RAIO-0919-67236

October 31, 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. 466 (eRAI No. 9482) on the NuScale Design Certification Application

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 466 (eRAI No. 9482)," dated May 04, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 466 (eRAI No.9482)," dated October 26, 2018
3. NuScale Power, LLC Supplemental Response to NRC "Request for Additional Information No. 466 (eRAI No. 9482)," dated January 28, 2019
4. NuScale Power, LLC Supplemental Response to NRC "Request for Additional Information No. 466 (eRAI No. 9482)," dated May 22, 2019

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 Enclosures to this letter contain NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9482:

- 06.02.01.01.A-18

Enclosure 1 is the proprietary version of the NuScale Supplemental Response to NRC RAI No. 466 (eRAI No. 9482). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

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

If you have any questions on this response, please contact Rebecca Norris at 541-602-1260 or at rnorris@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8H12
Omid Tabatabai, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Michael Dudek, NRC, OWFN-8H12

Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9482, proprietary

Enclosure 2: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9482, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0919-67237

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9482, proprietary

Enclosure 2:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9482, nonproprietary

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

eRAI No.: 9482
Date of RAI Issue: 05/04/2018

NRC Question No.: 06.02.01.01.A-18

Conservatism in the NPM Initial Conditions for the CNV Safety Analyses

To meet the General Design Criteria (GDCs) 16, 38, and 50 of Appendix A to 10 CFR Part 50 relevant to the containment design basis and guided by the Design-Specific Review Standard (DSRS) for NuScale Small Modular Reactor (SMR) Design Section 6.2.1, the staff is reviewing the applicant's analytical models and assumptions used in the containment response analysis methodology (CRAM) to determine if the licensing-basis safety analyses are acceptably conservative. Specifically, the staff needs to assess the conservatism of the licensing-basis models, constitutive/closure relations, model input parameters, and initial/boundary conditions used for the applicant's NPM design basis event (DBE) containment response analyses, in order to conclude that the results are valid over the applicable range of DBE conditions.

A limiting DBE model is expected to use the most conservative NPM initial and boundary conditions for the CNV safety analyses, based on the most biased reactor operating conditions and the limiting technical specifications. These initial conditions and assumptions should be based on the range of normal operating conditions with consideration given to maximizing the calculated peak containment pressure and temperature. In this regard, the applicant is requested to address the following three questions and update the FSAR, accordingly. The regulatory bases identified above are applicable to all questions in this RAI.

In Table 5-1 of the Containment Response Analysis Methodology Technical Report (TR-0516-49084-P, Rev. 0), the nominal CNV free volume is adjusted by {{ }}$^{2(a),(c)}$ percent as a conservative initial condition for the containment response analysis to account for uncertainty in design, blockage in containment by components, such as, piping, etc. However, the base NRELAP5 model described in {{ }}$^{2(a),(c)}$ has not been updated for numerous geometry changes reported in {{ }}$^{2(a),(c)}$, so it is not clear that a {{ }}$^{2(a),(c)}$ percent CNV free volume adjustment is adequate and would also cover the thermal expansion of the reactor pressure vessel (RPV) under operating conditions. The staff also noted that {{ }}$^{2(a),(c)}$ used {{ }}$^{2(a),(c)}$ percent conservatism in CNV free volume but that was reduced to {{ }}$^{2(a),(c)}$ percent in Rev. 2. Please explain why a {{ }}$^{2(a),(c)}$ percent reduction in containment volume is justified for a NRELAP5 base model that may not reflect the current design. If necessary, please update the NRELAP5 base model and resubmit the updated NRELAP5 models and their results for the limiting DBEs for CRAM, as submitted in response to RAI 8783; or justify how the peak CNV pressure and temperature results remain conservative with an outdated base model. When the licensing basis containment analyses are updated, the FSAR and the decks also need to reflect the rise of initial CNV pressure from 2 psia to 3 psia, as was concluded by RAI 8793, Question 29717 (06.02.01-2).

NuScale Response:

As described by NuScale letter, LO-0819-66702, dated August 26, 2019, the results of additional ECCS valve testing at reactor operating pressure and temperature necessitated an adjustment to the inadvertent actuation block (IAB) range. The letter indicated that conforming changes to the DCA would be transmitted to the NRC for review. This supplement transmits the conforming changes reflecting the revised IAB operating range.

Attached are updates to FSAR Sections 6.2, 6.3, TR-0516-49084, "Containment Response Analysis Methodology," and DCA Part 4, "Technical Specifications." These changes will be incorporated into the NuScale Standard Plant DCA Revision 4.

Impact on DCA:

FSAR Sections 6.2 and 6.3, DCA Part 4, FSAR Table 6.2, and related Technical Report TR-0516-49084, Containment Response Analysis Methodology, have been revised as described in the response above and as shown in the markup provided with this response.

BASES

BACKGROUND (continued)

b. De-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.2, “Containment Isolation Valves;” and

c. The sealing mechanism associated with each containment penetration (e.g. welds, flanges, or o-rings) is OPERABLE (i.e., OPERABLE such that the containment leakage limits are met).

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting Design Basis Accident (DBA) without exceeding the design leakage rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. The DBA analyses assume that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment is designed with an allowable leakage rate of 0.20% of containment air weight after a DBA per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure 994~~86~~ psia (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on containment leakage rate testing. L_a is assumed to be 0.20% per day in the safety analysis.

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The containment is designed to maintain leakage integrity < 1.0 L_a. Leakage integrity is assured by performing local leak rate testing (LLRT) and containment inservice inspection. Total LLRT leakage is maintained < 0.60 L_a in accordance with 10 CFR 50, Appendix J (Ref. 1). Satisfactory LLRT and ISI examination are required for containment OPERABILITY.

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-18S1, RAI 06.02.01.01.A-19

The overall limiting peak CNV pressure results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power. The overall limiting CNV pressure is ~~986~~994 psia, which is approximately ~~6~~5 percent below the CNV design pressure of 1050 psia. The LOCA event peak CNV pressure is 959 psia.

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-18S1, RAI 06.02.01.01.A-19

The overall peak CNV temperature is 526 degrees F, resulting from a reactor coolant system injection line break. The peak pressure and CNV wall temperature results for secondary system line break events are bounded by the LOCA results. These results demonstrate that the CNV design provides margin to the CNV design pressure of 1050 psia and CNV design temperature of 550 degrees F.

The supporting analyses results are presented in Chapter 5 of the containment response analysis methodology report (Reference 6.2-1). The supporting analyses are discussed by Reference 6.2-1, as well as Section 6.2.1.3 and Section 6.2.1.4.

The CNV is evaluated to demonstrate it can withstand deflagration, incident detonation and deflagration-to-detonation events for 72 hours after event initiation. Structural analysis demonstrates that the CNV is capable of withstanding the resultant combustion loads with margin to stress and strain limits as required by 10 CFR 50.44. Further details are provided in Section 6.2.5.

The structural and pressure retaining components of the CNV consist of the closure flanges and bolting, vessel shells, vessel top and bottom heads, nozzles and penetrations for piping and instrumentation, access and inspection ports, CNV support skirt, CNV support lugs, bolting for the RPV upper support ledge and the NuScale Power Module top support structure mounting assemblies. Section 3.8.2 provides additional design detail that includes a physical description of the geometry of the CNV and supports, plan views, and design criteria relating to construction techniques, static loads, and seismic loads.

Instrumentation is provided to monitor containment parameters for normal operation, anticipated operational occurrences and accidents to include temperature, pressure, isolation valve position and liquid level (GDC 13 and 64). Section 7.1 discusses the containment parameters monitored.

The integrated design of the RPV and CNV ensures that RCS leakage is collected within the CNV. In the event of primary system releases (e.g., LOCAs or valve opening events) the CNV provides for the retention of adequate reactor coolant inventory to prevent core uncovery or loss of core cooling. The reactor coolant water that collects in the CNV is passively returned to the reactor vessel by natural circulation via the emergency core cooling system (ECCS) described in Section 6.3.

Under these conditions, the CNV transfers the sensible and core decay heat through its walls to the ultimate heat sink (UHS) and provides effective passive, natural circulation emergency core cooling flow. The containment is designed so

- core power equal to rated thermal power plus an allowance for uncertainties
- conservative decay heat model
- maximized RCS and fuel stored energy
- secondary system stored energy sources

These sources of energy are also included in secondary break release analyses, with the addition of energy resulting from feedwater pump runout.

Reference 6.2-1, Section 5.1 presents the results of the NRELAP5 base case analyses of the spectrum of primary mass and energy release scenarios for the NPM, that are determined using the containment response analysis methodology. Section 5.1 also describes sensitivity analyses used to determine the limiting primary release case assumptions for CNV pressure and wall temperature and presents their results. Sections 5.2 and 5.3 of Reference 6.2-1 present the results of NRELAP5 limiting analyses of main steam line and feedwater line break (FWLB) scenarios, respectively. Table 6.2-2 presents the results of the base case and limiting CNV pressure and wall temperature analyses for primary release (LOCA and valve opening events), as well as, limiting secondary system break scenarios.

The sources and amounts of energy released to the containment and the post-accident time-dependence of mass and energy releases of postulated primary system events are described in Section 6.2.1.3.

The sources and amounts of energy released to the containment and the post~~H~~-accident time-dependence of mass and energy releases of postulated secondary system pipe ruptures inside containment (main steam and feedwater line breaks) are described in Section 6.2.1.4.

The capability to remove energy from the CNV (depressurization rate) is determined by the heat transfer rate from the CNV to the reactor pool. In all postulated events, containment pressure is shown to be reduced to less than 50 percent of the peak calculated pressure in less than 24 hours after the postulated accident (principal design criterion 38 (Section 3.1.4)). Specifically, for the limiting peak pressure case, the CNV pressure is reduced to less than 50 percent of its peak value in less than two hours.

The NuScale CNV does not include one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, to accommodate future installation of systems to prevent containment failure. As discussed in this section, the calculated peak containment pressures for design basis events remain less than the CNV internal design pressure. As discussed in Section 19.2.3, peak containment pressures do not challenge vessel integrity for any analyzed severe accident progression. Therefore, 10 CFR 50.34(f)(3)(iv) is not technically relevant to the NuScale design.

mass and energy releases are provided in Section 6.2.1.3, and secondary system mass and energy releases are provided in Section 6.2.1.4. Graphical results for the limiting CNV pressure, temperature, and mass and energy release rates are shown in Figure 6.2-9 through Figure 6.2-14. Table 6.2-7 and Table 6.2-8 provide the sequence of events for the CNV peak pressure and peak temperature cases, respectively.

In the event of a mass and energy release into CNV, a process of condensation and retention within the CNV facilitates the transfer of the energy to the UHS.

Reactor coolant released from the RPV or main steam or feedwater released from the secondary system condenses on the relatively cool inner surface of the CNV wall. The resulting condensate flows down the inner CNV wall and collects in the bottom of the CNV shell. The vapor condensation and heat removal from containment is accomplished passively by transferring the energy through the CNV wall to the reactor pool.

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-19

For releases from the RPV, the reactor coolant is condensed and collected until the condensate level within the CNV has increased to the ECCS actuation setpoint. Actuation of the safety system opens the RVVs and RRVs, further depressurizing the RPV and increasing the discharge of RPV inventory to the CNV. When RPV and CNV pressures approach equilibrium and the accumulated level in the CNV shell reaches a level where sufficient driving head is available, coolant flow from the CNV is returned to the RPV through the ECCS recirculation valves for core cooling. Opening of the RVVs and RRVs establishes the CNV shell as the outer boundary of the coolant circulation flow path. This method of passive coolant circulation and heat removal is further described in Section 6.2.2.

For a secondary system mass and energy release into containment, the released steam or feedwater is captured within the CNV by closure of the CIVs. The collected inventory is condensed and retained with the heat energy transferred to the reactor pool.

The design of the CNV is consistent with the functional requirements of the ECCS and its associated acceptance criteria. Acceptable models for evaluating emergency core cooling during the postulated mass and energy releases are defined in 10 CFR 50 Appendix K.

The CNTS design provides for the isolation of process systems that penetrate the CNV. The design allows for the normal or emergency passage of fluids, vapor or gasses through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products in the event of postulated events. The containment isolation valves are described in Section 6.2.4.

The CNV components and appurtenances are designed to ensure pressure boundary integrity for the life of the plant when considering fatigue, corrosion and wear. The CNV components and penetrations (piping, electrical and instrumentation and controls (I&C)) are designed for and tested to harsh

The above spectrum of postulated release events bounds the primary and secondary release events for the NPM.

The selection process used to determine initial conditions and boundary condition assumptions, reflecting the unique NuScale design, that are used for evaluation of containment response to postulated primary system mass and energy releases into containment are described in Reference 6.2-1, Section 3.5. Secondary system pipe break analysis initial and boundary condition assumptions and their selection process are described in Reference 6.2-1, Section 3.5. These initial conditions and assumptions are based on the range of normal operating conditions with consideration given to maximizing the calculated peak containment pressure and temperature.

The results of NRELAP5 primary system release event analyses are presented by Reference 6.2-1, Section 5.1. Additionally, Reference 6.2-1, Section 5.1 discusses the insights obtained from the sensitivity studies, used to determine limiting assumptions and single failures, that create a bounding set of assumptions. These assumptions result in the limiting CNV peak temperature and pressure for primary release event Cases 1 though 5. Similarly, Reference 6.2-1, Sections 5.2 and 5.3 present the limiting CNV pressure and temperature results for main steam line and feedwater events, respectively, along with the analysis assumptions that provide these limiting results.

Each mass and energy release event analyzed also includes the consideration of the worst case single active failure as identified by sensitivity cases and a determination of how the availability of normal AC and DC power affects the results, as described in detail by Reference 6.2-1.

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-18S1, RAI 06.02.01.01.A-18S3, RAI 06.02.01.01.A-19

The limiting LOCA peak calculated containment ~~pressure and~~ temperature, based on the mass and energy release spectrum analyses, is postulated to occur as the result of a double-ended break of the RCS injection line. (Case 2). ~~Considering the results of sensitivity analyses, the analysis assumes a combined simultaneous loss of normal AC power that occurs at event initiation, an inadvertent actuation block (IAB) release pressure of 1000 psid, conservatively biased ECCS actuation setpoint, fine CNV axial volume and radial CNV heat structure nodalization, RPV noncondensable release, and the single failure of one RRV to open.~~Considering the results of sensitivity analyses, the analysis assumes a combined simultaneous loss of normal AC power that occurs at event initiation, an inadvertent actuation block (IAB) release pressure of 1000 psid, conservatively biased ECCS actuation setpoint, fine CNV axial volume and radial CNV heat structure nodalization, no RPV noncondensable release, and the single failure of one RRV to open. The peak calculated pressure and temperature are listed in Table 6.2-2, which provide sufficient margin to CNV design pressure and temperature.~~is 959 psia, providing a margin of 91 psia to the CNV design pressure of 1050 psia. The peak calculated temperature is 526 degrees F, providing a margin of 24 degrees F to the CNV design temperature of 550 degrees F.~~

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-18S1, RAI 06.02.01.01.A-18S3, RAI 06.02.01.01.A-19

The overall limiting peak calculated containment pressure, based on the mass and energy release spectrum analyses, is postulated to occur as the result of the spurious opening of a RRV anticipated operational occurrence (Case 5). The analysis models an expansion of the RCS fluid into the CNV volume and includes all relevant energy input from RCS, secondary and fuel stored energy sources, along with conservatively modeled core power and decay heat. Additional assumptions accounting for the results of sensitivity analyses~~,~~ include the loss of both the normal AC power and the highly reliable DC power system (EDSS) postulated to occur at event initiation, an IAB release pressure of 1000 psid for the second RRV, and 900 psid for the RVVs, fine CNV axial volume and fine radial CNV heat structure nodalization, fine reactor pool nodalization, fast RPV noncondensable release rate, and minimum primary system flow.~~include the loss of normal AC power and highly reliable DC power system (EDSS) postulated to occur at event initiation and an inadvertent actuation block (IAB) release pressure of 1000 psid, fine CNV axial volume and radial CNV heat structure nodalization, fine reactor pool nodalization, RPV noncondensable release, minimum primary system flow, and single failure of one RRV to open.~~ The peak calculated pressure is ~~986~~994 psia, providing a ~~64~~56 psia margin to the CNV design pressure of 1050 psia. Reference 6.2-1, Section 5.4 discusses the analytical and design margin incorporated into the CNV design.

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-18S3, RAI 06.02.01.01.A-19

The peak calculated containment pressure resulting from a secondary side mass and energy release is postulated as the result of a double-ended steam line break inside containment. The analysis assumptions are documented by Reference 6.2-1, Section 5.2.~~assumes fine CNV axial volume and radial CNV heat structure nodalization, fine reactor pool nodalization, an IAB release pressure of 1200 psid, low RCS flow and a failure of the associated FWIV to close.~~ The peak calculated pressure is listed in Table 6.2-2.~~449 psia.~~

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-19

The peak calculated containment temperature resulting from a secondary side mass and energy release is postulated as the result of a double-ended steam line break inside containment. The analysis assumes normal AC and DC power available, and a failure of the associated feedwater isolation valve (FWIV) to close. The peak calculated temperature is listed in Table 6.2-2.~~433 degrees F.~~

The secondary system mass and energy release event results are bounded by the primary system mass and energy release events.

The CNV external design pressure is 60 psia which is based on an internal pressure of 0 psia and an external pressure resulting from 100 feet of pool water static pressure.

The environmental qualification of mechanical and electrical equipment exposed to the containment environment following a primary or secondary system mass and energy release inside containment is discussed in Section 3.11.

The CNV instrumentation provided to monitor and record the required containment parameters and the capability to operate in post-accident environments is discussed in Chapter 7 and Section 3.11.

6.2.1.2 Containment Subcompartments

A sub-compartment design basis and supporting analysis for mass and energy release is not relevant to the NuScale CNV because the CNV has no interior subcompartments.

6.2.1.3 Mass and Energy Release Analyses for Primary System Release Events

The NuScale containment receives the primary system mass and energy released following a postulated rupture of piping containing reactor coolant, inadvertent opening of an ECCS or RSV. The CNV response analysis methodology, described in technical report TR-0516-49084,~~–~~ “Containment Response Analysis Methodology Technical Report”~~, Rev 0~~ (Reference 6.2-1), is an extension of the NuScale LOCA evaluation model that was developed in accordance with the guidance of Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods" December 2005. Reference 6.2-1 identifies and justifies differences in the containment response methodology, in comparison to the LOCA evaluation model.

The containment response analysis methodology is based on the NRELAP5 system thermal-hydraulic code with appropriate initial and boundary conditions. The NRELAP5 code is a NuScale modified version of the RELAP5-3D Version 4.1.3 code. The NRELAP5 code is used for LOCA transient analyses. The NRELAP5 code has been qualified or assessed to separate and integral effects tests as described by Reference 6.2-2 to demonstrate the capability of the code to model LOCAs in the NPM. The LOCA evaluation model qualification activities in Reference 6.2-2 are adequate to demonstrate the capability of the NRELAP code to model containment response to LOCA events. Modifications to the inputs to the code are incorporated in order to predict maximum containment peak pressures and temperatures for the various event scenarios by conservatively maximizing the mass and energy release while minimizing the performance of containment heat removal.

The LOCA analysis break spectum modeled by the NuScale containment response analysis methodology considers the same break spectrum as considered by the LOCA evaluation model (Reference 6.2-2). The maximum break opening area is modeled. The inadvertent ECCS valve opening event analysis uses the largest valve area. Break locations are chosen such that mass and energy releases to containment are maximized. The dominant consideration is the timing of the second mass and energy release that occurs when the ECCS valves open. Sensitivity studies are performed on the timing of the ECCS valve opening, and this is effectively a break size sensitivity combined with the maximum initial mass and energy release for each break location. This approach assures that the limiting cases have been identified for the peak CNV pressure and peak CNV temperature.

Critical flow is evaluated using the Henry-Fauske and Moody models for subcooled and two-phase flow conditions as discussed by Reference 6.2-1. The maximum valve area and Cv values (RVV and RRV) are used to determine ECCS valve flows. Smaller values containment pressure and temperature have been completed, and a gradual depressurization and cooling phase begins.

Sensitivity cases are performed to determine the effect of loss of power (AC or DC) scenarios, as well as postulated single failures, on the primary system mass and energy release scenarios considered by the NuScale containment response analysis methodology. The insights obtained from the results of the sensitivity cases, discussed in Reference 6.2-1, are used to determine the limiting cases for CNV pressure and temperature.

6.2.1.3.1 Mass and Energy Release Data - Primary System Release Events

The maximum containment peak pressure and peak temperature scenarios are determined by conservatively modeling the mass and energy release and minimizing the performance of the containment heat removal function of containment.

Reference 6.2-1, Section 5.1 provides ~~the~~ results of ~~the~~ NRELAP5 ~~limiting~~ analyses of the spectrum of the five primary system mass and energy release scenarios for the NPM. The limiting primary system release event (Case 5) CNV pressure results are depicted by figures contained in Reference 6.2-1, Section 5.1. Graphical results for the limiting CNV pressure, temperature and mass and energy release rates are shown in Figure 6.2-9 through Figure 6.2-14. The limiting peak pressure and temperature results are below the CNV design pressure and temperature.

6.2.1.3.2 Energy Sources - Primary System Release Events

The containment response analysis methodology models available energy sources identified by 10 CFR Part 50, Appendix K, paragraph I.A, with the exception of energy associated with fuel clad metal-water reaction, since calculated cladding temperatures for design basis LOCAs remain below the threshold for cladding oxidation. Energy sources addressed in the containment response analysis analyses include

- core power initialized at 102 percent of rated thermal power (163.2 MW).
- decay heat modeled using the 1979 ANS standard decay heat model with a 1.2 multiplier.
- RCS stored energy based on conservative initial conditions of pressure, average RCS temperature and pressurizer level that consider the normal operating range including instrumentation uncertainties and deadband.
- stored energy in vessel internal structures.
- RCS piping inside containment.
- stored fuel energy.
- stored secondary energy (steam generator (SG) tubes, main steam and feedwater piping inside containment) based on conservative initial conditions of steam pressure and feedwater temperature that consider the normal operating range including instrumentation uncertainties and deadband.

The containment response analysis methodology for secondary system events is based on the NRELAP5 system thermal-hydraulic code with appropriate initial and bounding conditions. The NRELAP5 code is a NuScale modified version of the RELAP5-3D Version 4.1.3 code. The NRELAP5 code is used for non-LOCA transient analyses. The NRELAP5 non-LOCA model described in the non-LOCA transient analysis methodology report (Reference 6.2-4) is used to develop the MSLB and FWLB analyses in the containment response analysis methodology. Modifications to the inputs to the code are incorporated in order to predict maximum containment peak pressures and temperatures for various event scenarios accomplished by conservatively maximizing the mass and energy release while minimizing the performance of containment heat removal.

The limiting MSLB event and FWLB event are double-ended ruptures of the largest main steam line and feedwater line pipes.

Secondary system mass and energy releases consist of the MSLB and FWLB events with the asymmetric responses in SGs included. The affected SG blows down into the CNV and the feedwater supply and main steam lines are isolated.

Conservative modeling of secondary system mass and energy release scenarios ensures a bounding analysis. All breaks are considered with a maximum break size at each location. Critical flow is evaluated using the Moody and Henry-Fauske models for subcooled and two-phase flow conditions as discussed in Reference 6.2-1.

The containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code for secondary system pipe break analysis. The Non-LOCA transient analysis methodology report demonstrates these correlations are applicable to the NPM design (Reference 6.2-4). The local fluid conditions and the local heat structure surface temperatures determine the heat transfer mode. Nucleate boiling heat transfer is included in the code and is selected if the local conditions are appropriate. For the helical coil SG, other heat transfer modes exist as the coolant enters as subcooled liquid and exits as superheated steam. Initial and boundary conditions are selected to maximize containment pressure and temperature response.

A description of each postulated secondary system mass and energy release event is provided in Reference 6.2-1. Results of the limiting analyses are provided in Reference 6.2-1~~ and for FWLBs~~. The secondary system mass and energy analyses are fully bounded by the primary system limiting events.

6.2.1.4.1 Mass and Energy Release Data - Secondary System

Similar to primary system mass and energy release scenarios, the maximum containment peak pressure and peak temperature scenarios for secondary system releases into containment are determined by conservatively modeling the mass and energy release and minimizing the performance of the heat removal function of containment.

operating, maintenance, testing, and postulated accident conditions over its 60-year design life. Section 6.2.1 provides additional detail.

For ferritic materials classified as pressure-retaining components of the RCPB, the requirements of ASME BPVC Section XI Appendix G apply.

The NuScale ferritic containment pressure boundary conforms to ASME BPVC, Section II material specifications and meets the fracture toughness criteria and testing requirements identified in ASME BPVC Section III, Division 1, NB-2300 for containment pressure boundary components fabricated of ferritic materials. Fracture prevention of the containment pressure boundary is assured.

RAI 06.01.01-8S1, RAI 06.01.01-9S1

The Alloy 718 heat treatment requirements discussed in Section 3.13.1.1 are applied to the Alloy 718 CNV main closure flange studs.

Portions of the lower CNV have 60-year design fluence in excess of 1E+17 neutrons/cm^2, E > 1 MeV, with the peak fluence in the lower CNV not exceeding 5.5E+18 neutrons/cm^2, E > 1 MeV. To remove neutron embrittlement concerns in the CNV, the lower CNV is made of austenitic stainless steels which are more resistant to neutron embrittlement than ferritic materials.

6.2.8 References

6.2-1 NuScale Power, LLC, "Containment Response Analysis Methodology Technical Report," TR-0516-49084, Rev. 2~~0~~.

6.2-2 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Rev. 0~~, December 2016~~.

6.2-3 NuScale Power, LLC, "Combustible Gas Control," TR-0716-50424, Rev. ~~0~~1.

6.2-4 NuScale Power, LLC, "Non-Loss-of-Coolant Transient Analysis Methodology Report," TR-0516-49416, Rev. 1~~0, January 2017~~.

6.2-5 NuScale Power, LLC, "Long-Term Cooling Methodology," TR-0916-51299, Rev. 1~~, August 2019~~.

6.2-6 NuScale Power, LLC, "NuScale Containment Leakage Integrity Assurance," TR-1116-51962, Rev. 1~~0, December 2016~~.

6.2-7 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section XI Division 1, "Rules for Inservice Inspection of Nuclear Components," ~~2013 edition, Section XI Division I,~~ New York, NY.

RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-18S3, RAI 06.02.01.01.A-19

Table 6.2-2: Containment Response Analysis Results[3]

Event Description	Case Description	CNV Pressure (psia)	CNV Wall Temperature (°F)
RCS Discharge Break	Base Case	705	492
RCS Discharge Break	Limiting Sensitivity Case Results	943	510
RCS Injection Line Break	Base Case	894	514
RCS Injection Line Break	Limiting Sensitivity Case Results	959	526[2]
RPV High Point Vent Degasification Line Break	Base Case	554	471
RPV High Point Vent Degasification Line Break	Limiting Sensitivity Case Results	901	489
Inadvertent RVV Actuation	Base Case	856	483
Inadvertent RVV Actuation	Limiting Sensitivity Case Results	911	486
Inadvertent RRV Actuation	Base Case	941	492
Inadvertent RRV Actuation	Limiting Sensitivity Case Results	~~986~~994[1]	512
Main Steam Line Break	Limiting Results	449	433
Feedwater Line Break	Limiting Results	416	408

[1] Limiting NPM primary/secondary release event peak pressure, includes IAB operating range sensitivity.

[2] Limiting NPM primary/secondary release event peak temperature.

[3] Results reflected in this Table do not consider the impact of sensitivity studies performed to address a revised IAB operating range as discussed by Reference 6.2-1, Section 5.1.1, except as stated in Note 1.

Table 6.2-7: Sequence of Events - Peak Containment Vessel Pressure Case

Event	**Time (s)**
Transient initiated by an inadvertent RRV #1 opening into the CNV	0
High CNV pressure analytical limit is reached	0.4
Reactor trip	2
The upper end of the IAB release pressure range is reached (1000 psid) and RRV #2 opens	75
The lower end of the IAB release pressure range is reached (900 psid) and the RVVs open	84
Peak CNV internal pressure is reached (994 psia)	99
Peak CNV wall temperature is reached (for peak pressure case)	616
CNV drops to less than 50% of peak pressure value	Approximately 1800

Table 6.2-8: Sequence of Events - Peak Containment Vessel Temperature Case

Event	Time (s)
Transient initiated by a CVCS injection line break into the CNV	0
High CNV pressure analytical limit is reached	3
Reactor trip	5
High containment level actuation limit is reached	952
All ECCS valves open	955
Peak CNV internal pressure is reached for peak CNV temperature case	967
Peak CNV wall temperature is reached (526 °F)	978
CNV pressure drops to less than 50% of peak pressure	Approximately 2500

Figure 6.2-9: Maximum Containment Internal Pressure - Peak Containment Vessel Pressure Case

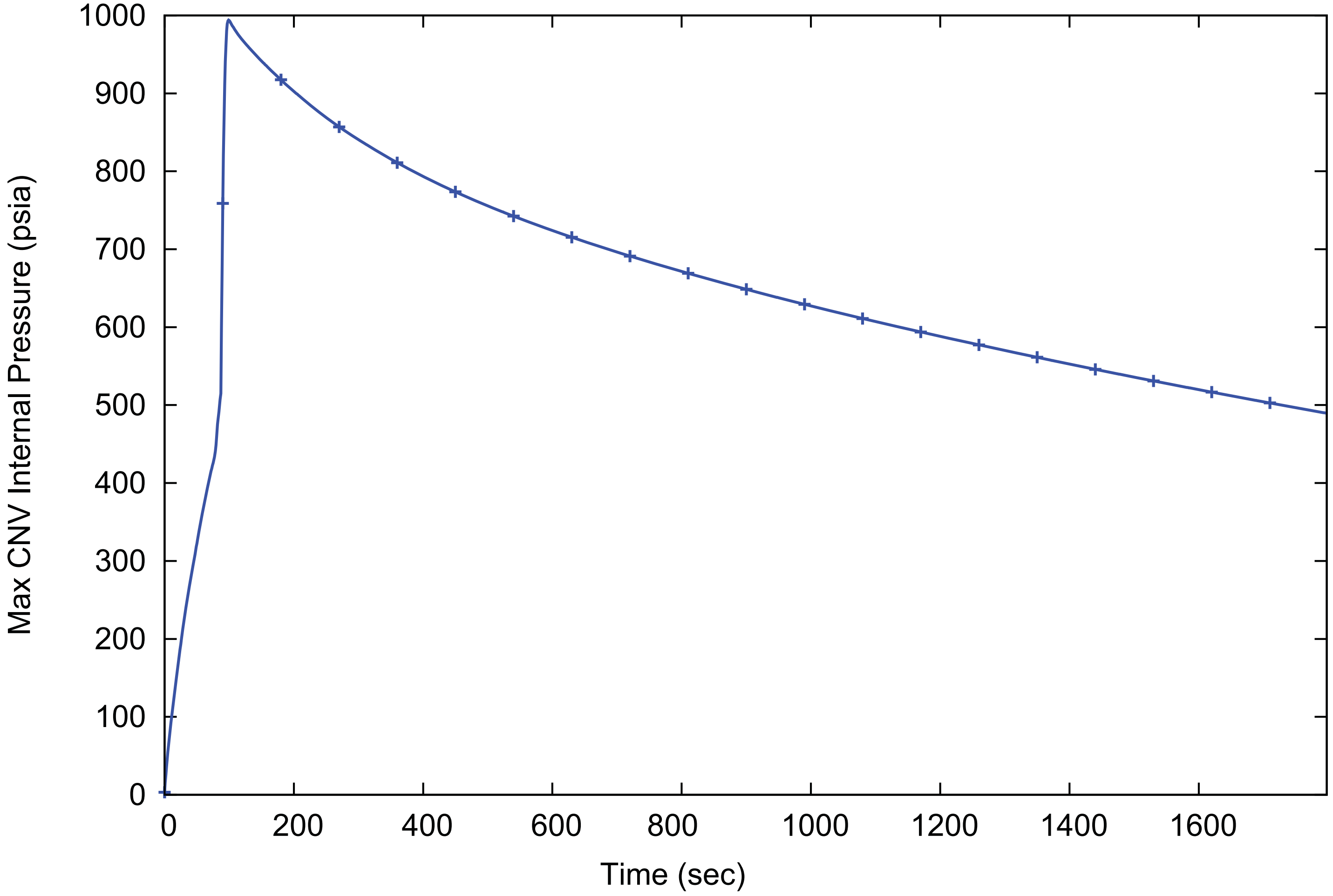


Figure 6.2-10: Break and Emergency Core Cooling System Mass Release Rate - Peak Containment Vessel Pressure Case

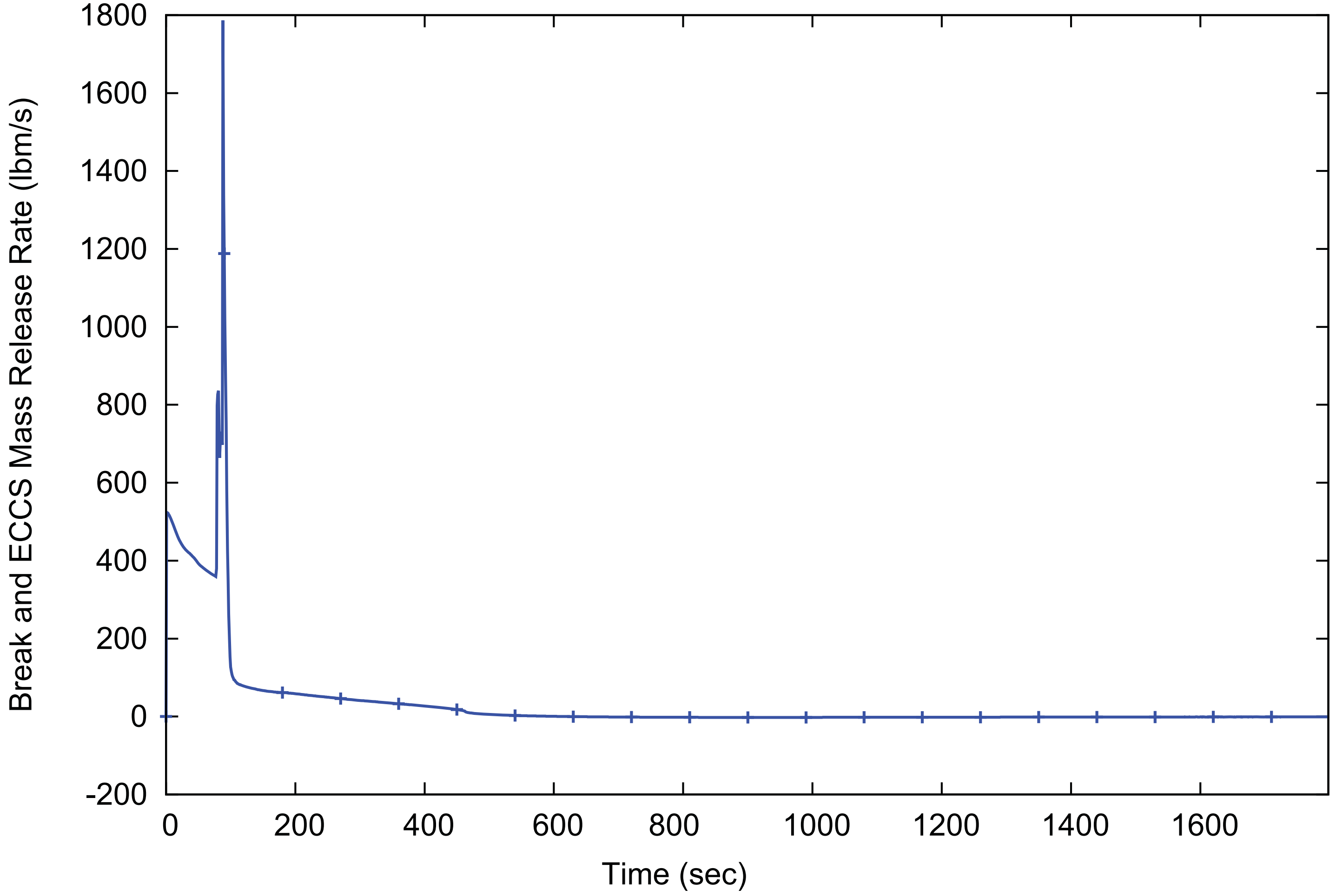


Figure 6.2-11: Break and Emergency Core Cooling System Energy Release Rate - Peak Containment Vessel Pressure Case

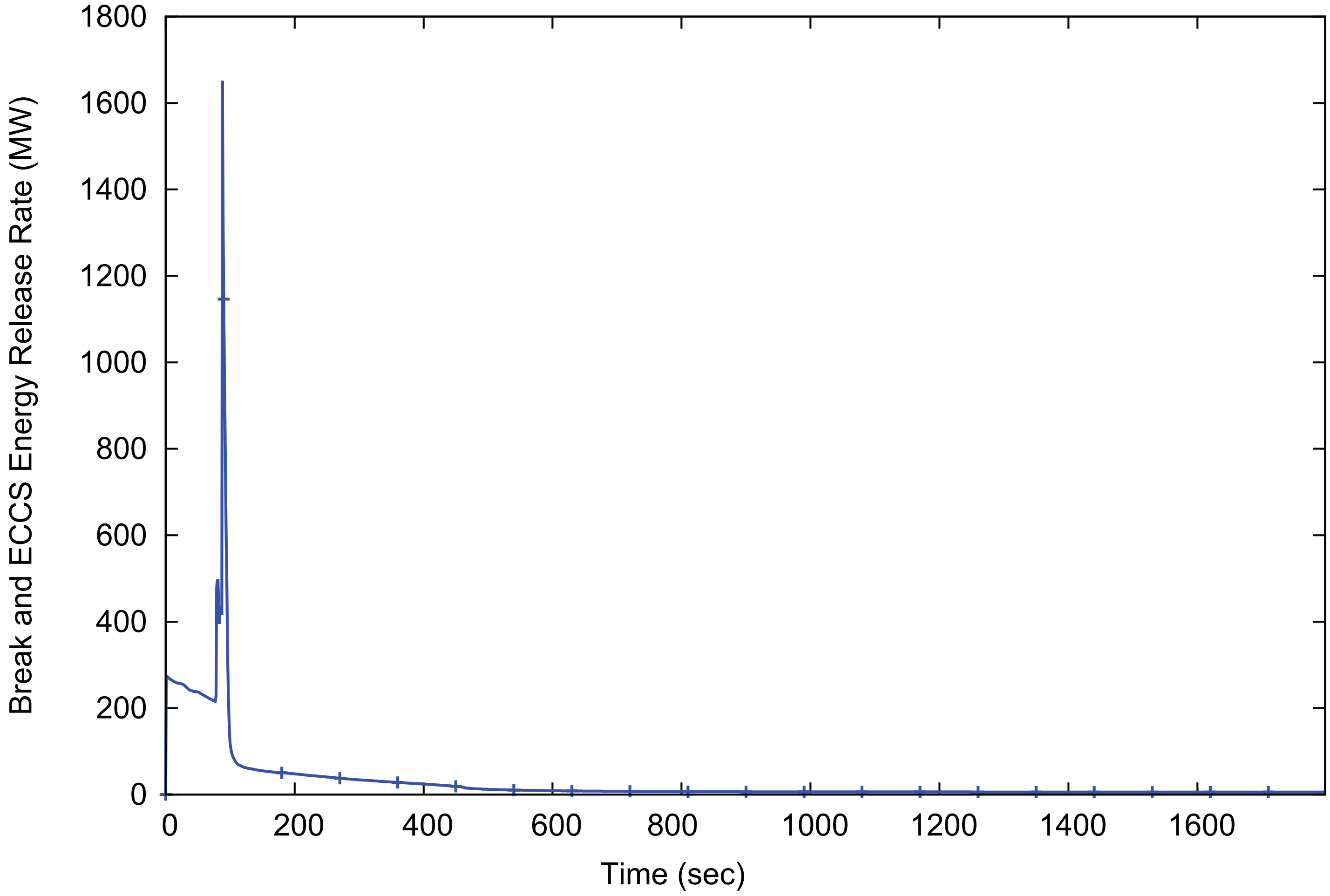


Figure 6.2-12: Maximum Containment Wall Temperature - Peak Containment Vessel Temperature Case

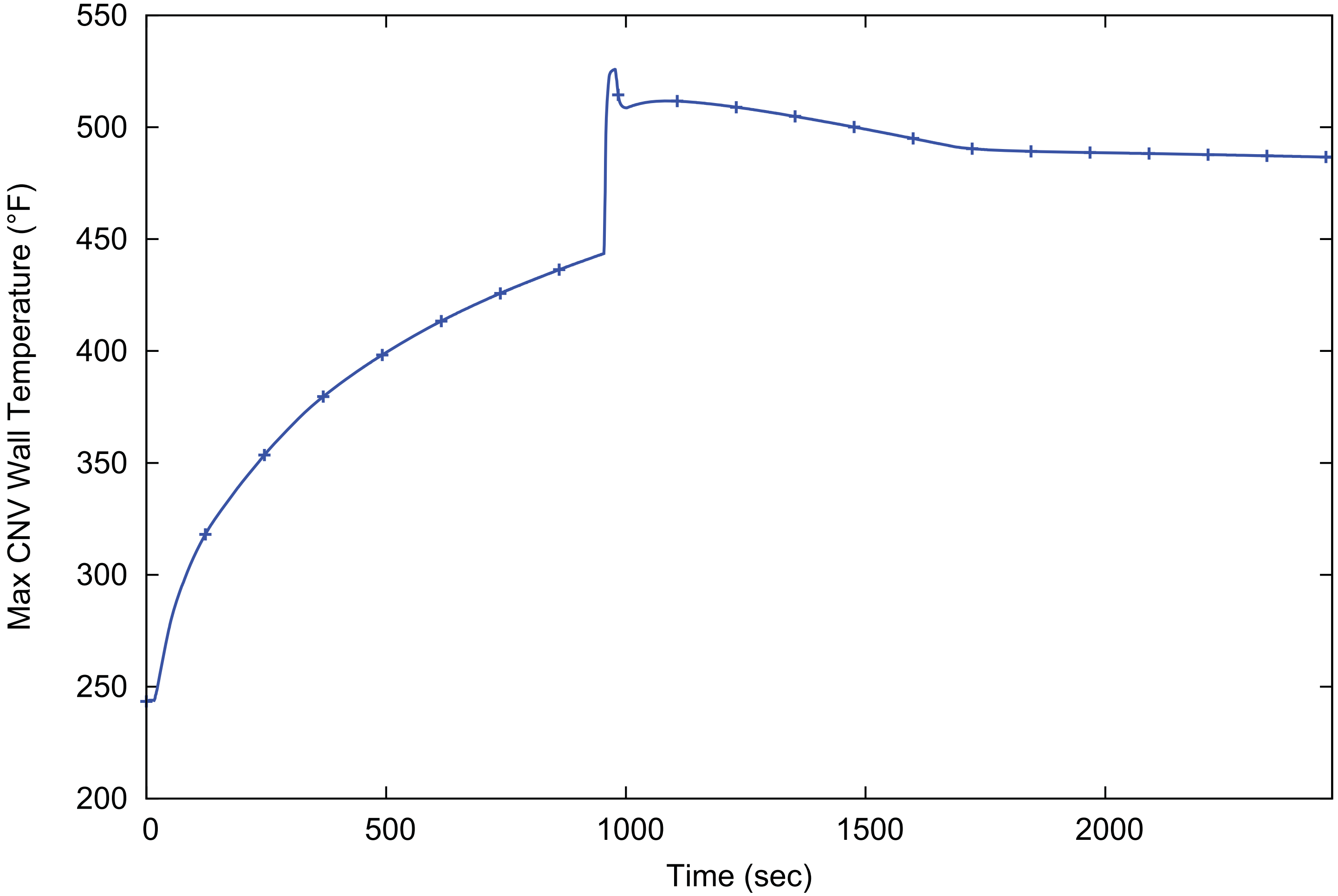


Figure 6.2-13: Break and Emergency Core Cooling System Mass Release Rate - Peak Containment Vessel Temperature Case

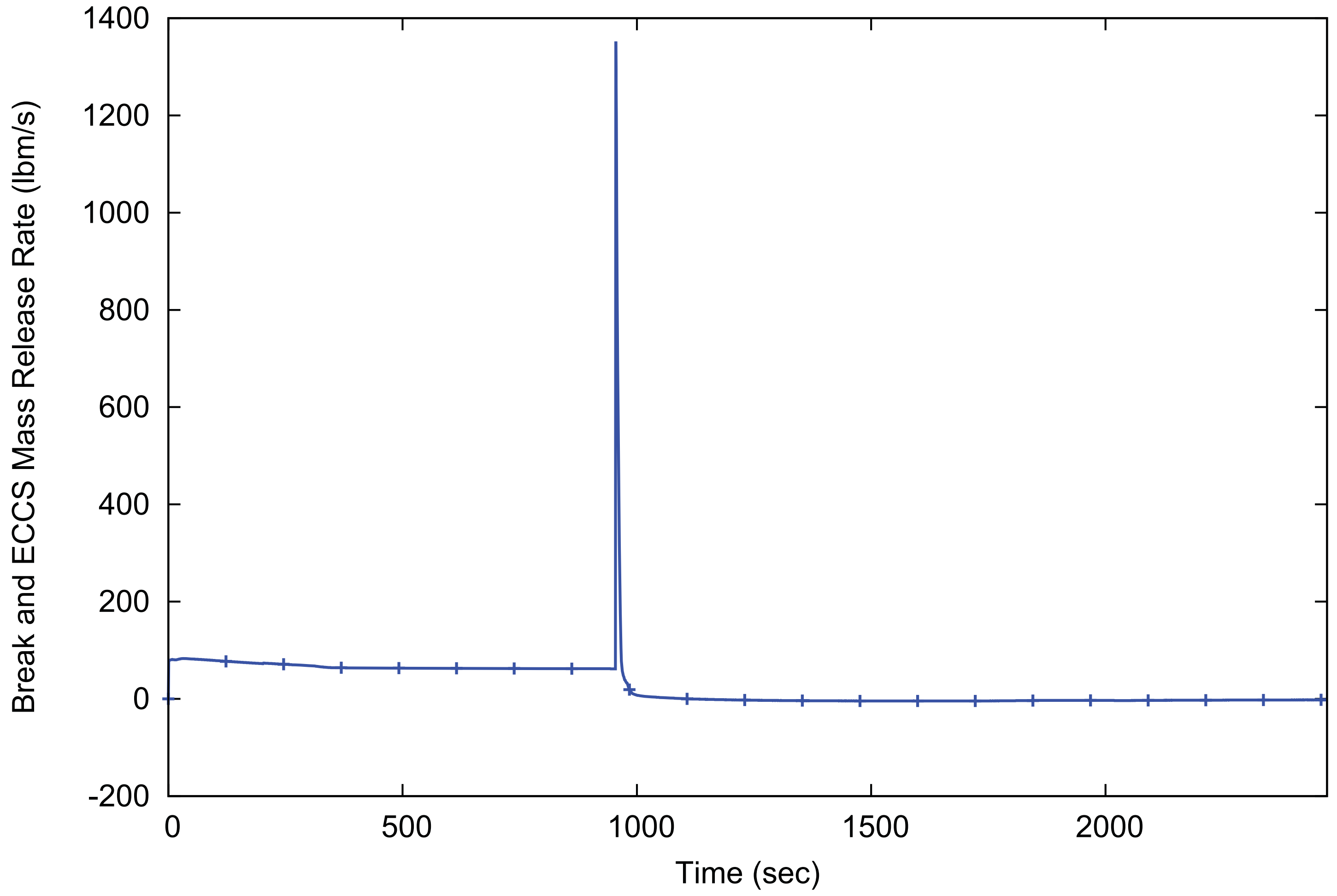
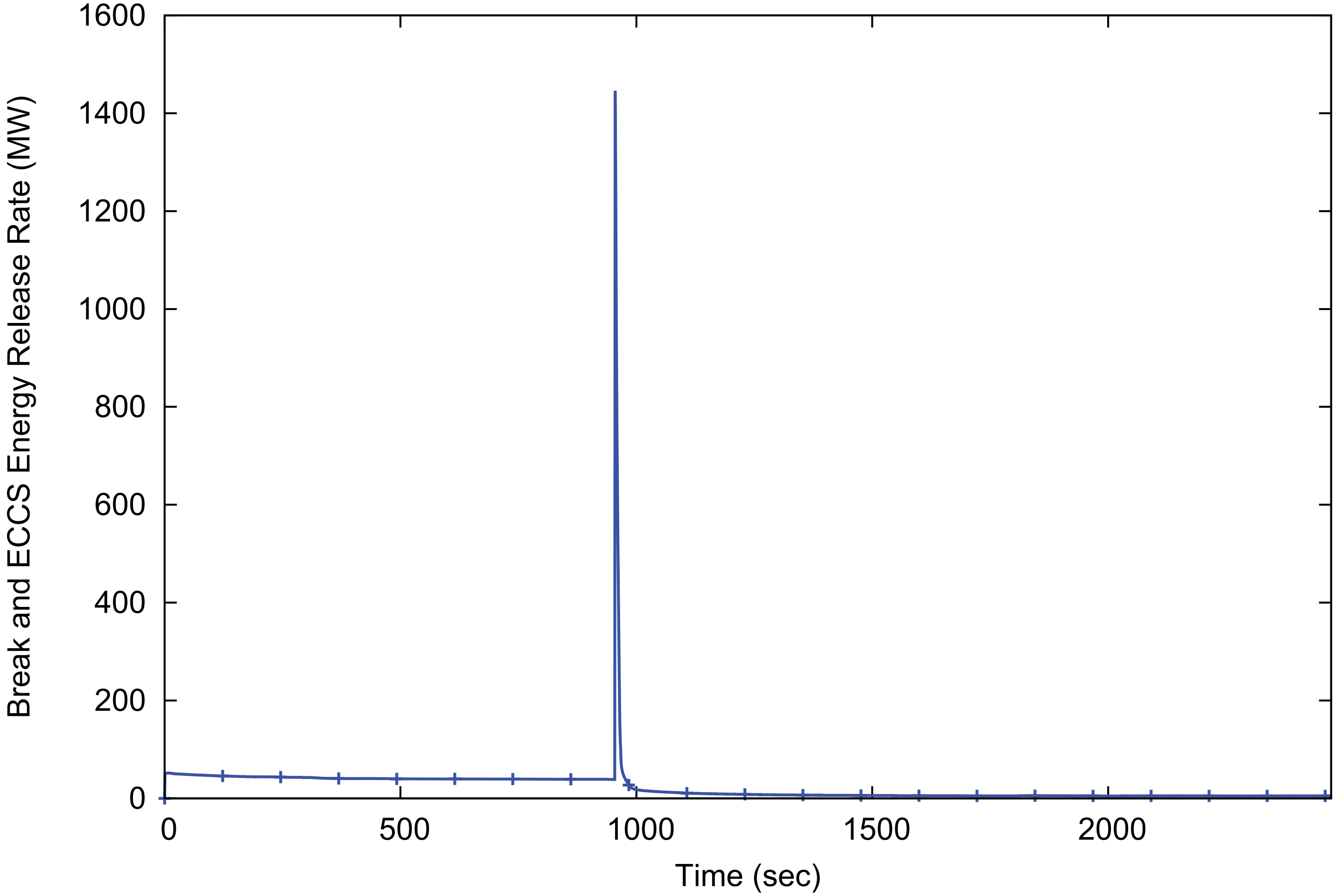


Figure 6.2-14: Break and Emergency Core Cooling System Energy Release Rate - Peak Containment Vessel Temperature Case



Executive Summary

This report presents the NuScale Power, LLC, (NuScale) methodology used to analyze the mass and energy release into the containment vessel (CNV) for the spectrum of design basis transients and accidents, and the resulting pressure and temperature response of the CNV. The NuScale Power Module (NPM) limiting peak pressure and temperature results determined using the methodology are presented.

The containment response analysis methodology uses the NRELAP5 thermal-hydraulic code, which is a NuScale-modified version of the RELAP5-3D© v 4.1.3 code used for loss-of-coolant accident (LOCA) and non-LOCA transient and accident analyses, including the response of the CNV.

The NRELAP5 model used to model NPM performance for primary system LOCA and emergency core cooling system valve-opening event analyses is similar to the model used in the LOCA evaluation model, described by Reference 7.2.1. The NRELAP5 model used for secondary system pipe-break analysis in the containment response analysis methodology is similar to the non-LOCA model described by the Non-LOCA Evaluation Model Report (Ref: 7.2.2). Changes made to these models that maximize containment pressure and temperature response to primary and secondary system release events are described in this report. These changes conservatively maximize the mass and energy release and minimize the performance of the containment heat removal system and are consistent with acceptance criteria given by Design Specific Review Standard Section 6.2.1.3 (Ref: 7.1.6) and Design Specific Review Standard Section 6.2.1.4 (Ref: 7.1.7).

Initial and boundary conditions for the spectrum of primary system release containment response analyses and secondary system pipe break analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. These initial and boundary conditions are described in this report, along with the rationale for their selection.

The results of the NRELAP5 limiting analyses using the containment response analysis methodology are presented in this report. These analyses cover the spectrum of primary system mass and energy release scenarios for the NPM, and secondary system pipe break scenarios.

The limiting LOCA peak pressure and CNV wall temperature are a result of the reactor coolant system (RCS) injection line break. The LOCA limiting peak CNV wall temperature is approximately 526 degrees F and it results from a reactor coolant system injection line break case, with a loss of normal alternating current (AC) power. The LOCA limiting peak internal pressure is approximately 959 psia, which results from a reactor coolant system injection line break case with a loss of normal AC and direct current (DC) power. The LOCA event peak CNV pressure is below the CNV design pressure of 1050 psia. The LOCA peak CNV pressure and wall temperature bound the main steamline break (MSLB) and feedwater line break (FWLB) results.

The overall limiting peak CNV accident pressure is approximately ~~986~~994 psia, which is approximately 5~~6~~ percent below the containment design pressure of 1050 psia. It results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power, considering an inadvertent actuation block (IAB) release pressure

range of 950 psi +/- 50 psi. The CNV pressure for this limiting case is reduced to below 50 percent of the peak value in less than 2 hours, demonstrating adequate NPM containment heat removal.

Section 5.4 discusses margin in the NPM design that is not included in the CNV design pressure rating or modeled in the containment response analyses. Design factors conservatively not credited include atmospheric pressure acting against the CNV exterior surface and the availability of the decay heat removal system (DHRS).

The containment response analysis methodology demonstrates that the NPM design has adequate margin to design limits and that it satisfies the requirements of General Design Criteria (GDC) 16, 50, and Principal Design Criterion (PDC) 38.

will not exceed these profiles anywhere within the specified environmental zones, except in the break zone).	

2.1.3.2 Design Specific Review Standard 6.2.1.1.A Containment

The DSRS Section 6.2.1.1.A, "Containment" (Ref: 7.1.5), includes content related to containment design, including some elements that are associated with the capability to withstand M&E releases. The comparison of the containment response analysis methodology to applicable content in DSRS Section 6.2.1.1.A is provided in Table 2-2:

Table 2-2 Compliance with Design Specific Review Standard Section 6.2.1.1.A

DSRS Section 6.2.1.1.A, p. 1	**Containment Response Analysis Methodology**
The temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (LOCAs) (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks	The containment response analysis methodology includes the spectrum of primary release events resulting from postulated limiting breaks (LOCAs) and valve openings, MSLB accidents, and FWLB accidents. The limiting results are less than the CNV design pressure and temperature.
The effectiveness of static (passive) and active heat removal mechanisms.	The containment response analysis methodology includes conservative modeling of passive heat removal systems (there are no active heat removal systems in the NuScale design). Specifically, conservatisms are employed in conservative assumed initial and boundary conditions, including the reactor pool to ensure a bounding peak CNV peak pressure and temperature following events involving release of mass and energy into the CNV. The performance of these systems is shown to be effective in limiting the CNV pressure and temperature response to within acceptable design limits. Conservatism in initial and boundary conditions is discussed in Section 3.5
DSRS Section 6.2.1.1.A, p. 4	Containment Response Analysis Methodology
To satisfy the requirements of GDC 16 and 50 regarding sufficient design margin, for plants in the design stage (i.e., at the construction permit (CP) or design certification (DC) stage) of review, the containment design pressure should provide at least a 10% margin above the accepted peak calculated containment pressure following a LOCA, or a steam	For the NuScale FSAR submittal, the results of the containment response analysis methodology for the limiting event scenarios are less than the CNV design pressure and temperature. The overall limiting peak CNV accident pressure is approximately ~~986~~994 psia, which is approximately ~~6~~5 percent below the containment design pressure of 1050

The results of the LOCA scenario PIRT are directly applicable to the primary system M&E release and resultant CNV pressure and temperature response that are the focus of the containment response methodology. The basis for this statement is that “CNV pressure and temperature” is a figure-of-merit in the LOCA phenomena identification and ranking table. Therefore, the LOCA scenario PIRT is also considered to be the LOCA containment response analysis methodology PIRT.

3.3.1.2 Module Response

The typical response of the NPM to a primary system M&E release is characterized by a simultaneous depressurization of the primary system and pressurization of the CNV. The module response depends on the size of the break or valve opening, the location of the release as that determines if the release is steam or liquid or two-phase, and the timing of the M&E releases. The resulting high containment pressure signal causes an immediate actuation of the following safety features:

- containment isolation, including
  - closure of MSIVs
  - closure of FWIVs
  - closure of backup MSIVs (non-safety)
  - closure of FWRVs (non-safety)
- reactor trip
- turbine trip

Any steam that is released through the break or valve condenses on the cold inner surface of the CNV. Condensate and any unflashed break liquid accumulates into a pool on the bottom of the CNV. The primary system level decreases due to the break or valve flow. The ECCS actuates on the following conditions:

- high CNV level
- loss of normal AC power and the highly reliable DC power system

The following design criteria govern RVVs and RRVs opening:

- If the pressure differential across the valves is greater than the IAB threshold when the ECCS signal actuates, then the valves stay closed until the pressure differential decreases to below the IAB release pressure
- If the pressure differential across the valves has decreased to below the IAB threshold pressure when the ECCS signal actuates, then the valves open and the IAB release pressure is not used. As discussed in FSAR Section 6.3.2.2, the threshold pressure for IAB operation to prevent spurious opening of the main ECCS valve is 1300 psid. Therefore, the IAB prevents the main valve from opening for all reactor pressures 1300 psid and greater, with respect to containment. Given an initial IAB block, the IAB

releases at 950 psid +/- 50 psi once reactor pressure is reduced. The IAB does not prevent the main valve from opening for initial pressures of 900 psid and below.

Opening of the RVVs increases the depressurization rate, and the primary system and CNV pressures approach equalization. As the pressures equalize, the break/valve flow decreases. With pressure equalization and the increase in the CNV pool level, flow through the RRVs into the reactor vessel starts to provide long-term core cooling via recirculation. This terminates the reactor vessel level decrease prior to core uncovery. Heat transfer to the CNV wall and to the reactor pool eventually exceeds the energy addition from the break flow and the RVV flow. When this occurs the period of peak containment pressure and temperature have been completed, and a gradual depressurization and cooling phase begins.

3.3.1.3 Event Scenarios and Break Spectrum

The postulated primary system M&E release events include the following pipe break accidents and valve actuations. For the valve opening events, the specific FSAR events that result in actuation of that valve are listed.

- Pipe breaks (LOCAs)
  - FSAR 15.6.5 - RCS discharge line break LOCA {{ }}$^{2(a),(c)}$
  - FSAR 15.6.5 - RCS injection line break LOCA {{ }}$^{2(a),(c)}$
  - FSAR 15.6.5 - Pressurizer spray supply line break LOCA {{ }}$^{2(a),(c)}$
  - FSAR 15.6.5 – RPV high point degasification line LOCA {{ }}$^{2(a),(c)}$
- RSV actuation {{ }}$^{2(a),(c)}$
  - FSAR 15.6.1 – Inadvertent RSV opening
- RVV actuation ({{ }}$^{2(a),(c)}$)
  - FSAR 15.6.6 – Inadvertent RVV opening
- RRV actuation {{ }}$^{2(a),(c)}$
  - FSAR 15.6.6 – Inadvertent RRV opening

The RPV high point degasification line, the pressurizer spray supply line, and the RSVs are all located near the top of the RPV. A LOCA in the RPV high point degasification line is the largest break size in this location and is analyzed in the containment response analysis methodology. The other two are non-limiting and are not analyzed.

One RVV or one RRV can open as an initiating event due to an assumed mechanical failure. The RVVs and RRVs all open following ECCS signal actuation and when the IAB design criteria discussed in Section 3.3.1.2 are met .

The RPV high point degasification line break LOCA differs in that the break flow will be steam. The RCS break locations differ in that the discharge line connects to the downcomer, and the injection line connects to the riser. These three break locations plus the valve opening event locations fulfill the adequacy of the break spectrum with regard to location.

The adequacy of the break spectrum with regard to break size is important in the timing of the ECCS valve opening, as the second M&E release resulting from the opening of the three RVVs is the dominant event for CNV pressure and temperature response. First, the maximum break size at each location is analyzed to ensure the maximum initial M&E release rate into the CNV during the first phase of CNV pressurization. Then, the sensitivity of the opening time of the three RVVs is addressed by analysis of a range of IAB release pressures for each break location. In this manner a lower IAB release pressure results in a delay in the RVV opening time. This is similar to a break size sensitivity because a range of break sizes would result in a range of depressurization rates and RVV opening times. However, by using the maximum break size for all cases the maximum initial M&E release rate is used for all cases. This approach fulfills the adequacy of the break spectrum with regard to break size.

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}}$^{2(a),(c)}$

In summary, the limiting postulated primary system M&E release scenarios consist of an initiating anticipated operational occurrence or accident, which may include a pipe break or RVV or RRV valve opening, with a resultant an ECCS actuation signal causing all RVVs and RRVs to fully open after the IAB design criteria discussed in Section 3.3.1.2 are met. Table 3-3 shows the primary system M&E release scenarios that are used to determine the limiting cases.

5.0 Results

5.1 Primary System Release Scenario Containment Response Analysis

This section presents the results of the NRELAP5 limiting analyses of the spectrum of primary system M&E release scenarios for the NPM, listed in Table 3-3, and secondary system break scenarios that are determined using the containment response analysis methodology presented earlier in this report. The case labels from Table 3-3 are used in the following discussion.

5.1.1 Analysis Approach

The approach to determine the limiting peak CNV pressure event from the the spectrum of primary mass and energy release scenarios for the NPM, listed in Table 3-3, and the limiting peak CNV temperature for each primary release event was as follows:

{{

}}2(a),(c)

{{

}}2(a),(c)

The threshold pressure for IAB operation to prevent spurious opening of the main ECCS valve is 1300 psid. Therefore, the IAB prevents the main valve from opening for all reactor pressures 1300 psid and greater, with respect to containment. Given an initial IAB block, the IAB releases at 950 psid +/- 50 psi once reactor pressure is reduced. The IAB does not prevent the main valve from opening for initial pressures of 900 psid and below.

{{

}}2(a),(c)

5.1.2 Base Case~~Reference~~ Analysis and Sensitivity Results

The following insights were obtained from the results of the NRELAP5 analyses of the five primary system M&E release cases and associated sensitivity studies.

- The peak CNV pressure scenario is the RRV release (Case 5). The RRV mass and energy release causes an initial heatup and pressurization of the CNV, and then ECCS actuation results in a second M&E release with all three RVVs and second RRV opening that pressurizes the CNV to the highest peak pressure.
- The peak CNV wall temperature scenario is the CVCS injection line LOCA (Case 2). The break in this location combines a high temperature liquid initial M&E release followed by a high temperature M&E release through all three RVVs following an ECCS actuation signal.
- The sensitivity parameters have only a small effect on the peak CNV pressure and temperature results of the limiting cases. No single failures had a significant impact on the results for the limiting cases. The loss of power sensitivity that results in early ECCS actuation, and the IAB release pressure sensitivity that affects the timing of the opening of the ECCS valves, were the more important sensitivity parameters.
- {{

}}2(a),(c)

temperature response are shown in Figures 5-1 and 5-2. The CNV peak pressure is 705 psia for the base~~reference~~ case, and 943 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, ~~1200 psid~~adverse IAB release pressure, low-biased High CNV Level setpoint, fine CNV volume nodalization). The peak CNV wall temperature is 492 degrees F for the base~~reference~~ case, and 510 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse~~1200 psid~~ IAB release pressure, low-biased High CNV Level setpoint, fine CNV volume nodalization). The results of this case are adequately representative of the RCS discharge line break, although they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The effect of the IAB release pressure range of 950 psi +/- 50 psi, including effect of valves opening at different pressures within that range, has been evaluated. Case 1 is non-limiting and was confirmed to be non-limiting in comparison to the RRV opening event.

Table 5-2 Case 1 sequence of events - reactor coolant system discharge line break loss-of-coolant accident

Peak CNV Pressure Case Time (sec)	**Event**	**Peak CNV Temperature Case Time (sec)**
0	LOCA in RCS discharge line For peak pressure case • Loss of normal AC and DC power • FW/MS isolation • Reactor trip For peak temperature case • Same	Same
1	High CNV pressure resulting in For peak pressure case • Containment isolation For peak temperature case • Same	Same
92	ECCS actuation on IAB release pressure	Same
95	ECCS valve opening on IAB release pressure	Same
109	Peak CNV temperature reached: For peak pressure case: 510 °F For peak temperature case: Same	Same
112	Peak CNV pressure is reached: For peak pressure case: 943 psia For peak temperature case: Same	Same
~1900	CNV pressure decreases to <50% of peak pressure	Same

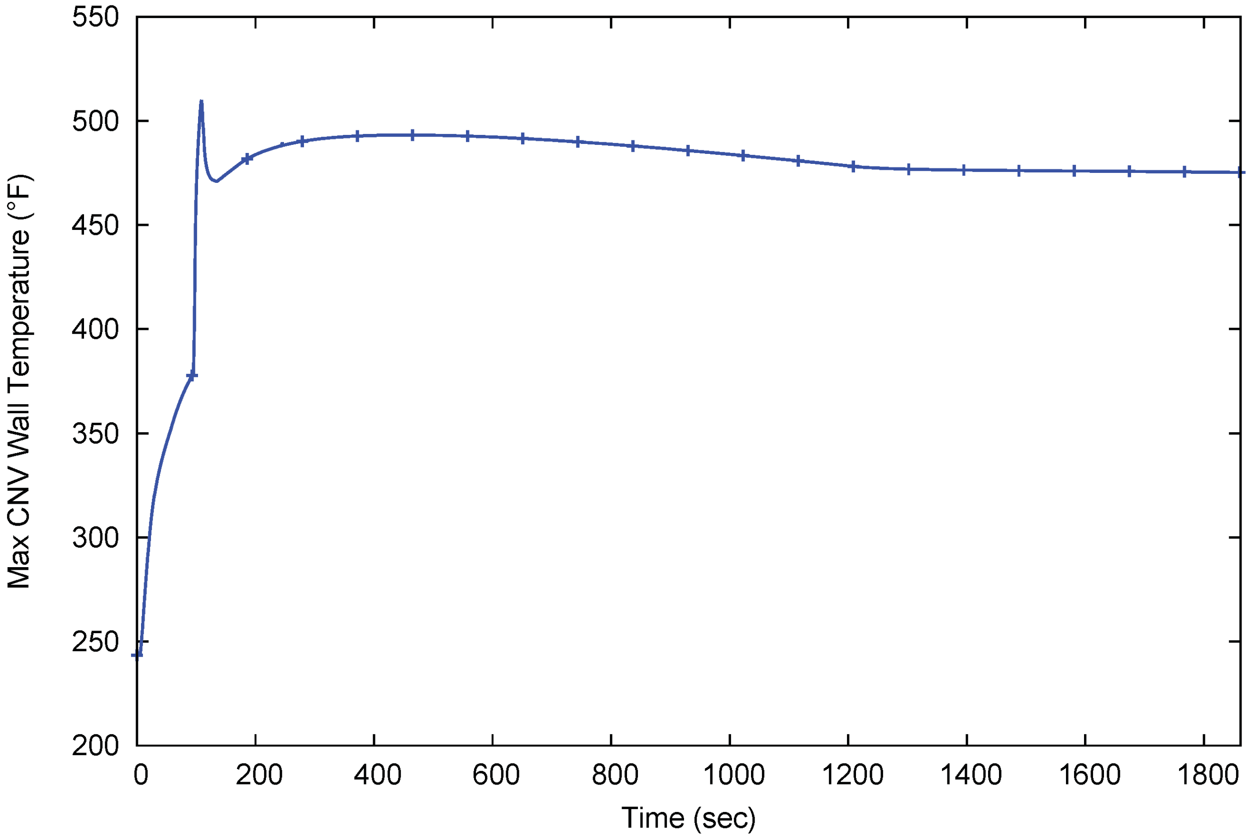

Figure 5-2 Case 1 containment vessel wall temperature - reactor coolant system discharge line break loss-of-coolant accident

5.1.3.2 Case 2: Limiting Loss-of-Coolant Event - Reactor Coolant System Injection Line Break Loss-of-Coolant Accident

The LOCA in the RCS injection line initiates an M&E release from the riser into the CNV. The results of the primary release event M&E release break spectrum analysis and sensitivity analyses have determined that Case 2 is the limiting LOCA peak pressure and overall limiting CNV wall temperature event. In addition, the analyses have shown that the Case 2 peak pressure results and CNV wall temperature results are ~1.7 and ~3.1 percent higher, respectively, than the next highest result (Case 1); therefore, there is confidence that the overall limiting break location and scenario has been identified. The sequence of events is shown in Table 5-3 and detailed results for key parameters are shown in Figures 5-3 through 5-16. The peak CNV wall temperature is 514 degrees F for the base~~reference~~ case, and 526 degrees F with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +12 degrees F (~2.4 percent) increase are: (1) the timing of ECCS valve opening as determined by the IAB release and high CNV level setpoints (2) the assumption of a loss of normal AC power (3) fine CNV volume & heat structure nodalization and (4) single failure of one RRV to open. The peak CNV pressure is 894 psia for the base~~reference~~ case, and 959 psia with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +65 psi (~7.3 percent) increase are: (1) the timing of ECCS valve opening as determined by the ECCS actuation setpoint (2) the assumption of a loss of normal AC and DC power (3) single failure of one RRV to open (4) fine CNV volume & heat structure nodalization and (5) the RPV noncondensable release to CNV. The effect of the IAB release range of 950 psi +/- 50 psi was evaluated for Case 2 and determined to be equivalent to or non-limiting compared to the previously analyzed range of 1000-1200 psi. The detailed discussion of the Case 2 results that follow are for the limiting peak CNV pressure and temperature cases.

The sequence of events (Table 5-3) show that in the first seconds following the occurrence of a LOCA in the RCS injection line many automatic responses occur to transition the module from full power operation to an alignment that mitigates the initial LOCA blowdown phase. The break flow into the CNV causes a rapid pressurization that reaches the 9.5 psia high pressure setpoint. The following automatic actions occur on high CNV pressure:

- containment isolation
- reactor trip

As a conservative assumption, either a loss of normal AC power or a loss of normal AC and DC power is also assumed to occur at the time of the break and the ECCS signal is actuated on high CNV level or IAB release pressure. In the containment response analysis methodology the ECCS setpoints are important analysis input as they determine the time of the second primary system M&E release into the CNV via the ECCS valves. The peak CNV pressure and peak CNV wall temperature occur following the ECCS valve actuation, after the CNV has been preheated by the initial LOCA M&E release.

Following the alignment of the module for the LOCA blowdown phase, the primary system pressure and inventory decrease due to the loss of inventory through the LOCA. The CNV pressurizes and the steam condenses on the cold ID of the CNV. The condensate flows down the CNV walls and accumulates in a pool in the CNV lower head. The cold CNV wall absorbs the energy of the condensed steam and starts to heat up by conduction. Eventually the energy is transferred through the CNV wall to the reactor pool, and the pool temperature slowly increases. For the peak CNV wall temperature case, the ECCS signal actuates on high CNV level at 952 seconds, and the opening of the ECCS valves occurs at 955 seconds (after a 3-second signal delay). The ECCS actuation and opening of the three RVVs and one RRV causes the peak CNV wall temperature to occur at 978 seconds. For the peak pressure case, the ECCS signal actuates on IAB release pressure at 364 seconds, and the opening of the ECCS valves occurs at 367 seconds. The ECCS actuation and opening of the three RVVs and one RRV causes the peak CNV pressure to occur at 385 seconds. Then, as flow through the RVVs dimishes, the primary and CNV pressures converge, and continued heat transfer to the CNV leads to a gradual cooldown and depressurization phase. Pressure equalization enables recirculation flow from the CNV pool through the RRVs to establish the long-term cooling recirculation alignment.

The primary system response for the RCS injection line LOCA CNV peak pressure case is shown in Figures 5-3 through 5-9. Figure 5-3 shows the primary pressure response.

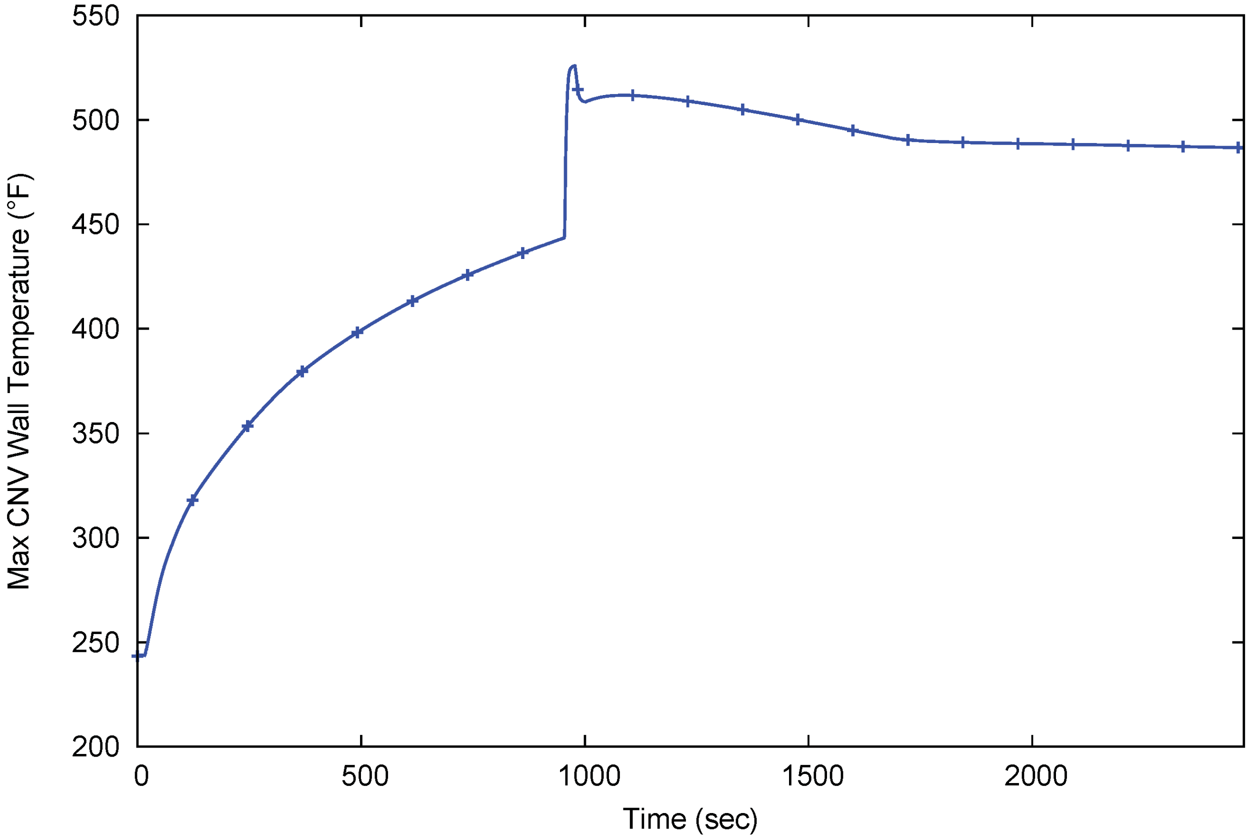

Figure 5-17 Case 2 containment vessel peak wall temperature - reactor coolant system injection line break loss-of-coolant accident (peak CNV wall temperature case)

5.1.3.3 Case 3: Reactor Pressure Vessel High Point Degasification Vent Line Loss-of-Coolant Accident

The LOCA in the RPV high point degasification line initiates an M&E release from the top of the pressurizer into the CNV. The sequence of events is shown in Table 5-4. The CNV pressure response and temperature response are shown in Figures 5-18 and 5-19. The CNV peak pressure is 554 psia for the base~~reference~~ case, and 901 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse~~1200 psid~~ IAB release pressure, high RCS flow, fine CNV volume nodalization). The peak CNV wall temperature is 471 degrees F for the base~~reference~~ case, and 489 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse~~1000 psid~~ IAB release pressure, high RCS flow, fine CNV volume nodalization). The results of this case are adequately representative of the inadvertent RVV opening event, although they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The effect of the IAB release pressure range of 950 psi +/- 50 psi, including effect of valves opening at different pressures within that range, has been evaluated. Case 3 is non-limiting and was confirmed to be non-limiting in comparison to the RRV opening event.

Table 5-4 Case 3 sequence of events - RPV high point degasification line break loss-of-coolant accident

Peak CNV Pressure Case Time (sec)	**Event**	**Peak CNV Temperature Case Time (sec)**
0	LOCA in RPV high point degasification line For peak pressure case only: • Loss of normal AC and DC power • FW/MS isolation • Reactor trip	0
1	High CNV pressure resulting in For peak pressure case: • Containment isolation For peak temperature case: • Reactor trip • FW/MS isolation • Loss of normal AC and DC power assumed at turbine trip	1
58	ECCS actuation on: IAB release pressure	106
61	ECCS valve opening	109
82	Peak CNV pressure reached: For peak pressure case: 901 psia For peak temperature case: 894 psia	128
454	Peak CNV temperature reached: For peak pressure case: 487 °F For peak temperature case: 489 °F	478
~1900	CNV pressure decreases to <50% of peak pressure	~2000

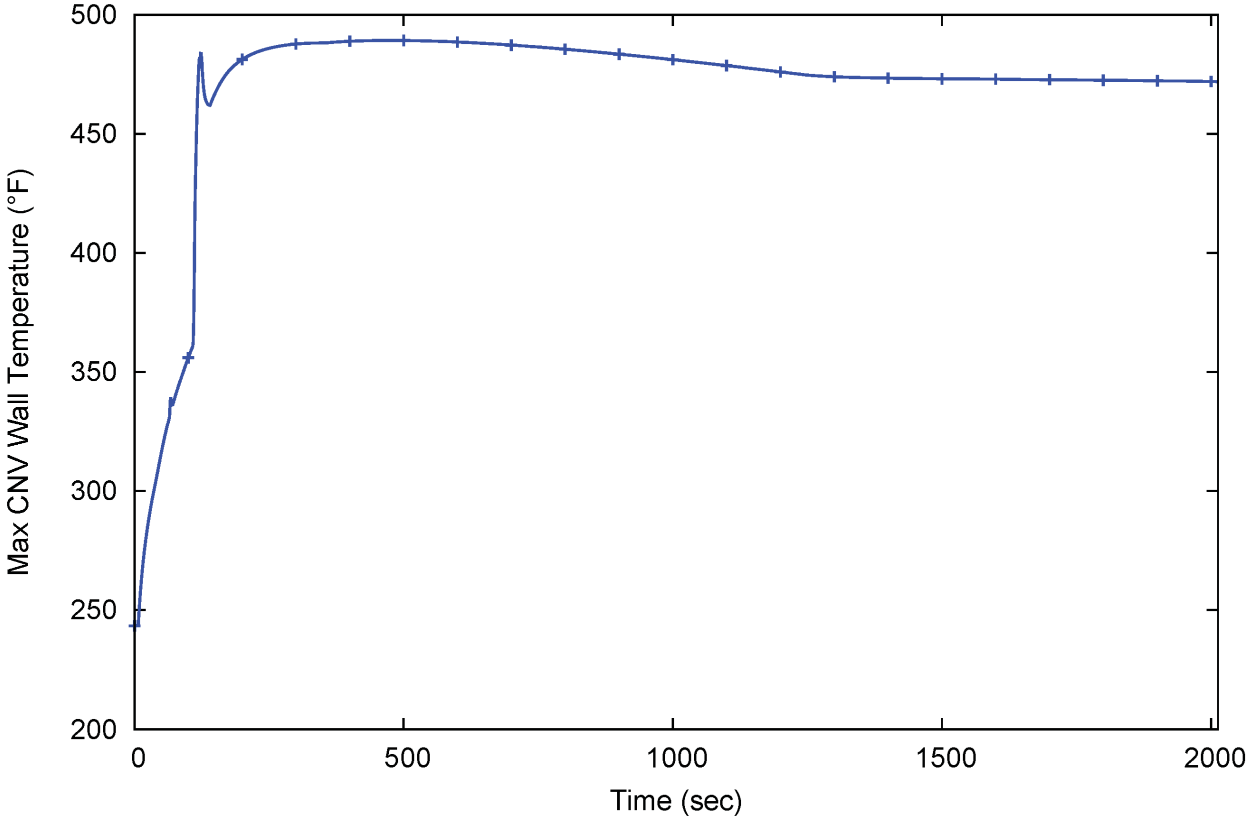

Figure 5-19 Case 3 containment vessel wall temperature – high point vent line break loss-of-coolant accident

5.1.3.4 Case 4: Inadvertent Reactor Vent Valve Opening Anticipated Operational Occurrence

The inadvertent RVV actuation anticipated operational occurrence (AOO) initiates an M&E release from the top of the pressurizer into the CNV. The sequence of events is shown in Table 5-5. The CNV pressure response and temperature response are shown in Figures 5-20 and 5-21. The CNV peak pressure is 856 psia for the base~~reference~~ case, and 911 psia with the combined effect of the adverse sensitivity parameters (loss of normal AC and DC power, adverse~~1200 psid~~ IAB release pressure, low RCS flow, fine CNV heat structure & reactor pool nodalization). The peak CNV temperature is 483 degrees F for the base~~reference~~ case, and 486 degrees F for the case with the combined effect of the adverse sensitivity parameters (normal AC and DC power available, fine CNV volume nodalization). The results of this case are adequately representative of the high point line break, even though they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The effect of the IAB release pressure range of 950 psi +/- 50 psi, including effect of valves opening at different pressures within that range, has been evaluated. Case 4 is non-limiting and was confirmed to be non-limiting in comparison to the RRV opening event.

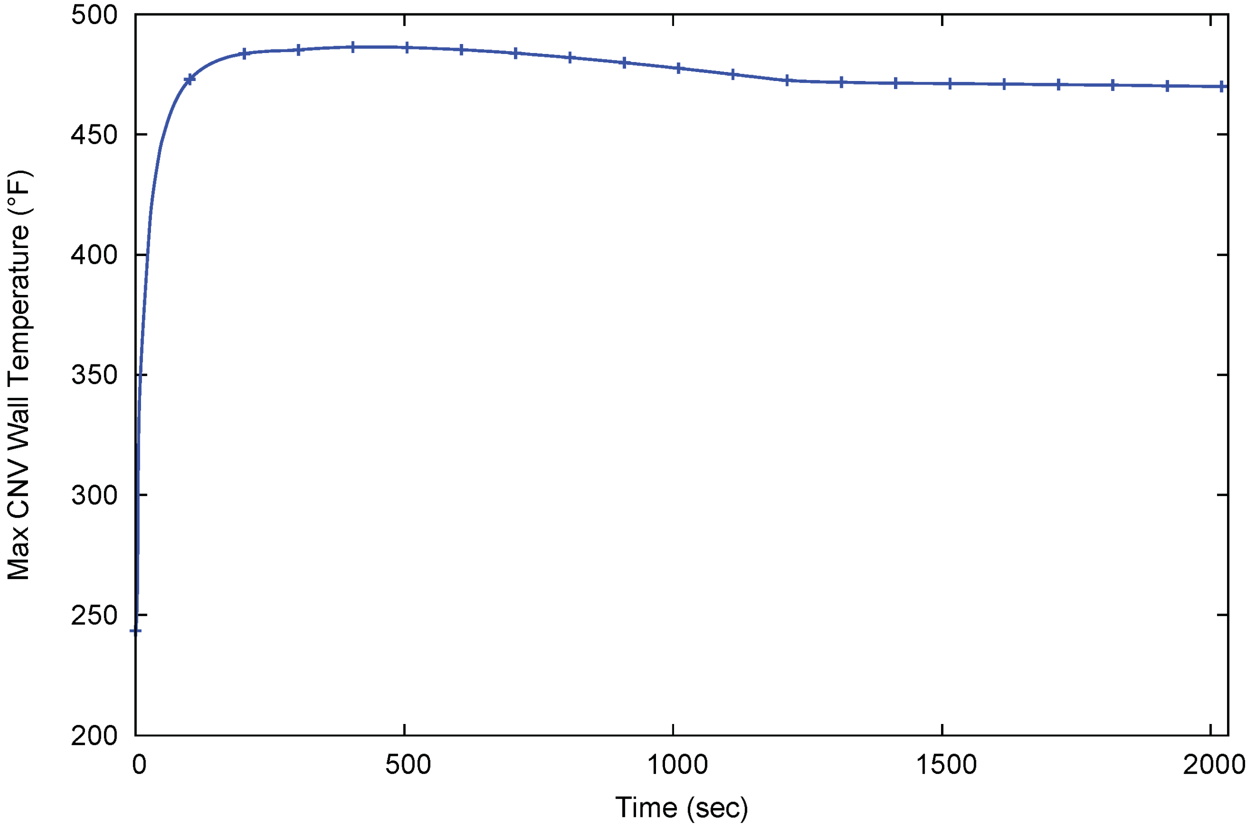

Figure 5-21 Case 4 containment vessel wall temperature – inadvertent reactor vent valve opening event

5.1.3.5 Case 5: Limiting Overall Containment Vessel Pressure Event - Inadvertent Reactor Recirculation Valve Opening Anticipated Operational Occurrence

The inadvertent RRV actuation initiates an M&E release from the downcomer into the CNV. The results of the primary release event M&E release break spectrum analysis and sensitivity analyses have determined that this AOO (Case 5) results in the limiting peak CNV pressure for all postulated events. The limiting case, which accounts for the IAB release pressure of 950 psid +/- 50 psi and the potential for ECCS valves to open at different differential pressures over this range, is summarized in FSAR Section 6.2.

The following discussion reflects the RRV opening event analysis when the ECCS valves are assumed to open at 1000 psid, resulting in a peak pressure of 986 psia. This case is representative of the RRV opening event. The sequence of events for this representative case is shown in Table 5-6, and detailed results for key parameters are shown in Figures 5-22 through 5-35. The CNV peak pressure is 941 psia for the ~~reference~~base case, and 986 psia with the combined effect of the adverse sensitivity parameters. The sensitivity parameters that contribute to the +45 psi (~4.8 percent) increase are: (1) the timing of the ECCS valve opening as determined by the IAB release and high CNV level setpoint; (2) the assumption of a loss of normal AC and DC power; (3) single failure of an RRV; (4) fine

CNV volume & heat structure and reactor pool nodalization; (5) fast RPV non-condensable release to CNV; and (6) low RCS flow. The peak CNV temperature is 492 degrees F for the base~~reference~~ case, and 512 degrees F with the combined effect of the adverse sensitivity parameters (loss of normal AC power, low-biased high CNV level setpoint, single failure of an RRV, fine CNV volume & reactor pool nodalization).

The sequence of events (Table 5-6) shows that in the first seconds following the occurrence of an inadvertent RRV event many automatic responses occur to transition the module from full-power operation to an alignment that mitigates the initial blowdown phase. The break flow into the CNV causes a rapid pressurization that reaches the 9.5 psia high pressure setpoint. The following automatic actions occur on high CNV pressure:

- containment isolation resulting in MSIV and FWIV closure
- reactor trip

For the peak temperature case, a loss of normal AC power~~, along with DC power~~ is assumed to occur at the time of the break. RRVs and RVVs opening does not occur until the high CNV level setpoint is reached. In the containment response analysis methodology the high CNV level setpoint is an important analysis input as it determines the second primary system M&E release into the CNV through the RVVs and the second RRV. The peak CNV wall temperature occurs following the RVVs opening after the CNV has been preheated by the initial M&E release.

For the peak pressure case, a loss of normal AC and DC power is also assumed to occur at the time of the break. This results in an ECCS signal. However, RRVs and RVVs opening does not occur until the differential pressure across the valve decreases to below the IAB release pressure. In the containment response analysis methodology the IAB release pressure is an important analysis input as it determines the second primary system M&E release into the CNV through the RVVs and the second RRV. The peak CNV pressure occur following the RVVs opening after the CNV has been preheated by the initial M&E release.

Following the alignment of the module for blowdown, the primary system pressure and inventory decrease due to the loss of inventory. The CNV pressurizes and the steam condenses on the cold interior wall of the CNV. The condensate flows down the CNV walls and accumulates along with unflashed break liquid in a pool in the CNV lower head. The cold CNV wall absorbs the energy of the condensed steam and starts to heat up by conduction. Eventually the energy is transferred through the CNV wall to the reactor pool, and the pool temperature slowly increases. Opening of the ECCS valves occurs at 171 seconds for the peak temperature case (when the high CNV level setpoint is reached) and at 77 seconds for the peak pressure case (when the RCS pressure decreases to ~~below the 1000 psid~~an adverse IAB release pressure), as determined by the results of sensitivity analyses. For the peak temperature case, opening of the three RVVs and the second RRV results in the peak CNV pressure and wall temperature at 182 and 180 seconds, respectively. For the peak pressure case, opening of the three RVVs and the second RRV results in the peak CNV pressure and wall temperature at 91 and 596 seconds, respectively. As flow through the RVVs diminishes, the primary and CNV pressures converge, and continued heat transfer to the CNV leads to a gradual cooldown and depressurization phase. Pressure equalization enables recirculation flow from the CNV pool through the RRVs to establish the long-term cooling recirculation alignment.

The primary system response for the representative Case 5 inadvertent RRV opening event (peak pressure case) is shown in Figures 5-22 through 5-28. Figure 5-22 shows the primary pressure response. The initial depressurization phase due to the RRV opening is continued by the rapid depressurization when the RVVs open. Figures 5-23 and 5-24 show the inventory in the pressurizer and in the riser. These figures show the expected trend of a decreasing level in the primary followed by a stabilization in inventory, with some liquid holdup in the pressurizer. A sensitivity study that decreased the interphase drag in the upper riser, riser upper plenum, pressurizer baffle, pressurizer, and the downcomer with the intent of reducing liquid entrainment, showed that there was no adverse impact on the peak CNV pressure for this case. Figure 5-25 shows the primary coolant temperatures at six locations. Following ECCS actuation the temperatures converge and the cooldown proceeds. Figure 5-26 shows the RRV opening and ECCS mass flowrates. It is evident that the ECCS flow immediately following ECCS actuation, mainly the flow through the three RVVs into the CNV, is significant. It is this flow spike that causes the peak CNV pressure and wall temperatures to occur shortly thereafter as shown in Table 5-6. Figures 5-27 and 5-28 show the integrated LOCA and ECCS mass flowrate and energy flowrate.

The CNV and reactor pool response for the representative Case 5 inadvertent RRV opening event~~actuation LOCA~~ is shown in Figures 5-29 to 5-34. Figure 5-29 shows the CNV pressure response and how pressure rapidly increases to the limiting peak value of 986 psia. This limiting NRELAP5 result can be compared to the CNV design pressure of 1050 psia. ~~This is a key result of this limiting containment response analysis case.~~ Figure 5-29 also demonstrates the long term cooling capability of the UHS. CNV pressure is reduced to below 50 percent of the peak value within two hours of accident initiation.

Figure 5-30 shows the CNV liquid level increase as the unflashed break flow and condensed steam accumulates. Figure 5-31 shows the CNV vapor temperature. Initially, flashing of the break flow at low CNV pressure results in a temperature decrease. {{

}}$^{2(a),(c)}$ Figure 5-32 shows the peak CNV wall temperature and the limiting value of 492 degrees F. Figure 5-33 shows the temperature profile across the CNV wall at the 45 foot elevation. There is a large temperature gradient across the CNV wall. Figure 5-34 shows the reactor pool temperatures for a range of elevations. The reactor pool temperature does not increase significantly for the short duration of these M&E release analyses. From Figures 5-31 through 5-34 it is evident that the CNV wall is the significant heat sink in the short-term. Even with the conservative initial reactor pool level of 65 ft above the pool floor and a temperature of 110 degrees F assumed in these analyses, the CNV wall is capable of maintaining the peak CNV pressure within the design limit.

Figure 5-35 shows the energy balance during the RRV opening event~~loss-of-coolant accident~~ and the trends of the heat sources and sinks. At approximately 750 seconds the energy release from the LOCA and the RVV valves decreases to below the energy transferred through the CNV wall. The CNV wall then continues to provide a strong heat sink for the sustained cooldown and depressurization of the module.

Table 5-6 Case 5 sequence of events – inadvertent reactor recirculation valve opening event

Peak CNV Pressure Case Time (sec)	**Event**	**Peak CNV Temperature Case Time (sec)**
0	Inadvertent RRV actuation: For peak temperature case • Loss of normal AC power • FW/MS isolation For peak pressure case • Loss of normal AC and DC power • FW/MS isolation • Reactor trip	0
0.4	High CNV pressure resulting in: For peak temperature case • Containment isolation • Reactor trip For peak pressure case • Containment isolation	0.4
74	ECCS actuation on : For peak temperature case • high CNV level For peak pressure case • IAB release pressure	168
77	ECCS valve opening	171
91	Peak CNV pressure reached: For peak pressure case: 986 psia For peak temperature case: 967 psia	182
596	Peak CNV temperature reached: For peak pressure case: 492 °F For peak temperature case: 512 °F	180
~1800	CNV pressure decreases to <50% of peak pressure	~1800

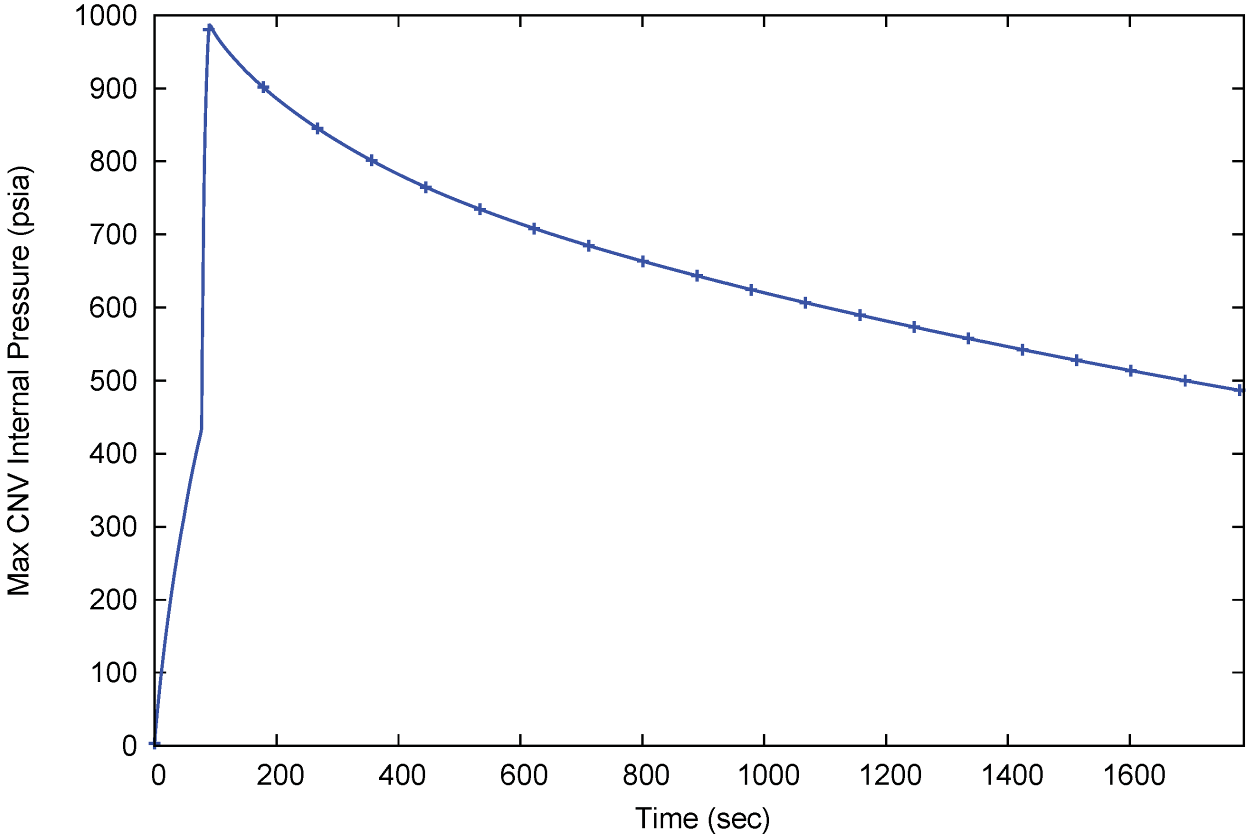

Figure 5-29 Case 5 containment vessel pressure - inadvertent reactor recirculation valve opening event (representative peak~~overall limiting~~ pressure~~ case~~)

differential decreases to below the IAB release pressure. In the containment response analysis methodology the IAB release pressure is an important analysis input as it determines the second M&E release into the CNV via the RVVs. A higher IAB release pressure results in an earlier opening of the ECCS valves when the RCS is hotter. The results of this case are adequately representative of the feedwater line break case although they do not reflect the IAB release pressure range of 950 psi +/- 50 psi. The IAB release pressure range of 950 psi +/- 50 psi was evaluated and determined to be non-limiting. The peak CNV pressure and peak CNV wall temperature occur following the RVV actuation, after the CNV has been preheated by the initial M&E release. Sensitivity studies of single failures have determined that a failure of a MSIV to close had an adverse impact on the CNV peak pressure and temperature results.

Following the alignment of the module to mitigate the initial secondary blowdown phase, the secondary system pressure and inventory decrease due to the loss of inventory. The CNV pressurizes and the steam condenses on the cold ID of the CNV. The condensate flows down the CNV walls and accumulates with unflashed secondary break liquid in a pool in the CNV lower head. The cold CNV wall absorbs the energy of the condensed steam and starts to heat up by conduction. Eventually the energy is transferred through the CNV wall to the reactor pool, and the pool temperature slowly increases. After the end of the secondary blowdown phase decay heat removal is via the DHRS. Opening of the ECCS valves occurs at 11,566 seconds when the pressure differential decreases to below the 1200 psid IAB release pressure. This causes the CNV peak pressure (416 psia) and the peak CNV wall temperature (407 degrees F) at ~11,600 and ~11,870 seconds, respectively. As flow through the RVVs dimishes, the primary system and CNV pressures converge, and continued heat transfer to the CNV leads to a gradual cooldown and depressurization phase. Pressure equalization enables recirculation flow from the CNV pool through the RRVs to establish the long-term cooling recirculation alignment.

The module response for the FWLB analysis is shown in Figure 5-50 through Figure 5-64. Figure 5-50 shows the SG pressure response with the affected SG (SG2) depressurizing via blowdown out the break into the CNV and stabilizing at a low pressure . The unaffected SG (SG1) pressure fluctuates in response to DHRS heat transfer. The affected SG repressurizes by reverse break flow on ECCS valve opening. Then, both SGs depressurize as ECCS heat transfer dominates. Figure 5-51 shows the gradual primary system cooldown due to DHRS, and the increase in the cooldown rate with the opening of the ECCS valves. Figure 5-52 shows the relatively steady pressurizer level decrease during DHRS cooling and then a rapid level decrease when ECCS valves open. Figure 5-53 shows the riser level remaining full until the ECCS valves open, and then level rapidly decreases before stabilizing. Primary system pressure (Figure 5-54) gradually decreases during the DHRS cooldown period due to loss of pressurizer heaters and then rapidly depressurizes on ECCS valves opening. Figure 5-55 through Figure 5-57 show the break and ECCS mass release, the integrated mass release, and the integrated energy release into the CNV, respectively. The FWLB flow rate and integrated mass release is not significant due to the small SG inventory. Due to the insignificance of the secondary break flow, the effect of liquid entrainment is also insignificant. The primary system M&E release through the three RVVs is the significant M&E release event for the FWLB accident.

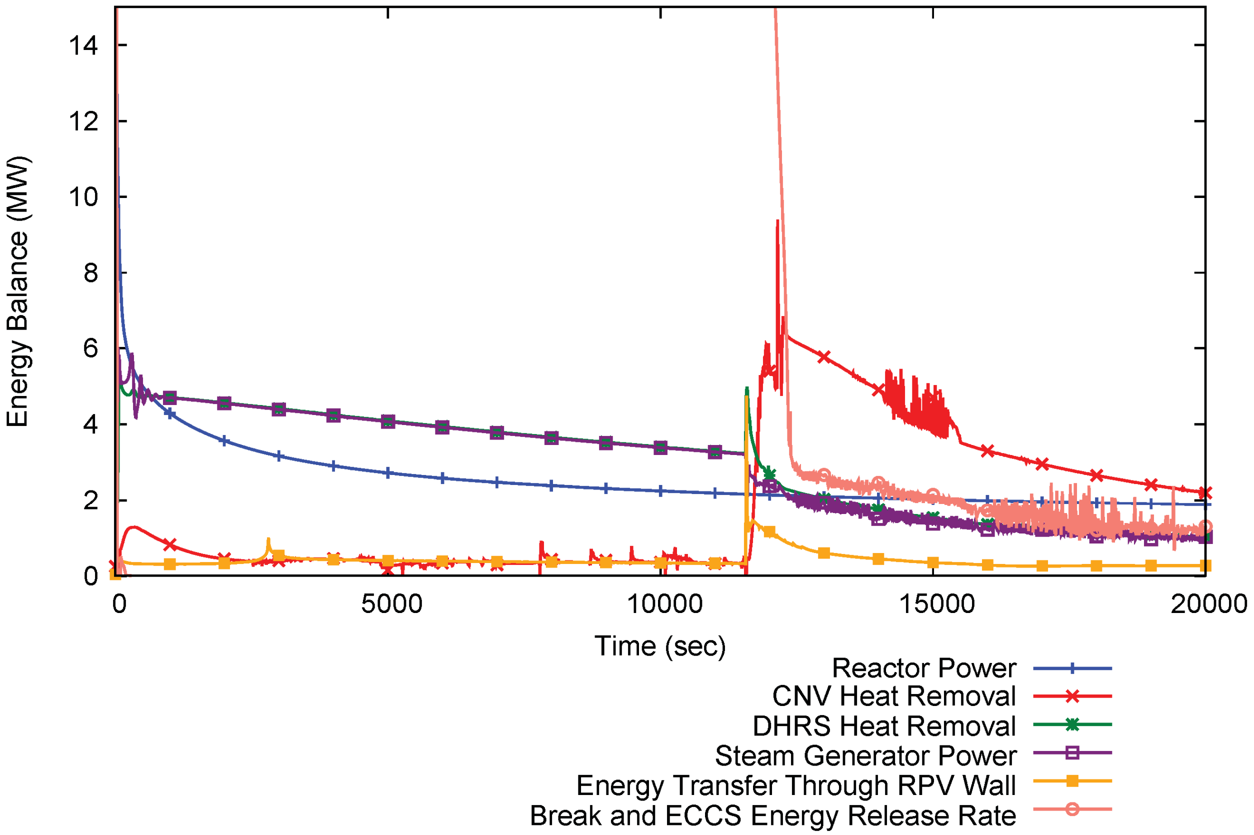

Figure 5-64 Feedwater line break energy balance

5.4 Margin Assessment

The following subsections discuss the analytical and design margin incorporated into the NPM design. Section 5.4.1 describes margin inherent in the enhanced requirements imposed on the CNV as an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Class 1 vessel. Section 5.4.2 describes conservative modeling assumptions in the containment peak pressure and temperature analysis.

5.4.1 Hydrostatic~~Atmospheric~~ Pressure

~~NuScale analyses do not couple the internal and external containment pressures. Internal pressures are conservatively evaluated with an assumed external pressure of 0 psia. The atmospheric pressure, acting against the exterior CNV surface, is neglected.~~

The overall limiting peak CNV peak pressure results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power. The overall limiting CNV peak pressure~~,~~ is ~~986~~994 psia, which is approximately ~~6~~5 percent below the design pressure of 1050 psia, which occurs at a CNV elevation at the bottom of the CNV. ~~Atmospheric~~The peak pressure occuring in the vapor space of the CNV is 987 psia; the difference is due to the hydrostatic pressure of liquid accumulation within the CNV at the time of peak pressure. The ~~pressure and~~ reactor pool hydrostatic head, which acts~~ing~~ against the CNV exterior surface, provides ~~approximately 22 psi~~ additional margin~~,~~ that is not credited by the CNV response analysis methodology.

~~This demonstrates additional margin in the CNV design that is not considered by the containment response analysis.~~

5.4.2 Decay Heat Removal System Availability

The LOCA (Case 2) and AOO (Case 5) are performed with and without DHRS available to estimate the impact of DHRS availability on the CNV peak pressure response. The DHRS is conservatively not credited in the design basis containment response analysis cases. The NRELAP5 code has not been validated to cover DHRS performance during LOCAs or valve opening events. However, the DHRS is a single-failure proof safety-related system that can be credited in the future, with additional NRELAP5 validation, if the CNV pressure margin is reduced for any reason (design changes). The results of the DHRS available cases indicate that about 37 psi additional margin could be gained by credit for DHRS availability.

5.4.3 Conclusion

The NPM design provides sufficient margin to satisfy the requirements of GDC 16 and 50. The LOCA peak pressure ~~provides approximately 9%~~ and the AOO peak pressure analyses demonstrate that ~~provides approximately 6% margin~~sufficient margin to the CNV design pressure of 1050 psia~~,~~ is available to address the acceptance criteria given by DSRS Section 6.2.1.1.A (See Table 2-2). The CNV response to the limiting LOCA event and AOO transient are conservatively calculated and demonstrate that the peak calculated pressures are below the CNV design pressure and decrease in pressure to one-half of the peak value within 24 hours.

Further assurance of sufficient margin is provided through consideration of hydrostatic head and availability of the DHRS in the containment response analysis.~~Further assurance of sufficient margin is provided through consideration of atmospheric pressure and hydrostatic head, acting against the CNV exterior surface and availability of the DHRS system in the containment response analysis. Consideration of external pressure acting against the CNV exterior surface reduces the differential pressure across the CNV wall by about 22 psi. The determination of NPM design pressure, in accordance with ASME Class 1 criteria, is conservative relative to Class MC and CC containments. This design pressure does not consider the additional margin provided by the internal and external cladding of the upper CNV shell. The effect of DHRS actuation in reducing peak containment pressure was not credited in the containment response analysis. Consideration of the effect of DHRS actuation, along with external pressure acting against the CNV exterior surface, reduces the differential across the CNV wall by approximately 59 psi.~~

Sensitivity studies determined an approximate 8 psi increase in the CNV peak pressure, documented in FSAR Section 6.2, for the limiting inadvertent opening of a reactor recirculation valve event if a lower IAB release pressure (listed in Section 5.1.1) is considered. The sensitivity calculations considered effects of different ECCS valves opening at different differential pressures over this range. The limiting pressure result, accounting for different ECCS valves opening at different pressures, of 994 psia is presented in FSAR Section 6.2. Considering the maximum pressure, and the margin not credited by the analysis discussed above, sufficient margin is provided to satisfy the requirements of GDC 16 and 50.

The containment response analysis methodology, analysis results and further conservatisms related to design and system operation provide assurance that the NPM design demonstrates sufficient margin to satisfy the requirements of GDC 16 and 50.

- For the limiting cases the results of the sensitivity studies, including postulated single failures, showed only a limited impact (<1 percent) on the key figures-of-merit. The loss of normal AC and DC power and the timing of ECCS valve opening were the most important sensitivity parameters.

The limiting LOCA peak pressure and CNV wall temperature are a result of the reactor coolant system (RCS) injection line break. The LOCA limiting peak CNV wall temperature is approximately 526 degrees F and it results from a reactor coolant system injection line break case, with a loss of normal alternating current (AC) power. The LOCA limiting peak pressure is approximately 959 psia, which results from a reactor coolant system injection line break case, with a loss of normal AC and DC power. The LOCA event peak CNV pressure is below the CNV design pressure of 1050 psia. The LOCA peak CNV pressure and wall temperature bound the main steamline break (MSLB) and feedwater lind break (FWLB) results.

The overall limiting peak CNV accident pressure is approximately 994~~986~~ psia, which is approximately 5~~6~~ percent below the containment design pressure of 1050 psia. It results from an inadvertent reactor recirculation valve opening anticipated operational occurrence with a loss of normal AC and DC power. The peak pressure of the limiting anticipated operational occurrence is also less than the CNV design pressure of 1050 psia. The CNV pressure for this limiting case is reduced to below 50 percent of the peak value in less than 2 hours, demonstrating adequate NPM containment heat removal.

Section 5.4 discussed margin in the NPM design that is not included in the CNV design pressure rating or modeled in the containment response analyses. Design factors conservatively not credited include static water pressure and~~atmospheric pressure acting against the CNV exterior surface and~~ the availability of the DHRS.

The containment response analysis demonstrates that the NPM design has adequate margin to design limits and that it satisfies the requriements of GDC 16 and 50 and PDC 38.

8.0 Appendicies

8.1 Mass and Energy Input

The purpose of this Appendix is to present the mass and energy release to the CNV during the limiting LOCA event (Case 2 maximum temperature case), the ~~limiting~~ overall peak CNV pressure event (Case 5 representative pressure case) and the limiting secondary system release event (MSLB), up to the time that the peak pressure is reduced to one half its value. The mass and energy releases provided in Table 8-1 are representative of the RRV opening event and reflect the case where all ECCS valves open at 1000 psid, resulting in peak containment pressure of 986 psia. The limiting peak pressure and temperature results are presented in FSAR Section 6.2.

~~8.2 Heat Sink Tables~~

~~The purpose of this Appendix is to present the passive heat sink characteristics credited in the containment response analysis methodology.~~

Table 8-1 Case 5 ~~Limiting~~ Representative Peak Pressure Case – Mass and Energy Release

Time (s) (1)	Mass Release (lbm/s)	Energy Release (Btu/s)
0.00	0.00	0.00
1.00	520.85	257627.45
2.00	523.17	258750.86
3.00	521.93	258164.72
4.00	519.26	256962.72
5.00	515.99	255607.71
6.00	512.44	254280.63
7.00	508.73	253062.33
8.00	504.94	251984.33
9.00	500.89	250943.23
10.00	497.07	250168.15
11.00	492.84	249319.52
12.00	488.50	248522.89
13.00	483.98	247730.73
14.00	479.49	247027.59
15.00	475.20	246477.05
16.00	470.79	245889.66
17.00	466.32	245272.77
18.00	461.90	244650.14
19.00	457.90	244207.37
20.00	454.78	244180.29
40.00	412.85	228110.54
60.00	378.12	216874.65
61.00	377.02	215906.03

8.2 Heat Sink Tables

The purpose of this Appendix is to present the passive heat sink characteristics credited in the containment response analysis methodology.

8.2.1 Listing of Passive Heat Sinks

The containment vessel shell is the only passive heat sink credited in the containment response analysis methodology.

8.2.2 Modeling of Passive Heat Sinks

Table 8-4 Passive heat sinks

Passive Heat Sink (Vessel steel plate)	**Material**	**Thickness, in**	**Group**	**Exposed Surface Area by Thickness Group, ft^2**	**Shell Volume, ft3**	**Total Mass, lbm**	**Total Surface Area, ft^2**
{{							
							}}$^{2(a),(c)}$

Enclosure 3:

Affidavit of Zackary W. Rad, AF-0919-67237

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

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

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

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

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

4. The information sought to be withheld is in the enclosed response to NRC Request for Additional Information No. 466, eRAI 9482. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
   a. The information sought to be withheld is owned and has been held in confidence by NuScale.
   b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
   c. The information is being transmitted to and received by the NRC in confidence.
   d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
   e. Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 31, 2019.

Zackary W. Rad