Removal System

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System[1]	Method of Failure Detection
DHRSAV (normally closed, fail open)	1) Maintain DHRS in standby	A) Spurious opening	Mechanical Electrical/I&C	Affected DHRS condenser has open flow path to SG. Turbine must be isolated to prevent damage. Normal cooling is available through FW. Unaffected DHRS train remains available.	• Valve position indication • Steam pressure • Steam temperature • Passive condenser temperature
		B) Spurious DHRS actuation	Electrical/I&C Operator error	Both DHRS trains initiate operation, and the MSIVs and FWIVs close. The engineered safety features actuation system generates a reactor trip signal regardless of if the DHRS actuation signal is erroneous or not.	• PZR pressure • Steam pressure • Valve position indication • PZR level • Passive condenser temperature • Passive condenser level • Reactor trip
		C) Leakage (passive failure)	Mechanical	Minor leakage does not impact DHRS operation. DHRS inventory is maintained by the continuous makeup of FW to the SG. Major valve seat leakage causes both DHRS trains to actuate.	• Minor seat leakage: -none • Major seat leakage: -Passive condenser temperature -Steam pressure -Steam temperature • Minor valve bonnet leakage: -periodic inspections • Major valve bonnet leakage: -passive condenser temperature

Table 5.4-8: Failure Modes and Effects Analysis - Decay Heat Removal System (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System[1]	Method of Failure Detection
SGS Thermal Relief Valve	1) Provide pressure boundary	A) Spurious opening of relief valve with failure to close (passive failure)	Mechanical	A spurious opening of the thermal relief valve with a failure to close causes the affected DHRS train to lose inventory and become inoperable. Safe shutdown without ECCS actuation is achieved with the remaining DHRS train.	• Steam pressure • DHRS level instrument switches • Containment leakage monitoring instrumentation (containment evacuation system)
	2) Provide overpressure protection when SGS is water solid	A) Failure of valve to lift at setpoint	Mechanical	A failure of the valve to lift during a water solid over pressure scenario could cause deformation or rupture of pressure boundary components. A rupture would cause the affected DHRS train to be inoperable. The module is in safe shutdown before this event, and cooling resumes with the unaffected DHRS train or through flooding the CNV with the containment flooding and drain system.	• Steam pressure • DHRS level instrument switches
DHRS Loop Pressure Boundary (Includes piping inside CNV, SG tubes)(~~1~~2)	1) Provide a pressure boundary for the SGS and DHRS	A) Pipe rupture and loss of DHRS loop inventory (passive failure)	Mechanical	A pipe rupture of any pressure boundary piping that is part of the DHRS loop (SGS and DHRS piping within containment isolation valves) causes the affected DHRS train to lose inventory and become inoperable. Safe shutdown without ECCS actuation is achieved with the remaining DHRS train.	• Steam pressure • DHRS level instrument switches • Containment leakage monitoring instrumentation (containment evacuation system)

Notes:

(1) This table identifies the impact of the failure on the DHRS system. The plant response to a potential DHRS failure as an initiating event is addressed in Chapter 15.

(2~~1~~) Pipe ruptures outside of the CNV are not postulated as the DHRS piping is specified to meet additional stress criteria specified in Branch Technical Position 3-4.
Containment leakage monitoring instrumentation only identifies pressure boundary failures within the CNV.

5) The most limiting plant systems single failure shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.

- There is no limiting single failure that could occur during an increase in steam flow event that could result in more severe conditions with respect to the acceptance criteria as discussed in Section 15.1.3.2.

15.1.4 Inadvertent Opening of Steam Generator Relief or Safety Valve

This section is typically used to provide an analysis for an inadvertent opening of an SG relief or an MSSV. The NuScale Power Plant US460 standard design does not have an SG relief valve, but does have two MSSVs. The event is initiated by the spurious opening of one of the two main MSSVs, which are located downstream of the secondary MSIVs. The MSSVs are sized to accommodate 100 percent of the full power steam flow. As both valves are required to meet this flow requirement, the spurious opening of one MSSV following a mechanical failure yields a steam flow increase less than 100 percent of the flow at full power conditions. A spurious opening of the turbine bypass valve yields a similar system response as it is also located downstream of the secondary MSIVs. However, the increase in steam flow due to a spurious opening of the turbine bypass valve could result in a 100-percent increase in steam flow. Therefore, the spectrum of possible steam increases because of a full or partial turbine bypass valve opening covers and bounds the steam increases because of an inadvertent opening of an MSSV. The analysis of a limiting increase in steam flow is presented in Section 15.1.3.

15.1.5 Steam Piping Failures Inside and Outside of Containment

15.1.5.1 Identification of Causes and Accident Description

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A steam line break (SLB) event for the NuScale Power Plant US460 standard design could range from a small break to a double-ended rupture of the main steam line. This event could occur inside or outside of the containment vessel (CNV). A spectrum of SLB locations with varied core and plant conditions are analyzed to determine the scenarios with the most severe results. The SLB spectrum is also adequate to address a postulated break of the DHRS piping between the connection to the steam line piping outside containment and the DHRS actuation valves outside containment.

An SLB inside the CNV increases the pressure inside containment, reaching the high containment pressure MPS setpoint. The high containment pressure signal actuates RTS, isolates the CNV and chemical and volume control system, deenergizes PZR heaters, and actuates SSI and DHRS. The break flow decreases because of SG depressurization and increasing CNV backpressure until dryout from FW isolation. The containment pressure is sensitive to any SLB size, so the MPS detects the break earlier than a comparable break outside of containment. A spectrum of breaks inside containment is evaluated to ensure containment pressure is acceptable. The peak containment pressure remains below the design limit for all postulated events, as shown in Section 6.2,

4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response in order to meet the requirements of GDCs 10 and 15.

   - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

5) The most limiting plant systems single failure shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.

   - No single failures are identified that have adverse impact on the acceptance criteria.

6) The guidance provided in SECY-77-439 and SECY-94-084 with respect to the consideration of the performance of nonsafety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.

   - The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.2.7.2 and Section 15.2.7.3 ensure the guidance provided is met.

15.2.8 Feedwater System Pipe Breaks Inside and Outside of Containment

15.2.8.1 Identification of Causes and Event Description

Audit Question A-5.4.3.3.4-1

A feedwater line break (FWLB) event could range from a small split crack to a double ended rupture of the FW line. This event can occur both inside and outside of the containment vessel (CNV) due to seismic events, thermal stress, or cracking of the FW piping. A spectrum of FWLB locations and break sizes, with varied core and plant conditions, are analyzed to determine the scenarios with the most severe results. The FWLB spectrum is also adequate to address a postulated break of DHRS piping between the DHRS actuation valves outside containment and the connection to the feedwater line inside containment.

Large FWLBs inside the CNV increase the pressure in the evacuated atmosphere, resulting in a loss of containment vacuum and actuating the high containment pressure MPS signal. The high containment pressure MPS signal actuates the reactor trip system (RTS), isolates containment, and actuates SSI and DHRS. The break depressurizes the impacted SG system and drains the DHRS piping and condenser in the affected loop. The non-impacted SG system and DHRS loop continues to provide cooling to the RCS. Smaller FWLBs inside containment reach the high PZR pressure MPS signal, resulting in reactor trip, SSI, and DHRS actuation, before the high containment pressure MPS signal is reached.