Response to SDAA Audit Question

Question Number: A-15.1.5-1

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Question:

EC-116465, which supports the “Steam System Piping Failures Inside and Outside Containment” analysis in FSAR Section 15.1.5, analyzes scenarios in which steam line breaks inside containment concurrent with loss of EDNS lead to release of primary coolant. This scenario differs from that considered in FSAR 15.0.3 and described in Appendix E to RG 1.183 in that primary coolant leaves the RPV through opening of the RVV and leaves containment through the ruptured steam line. While mass releases appear to be bounded by those assumed in the radiological consequence analysis documented in FSAR 15.0.3.7.3, “Main Steam Line Break Outside Containment Accident,” the FSAR does not document the analysis performed to ensure the FSAR 15.0.3 analysis is, in fact, bounding. Provide FSAR markup describing the analysis performed.

Response:

EC-116465, “NPM-20 Steam System Piping Failure Analysis,” Revision 1 has been previously provided in the electronic reading room (eRR). The audit question states that EC-116465 “analyzes scenarios in which steam line breaks inside containment concurrent with loss of **EDNS** lead to release of primary coolant,” with emphasis added. Note that the normal direct current power system (EDNS) plays no role in the primary coolant release scenario being described. The augmented direct current power system (EDAS) is the system that, if lost, allows a very brief (i.e., 10-second) release of primary coolant during a steam line break (SLB) inside containment because the main steam isolation valves (MSIVs) immediately close without EDAS power.

The scenario described in EC-116465 and in this audit question is addressed in the Final Safety Analysis Report (FSAR). FSAR Section 15.1.5.2 states:

"Section 15.0.0 describes the impact of a loss of the augmented direct current power system (EDAS), which includes the opening of the reactor vent valves. The limiting SLB cases assume EDAS is available, consistent with the discussion in Section 15.0.0. However, an SLB inside containment with a loss of EDAS creates a temporary pathway for primary coolant to be released through the open reactor vent valves into containment and then out of containment through the broken steam line. This potential release pathway is isolated by the MSIV closure which also occurs on a loss of EDAS. The amount of primary coolant release is therefore inherently limited by the MSIV closure time and is bounded by the assumed primary coolant releases analyzed in Section 15.0.3. A single failure of the MSIV in this scenario is inconsequential because the secondary MSIV has the same closure time."

Note that a specific dose analysis is not performed for this SLB inside containment with loss of EDAS scenario. Instead, the FSAR Section 15.1.5 event-specific analysis in EC-116465 confirms that the primary coolant release in this scenario {{ }}2(a),(c) is less than the analyses in FSAR Section 15.0.3 that assume a bounding primary coolant release of 23,000 lbm. This confirmation is similar to that performed for the failure of small lines carrying primary coolant outside containment event in FSAR Section 15.6.2 and the steam generator tube failure event in FSAR Section 15.6.3. In the response to audit questions A-15.0.3.7-1 and A-15.0.3.7-2 on those events, NuScale described the comparisons made and updated the FSAR to explicitly state that the comparisons were performed. The NRC subsequently closed audit questions A-15.0.3.7-1 and A-15.0.3.7-2. Therefore, NuScale is taking the same approach for this audit question: (1) the comparisons are described below and (2) the FSAR is updated to describe the comparisons performed.

Table 1 shows the comparison of the SLB inside containment with loss of EDAS results from EC-116465 to the failure of small lines carrying primary coolant outside containment. {{

}}2(a),(c)

Table 2 shows the comparison of the SLB inside containment with loss of EDAS results from EC-116465 to the steam generator tube failure. {{

}}2(a),(c)

Tables 1 and 2 show that the SLB inside containment with loss of EDAS event-specific results are bounded by {{

}}2(a),(c) the assumptions in the dose assessments of FSAR Sections 15.0.3.7.1 and 15.0.3.7.2, respectively.

FSAR Sections 15.0.3.7.3 and 15.1.5 are updated as shown in the attached markups.

Table 1: Comparison of Steam Line Break Inside Containment with Loss of Augmented Direct Current Power System to Failure of Small Lines Carrying Primary Coolant Outside Containment Dose Assessment

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}}2(a),(c)

Table 2: Comparison of Steam Line Break Inside Containment with Loss of Augmented Direct Current Power System to Steam Generator Tube Failure Dose Assessment

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}}2(a),(c)

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

Doses are determined at the EAB, the LPZ, and for personnel in the MCR and the TSC. The MCR model is described in Section 15.0.3.6.1. The potential radiological consequences of the SGTF event are presented in Table 15.0-10.

15.0.3.7.3 Main Steam Line Break Outside Containment Accident

Radiological consequences of the MSLB outside containment accident are calculated based on the guidance provided in Appendix E of RG 1.183 as described in Section 3.2.3 of Reference 15.0-6. Section 15.1.5 describes the sequence of events and thermal-hydraulic response to an MSLB outside containment. The analysis in Section 15.1.5 shows the reactor core remains covered and no fuel failures occur.

A bounding release of primary coolant is defined for the dose consequence analysis rather than using results from the transient analysis described in Section 15.1.5. The primary coolant associated with the maximum primary-to-secondary leak rate allowed by design-basis limits is assumed to be released directly to the environment. The secondary system is not modeled and no credit is taken for holdup or dilution in the secondary system. The direct release of primary coolant to the environment is assumed to be terminated after 30 hours when the RCS is depressurized and primary and secondary system pressures equalize.

Iodine spike assumptions for this event are listed in Reference 15.0-6. The primary coolant contains an assumed concentration of 5.8E-02 μCi/gm DE I-131 for the coincident iodine spike scenario and 3.5 μCi/gm DE I-131 for the pre-incident iodine spike scenario. For both iodine spiking scenarios, the primary coolant is assumed to contain 16 μCi/gm DE Xe-133.

There are no single failures for this event that could result in more severe radiological consequences.

Doses are determined at the EAB, the LPZ, and for personnel in the MCR and the TSC. The MCR model is described in Section 15.0.3.6.1. The potential radiological consequences of a steam system piping failure outside the primary containment are presented in Table 15.0-10.

Audit Question A-15.1.5-1

An MSLB inside containment with a loss of EDAS introduces an alternate pathway for primary coolant to be released. The primary coolant release is inherently limited by the secondary system isolation that occurs with the loss of EDAS. This alternate release pathway is not addressed in RG 1.183 or Reference 15.0-6, but is similar to a steam generator tube failure or failure of small lines carrying primary coolant outside containment. The primary coolant releases assumed in these other events are bounding of the results of the MSLB inside containment with a loss of EDAS transient analysis described in Section 15.1.5. Therefore, the radiological consequences of these other events presented in Table 15.0-10 are bounding of the radiological consequences of an MSLB inside containment with a loss of EDAS.

complete blowdown of the affected SG train. If the break location is downstream of the secondary MSIV, the single failure of either the MSIV or the secondary MSIV is inconsequential because the valves have the same closure time.

Normal AC power is assumed to be available for the limiting SLB cases. A loss of AC power, either at event initiation or at reactor trip, is not a conservative condition because FW is lost, which reduces the severity of the overcooling event.

Because the low AC voltage signal is credited with DHRS actuation in some of the non-limiting cases, a potential loss of the normal direct current power system (EDNS) is assessed based on the discussion in Section 15.0.0. However, the loss of EDNS is determined to have no impact on the results of those cases. The limiting SLB cases assume EDNS is available.

Audit Question A-15.1.5-1

Section 15.0.0 describes the impact of a loss of the augmented direct current power system (EDAS), which includes the opening of the reactor vent valves. The limiting SLB cases assume EDAS is available, consistent with the discussion in Section 15.0.0. However, an SLB inside containment with a loss of EDAS creates a temporary pathway for primary coolant to be released through the open reactor vent valves into containment and then out of containment through the broken steam line. This potential release pathway is isolated by the SSI~~MSIV closure~~ which also occurs on a loss of EDAS. The time of isolation is equal to the MSIV closure time. The amount of primary coolant release is therefore inherently limited by the MSIV closure time~~ and is bounded by the assumed primary coolant releases analyzed in Section 15.0.3~~. A single failure of the MSIV in this scenario is inconsequential because the secondary MSIV has the same closure time. Analyses of an SLB inside containment with a loss of EDAS are performed to determine the primary coolant release during the MSIV closure time to compare to the assumed primary coolant releases analyzed in Section 15.0.3.

15.1.5.3 Thermal Hydraulic and Subchannel Analyses

15.1.5.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to an SLB is performed using NRELAP5. A description of the NRELAP5 model is provided in Section 15.0.2. The NRELAP5 model is based on the design features of an NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The steam piping breaks are modeled in NRELAP5 as valves that instantly open at transient initiation and have a sudden infinite expansion loss. This modeling is appropriate for an SLB because the break vents either to a relatively large CNV or to an even larger Reactor Building or Turbine Building. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic similar power response is driven by moderator feedback instead. Reactor power reaches the high power limit, tripping the reactor.

In this SLB case, the overcooling effect initially lowers RCS pressure. However, as core power rises, pressure increases until the reactor trip. The pressure in the intact SG increases following SSI and DHRS actuation for the same reasons as the main steam pressure increase shown in the limiting RPV pressure SLB case. The peak primary and secondary pressures are bounded by the pressures presented for the maximum pressure cases.

The CHFR decreases as reactor power and RCS pressure increase. The MPS terminates this transient before the CHFR reaches the CHF analysis limit. The MCHFR for the limiting steam pipe break case does not violate the CHF analysis limit as shown in Figure 15.1-43.

As in the SLB case with limiting RPV pressure, the limiting single failure assumed in this SLB case is failure of the MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation. This single failure disables the DHRS functionality on the impacted SG and causes a higher heat load on the intact DHRS train. However, this failure does not affect the limiting CHFR because it occurs after the time of MCHFR.

The result of this SLB case is a stable plant condition in which the DHRS maintains core cooling.

15.1.5.4 Radiological Consequences

Audit Question A-15.1.5-1

The radiological consequences of the SLB outside containment event are discussed in Section 15.0.3. The radiological consequences of the SLB inside containment with loss of EDAS are discussed in Section 15.0.3 and are bounded by analyses with assumptions for primary coolant release (23,000 lbm) and SSI time (30 minutes) that are bounding of the limiting results from the spectrum of SLB inside containment conditions evaluated in this section.

15.1.5.5 Conclusions

The DSRS acceptance criteria for this accident are met for the limiting cases.

1) Pressure in the RCS and in the MSS should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

    - The limiting RPV pressure for an SLB is under the more conservative AOO acceptance criterion of 110 percent of design values. The limiting MSS pressures for an SLB must be less than or equal to 110 percent of the design value. The calculated RPV and MSS pressures demonstrate margin to the acceptance criterion. Therefore, the acceptance criterion for pressure is met for this event. The maximum primary and SG pressure values for the cases analyzed are shown in Table 15.1-15.

2) The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit.

   - The MCHFR for this event is above the CHF analysis limit as shown in Table 15.1-15. Therefore, this acceptance criterion is met.

Audit Question A-15.1.5-1

3) The radiological evaluations are~~is~~ in Section 15.0.3 and demonstrate~~s~~ the acceptance criteria are met.

4) System(s) provided for decay heat removal must be highly reliable and, when required, automatically initiated. For the NuScale Power Plant US460 standard design, the DHRS provides the safety-related means of decay heat removal.

   - The results of the analysis show the DHRS actuates and provides heat removal during an SLB, ensuring acceptance criteria are not challenged.

15.1.6 Loss of Containment Vacuum and Containment Flooding

15.1.6.1 Identification of Causes and Accident Description

Loss of containment vacuum and containment flooding events that result in an increase in RCS cooling are NuScale Power Plant US460 standard design-specific events. The containment net volume is less than conventional designs and the NPM is partially immersed in a pool of borated water during normal operation. Because the containment operates at a vacuum during normal operation, air or water ingress into containment could increase heat transfer from the RPV to the reactor pool. This overcooling could lead to higher reactor power, higher RCS pressure, and reduced MCHFR.

The containment evacuation system maintains the containment volume at a vacuum during normal operation. A failure in the containment evacuation system could result in loss of vacuum because containment pressure increases due to evaporation of RCS fluid leaking into containment. If the failure of the containment evacuation system or RCS fluid leakage is sufficiently severe, it could result in a loss of containment vacuum event. If the containment vacuum is lost, heat transfer from the reactor vessel increases. The analysis of a loss of containment vacuum shows a negligible effect on reactor power, and is therefore bounded by a containment flooding event.

The reactor component cooling water system (RCCWS) provides heat removal to the control rod drive system (CRDS). The RCCWS supplies reactor component cooling water to the CNV that then conducts reactor component cooling water to the CRDS piping that passes through containment to provide this function. If piping containing reactor component cooling water leaks or ruptures inside the CNV, a containment flooding event occurs. Other potential containment flooding sources include: FW containing line break, main steam containing line break,