THIS NRC STAFF DRAFT SE HAS BEEN PREPARED AND IS BEING RELEASED TO SUPPORT INTERACTIONS WITH THE ACRS. THIS DRAFT SE HAS NOT BEEN SUBJECT TO FULL NRC MANAGEMENT AND LEGAL REVIEWS AND APPROVALS, AND ITS CONTENTS SHOULD NOT BE INTERPRETED AS OFFICIAL AGENCY POSITION

15 TRANSIENT AND ACCIDENT ANALYSES

This chapter of the final safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 15, "Transient and Accident Analyses," of the NuScale Power, LLC (NuScale or the applicant), US460 Standard Design Approval Application (SDAA), Part 2, "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 1 of the SDAA, dated October 31, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23304A365) as supplemented by subsequent docketed information in ML25057A294 and ML25058A253, and as described in each subsection, which includes changes to FSAR Revision 1. In addition, the staff's regulatory findings documented in this report are based on the latest revision of the subject topical reports (TRs) currently under staff review, as supplemented by subsequent docketed information related to that TR to be included in the final approved TR. In the SDAA, the applicant provided the precise parameter values, as reviewed by the staff in this final safety evaluation (SER), using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this SER to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted. In this chapter, the NRC staff uses the term "non-safety-related" to refer to structures, systems and components (SSCs) that are not classified as "safety-related SSCs" as described in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.2, "Definition." However, among the "non-safety-related" SSCs, there are those that are "important to safety" as that term is used in the General Design Criteria (GDC) listed in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and others that are not considered "important to safety."

15.0 Introduction—Transient and Accident Analysis

15.0.0 Classification and Key Assumptions

This section describes the initiating events and their classification that are evaluated as part of the US460 standard design approval (SDA) design bases. Under FSAR Chapter 15, the following subsections are presented by the applicant and evaluated by the staff:

15.0.0.1 Initiating Event Selection

15.0.0.2 Design-Basis Event Classification

15.0.0.3 Licensing Methodology

15.0.0.4 Initial Conditions

15.0.0.5 Limiting Single Failures

15.0.0.6 Equipment Response and Physical Parameter Assumptions

15.0.0.7 Multiple Module Events

Further, in Part 4 of the SDAA, “US460 Generic Technical Specifications,” the applicant presented Generic Technical Specifications (GTS). Several of these GTS, listed below, apply to most of the Chapter 15 sections of this SER:

- SL 2.1, “Safety Limits”
- Limiting Condition for Operation (LCO) 3.1.1, “Shutdown Margin (SDM)”
- LCO 3.1.3, “Moderator Temperature Coefficient (MTC)”
- LCO 3.1.4, “Rod Group Alignment Limits”
- LCO 3.1.5, “Shutdown Bank Insertion Limits”
- LCO 3.1.6, “Regulating Bank Insertion Limits”
- LCO 3.2.1, “Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)”
- LCO 3.2.2, “Axial Offset (AO)”
- LCO 3.3.1, “Module Protection System (MPS) Instrumentation”
- LCO 3.3.2, “Reactor Trip System (RTS) Logic and Actuation”
- LCO 3.3.3, “Engineered Safety Features Actuation System (ESFAS) Logic and Actuation”
- LCO 3.4.1, “Reactor Coolant System (RCS) Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits”
- LCO 3.4.4, “Reactor Safety Valves (RSVs)”
- LCO 3.4.6, “Chemical and Volume Control (CVCS) Isolation Valves”
- LCO 3.5.1, “Emergency Core Cooling System (ECCS)”
- LCO 3.5.2, “Decay Heat Removal System (DHRS)”
- LCO 3.5.3, “Ultimate Heat Sink”
- LCO 3.7.1, “Main Steam Isolation Valves (MSIVs)”

- LCO 3.7.2, “Feedwater Isolation”

15.0.0.1 Initiating Event Selection

The applicant described its selection of events in FSAR Section 15.0.0.1. Initiating events are the internal events associated with a single NuScale Power Module (NPM) at power. A range of power operations is considered if it is thought to be more limiting for meeting the appropriate acceptance criteria. In general, most of the events described in FSAR Chapter 15, are similar to those at current pressurized-water reactors (PWRs) with some exceptions based on the NuScale design. Design-basis events (DBEs) that are not considered in a typical PWR but are relevant to the NuScale design include loss of containment vacuum, inadvertent operation of the DHRS, and inadvertent operation of an ECCS valve. FSAR Section 15.9, “Stability,” addresses RCS thermal-hydraulic stability.

The applicant stated that, consistent with NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 15.0, “Introduction—Transient and Accident Analyses,” and current light-water reactors (LWRs), the events are divided into seven categories:

(1) increase in heat removal by the secondary system;
(2) decrease in heat removal by the secondary system;
(3) decrease in RCS flow rate;
(4) reactivity and power distribution anomalies;
(5) increase in reactor coolant inventory;
(6) decrease in reactor coolant inventory; and
(7) radioactive release from a subsystem or component.

FSAR Table 15.0-1, “Design-Basis Events,” lists the events selected, the event classification, and the computer codes used for evaluation in FSAR Sections 15.1, “ Increase in Heat Removal by the Secondary System, “ through 15.9, “Stability.” The staff reviewed the table and confirmed that the DBE identification and frequency classification are consistent with the guidance in NuScale Design Specific Review Standard (DSRS) Section 15.0 (ML15355A302). The staff’s evaluations of the listed DBE analyses are found in their respective sections of this SER.

15.0.0.2 Design Basis Event Classification

In FSAR Section 15.0.0.2, “Design Basis Event Classification,” the applicant described how it classified Chapter 15 events. DBE classification by frequency of occurrence is based on four distinct categories:

(1) anticipated abnormal occurrences (AOO)s
(2) infrequent events (IEs)
(3) postulated accidents (PAs)
(4) special events

As stated in FSAR Section 15.0.0.2.1, “Classification by Event Frequency and Type,” events that are expected to occur one or more times during an NPM lifetime are classified as AOOs, consistent with the definition in Appendix A to 10 CFR, Part 50. IEs and PAs are not expected to occur in the lifetime of the plant and allow for the possibility of fuel failure. The applicant may choose to evaluate IEs or PAs against AOO acceptance criteria, as these criteria are more conservative. In general, the applicant stated that events that are not considered to be DBEs are evaluated in FSAR Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation”; however, those beyond-design-basis events (BDBEs) that are explicitly defined by regulation or involve defense-in-depth and common cause failure of digital control systems are addressed in FSAR Chapter 7, “Instrumentation and Controls,” Chapter 8, “Electrical Power,” or Chapter 15. These events are termed “special events.”

The NuScale Power Plant design life is 60 years, and the applicant has conservatively interpreted the criterion “one or more times in NPM life” as including any transient with a frequency of $1x10^{-2}$ events per year or more. The IE category identifies events that have a frequency of less than $1x10^{-2}$ events per year but greater than the frequency of PAs. Neither IEs nor PAs are expected to occur in the lifetime of the plant. IEs have more restrictive radiological acceptance criteria than PAs to keep the overall risk approximately constant across the spectrum of DBEs.

The staff finds the criteria specified by the applicant for analysis of DBEs acceptable and appropriate, since other aspects of the NuScale design (for example, accident consequences, equipment qualification, and source term) rely on the design’s ability to preclude fuel failures during DBEs.

FSAR Table 15.0-2, “Acceptance Criteria – Thermal and Hydraulic Fuel,” gives the thermal-hydraulic acceptance criteria associated with events other than rod ejection and loss-of-coolant accidents (LOCAs). Table 15.0-3, “Acceptance Criteria Specific to Rod Ejection Accidents,” and Table 15.0-4, “Acceptance Criteria Specific to Loss of Coolant Accidents,” provide the acceptance criteria for rod ejection and LOCAs, respectively. The staff agrees with the applicant’s acceptance criteria in FSAR Table 15.0-2 as they are consistent with and satisfy DSRS Section 15.0. Similarly, the staff agrees with the rod ejection acceptance criteria as they are consistent with or more restrictive than those in Regulatory Guide (RG) 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” and the LOCA criteria in 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.” The staff notes that the applicant has chosen to use the more restrictive AOO acceptance criteria given in FSAR Table 15.0-2 for IEs and PAs, except for the rod ejection specific acceptance which is either consistent with or more restrictive than the criteria in RG 1.236. The staff finds this acceptable.

FSAR Table 15.0-2 also includes criteria for events classified as AOOs specifying that an AOO should not develop into more serious plant conditions without other faults occurring independently (typically referred to as event escalation criteria). Consistent with the previous staff determination associated with the NPM-160 standard design, and documented in the safety evaluation for TR-0815-16497, Revision 1, “Safety Classification of Passive Nuclear Power Plant Electrical Systems,” issued February 23, 2018 (ML18054B607 (nonproprietary), ML18054B608 (proprietary)), a passive plant response to an AOO that results in an immediate and direct coolant flowpath to the containment, (such as the inadvertent opening of the ECCS valves due to the loss of the direct current (DC) power system) constitutes loss of the reactor coolant pressure boundary (RCPB) integrity and is considered AOO event escalation. This scenario would be contrary to the defense-in-depth purpose of GDC 15, “Reactor coolant system design,” which expects the RCPB to remain available as a fission product barrier during AOOs. However, the opening of the ECCS valves during normal, planned plant operations, including recovery from an AOO, is not considered event escalation and is deemed acceptable once a safe, stable state has been established. The applicant addressed this criterion by establishing augmented design requirements for the DC power system in FSAR Chapter 8 and documenting in FSAR Section 15.0.0.6.5.3, “Treatment of Augmented Direct Current Power,” that evaluation of the system concludes that its failure is not expected to occur within the lifetime of the module. The staff finds this acceptable because it meets the AOO escalation criteria, including the defense-in-depth purpose of GDC 15, by ensuring that the ECCS valves will not inadvertently open due to loss of the DC power system within the frequency of an AOO and that ECCS actuation following an AOO will only occur during the long-term cooling (LTC) phase which is well after the reactor has been shutdown and RCS pressure and temperature have substantially decreased.

The details of the applicant’s event-specific acceptance criteria are provided in the specific FSAR section for each event.

15.0.0.3 Licensing Methodology

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are in DSRS Section 15.0 and are summarized below.

- 10 CFR Part 20, “Standards for Protection Against Radiation”
- 10 CFR Part 50, especially 10 CFR 50.46 and Appendix A
- 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants”

The following GDC from 10 CFR Part 50, Appendix A, are relevant to DSRS Section 15.0.

- GDC 2, “Design bases for protection against natural phenomena,” as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.
- GDC 4, “Environmental and dynamic effects design bases,” as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and PA conditions, including such effects as pipe whip and jet impingement.
- GDC 5, “Sharing of structures, systems, and components,” as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.
- GDC 10, “Reactor design,” as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including AOOs.
- GDC 12, “Suppression of reactor power oscillations,” as it relates to the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 13, “Instrumentation and control,” as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.

- GDC 15, “Reactor coolant system design,” as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.

- GDC 17, “Electric power systems,” as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety. The safety function for each system (assuming the other system is not working) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the RCPB are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

- GDC 19, “Control room,” as it relates to the requirement that a control room be provided from which personnel can operate the nuclear power unit during both normal operating and accident conditions, including a LOCA.

- GDC 20, “Protection system functions,” as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed SAFDLs during any condition of normal operation, including AOOs.

- GDC 25, “Protection system requirements for reactivity control malfunctions,” as it relates to the requirement that the reactor protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.

- GDC 26, “Reactivity control system redundancy and capability,” as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

- GDC 27, “Combined reactivity control systems capability,” and GDC 28, “Reactivity limits,” as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

- GDC 29, “Protection against anticipated operational occurrences,” as it relates to the design of the protection and reactivity control systems and their performance (i.e., to accomplish their intended safety functions) during AOOs.

- GDC 31, “Fracture prevention of reactor coolant pressure boundary,” as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

- GDC 34, “Residual heat removal,” as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.

- GDC 35, “Emergency core cooling,” as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.

- GDC 55, “Reactor coolant pressure boundary penetrating containment,” as it relates to the isolation requirements of small-diameter lines connected to the primary system.

- GDC 60, “Control of releases of radioactive materials to the environment,” as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.

- GDC 61, “Fuel storage and handling and radioactivity control,” as it relates to the requirement that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and PA conditions.

The applicant incorporated the above criteria in FSAR Section 3.1, “Conformance with U.S. Nuclear Regulatory Commission General Design Criteria,” which the staff finds acceptable. The staff notes that the applicant requested an exemption to the GDC 17 electrical power requirements. In addition, the applicant requested an exemption to the electrical power provisions in GDC 34 and GDC 35 and developed Principal Design Criterion (PDC) 34 and PDC 35. Chapter 8 of this SER contains a detailed discussion of NuScale’s reliance on electrical power and the related exemptions to GDC 17, GDC 34, and GDC 35. The applicant also requested an exemption to GDC 55 and Chapter 6 of this SER contains the exemption evaluation.

15.0.0.4 Initial Conditions

FSAR Table 15.0-6, “Module Initial Conditions Ranges for Design Basis Event Evaluation,” establishes the range of initial conditions that are assumed in FSAR Chapter 15. The values presented in Table 15.0-6 are common to all Chapter 15 events, except where noted in the individual sections. The staff reviewed Table 15.0-6 and found that it identifies the important input parameters and establishes the range of conditions used in FSAR Chapter 15, which is consistent with the guidance in DSRS Section 15.0.

Further, the staff considered that GTS Section 5.6.3, “Core Operating Limits Report” (COLR), paragraph b, lists the analytical methods used to determine the core operating limits in GTS Section 5.6.3, paragraph a. The methodologies that are used to determine these core operating limits are included in the COLR list of references. Therefore, the staff concludes that the applicant included acceptable controls to ensure approved methods are used in determining the numerical values for the technical specification (TS) limits that ensure the initial conditions assumed in the FSAR Chapter 15 analyses are met during operation.

15.0.0.5 Limiting Single Failures

Appendix A to 10 CFR Part 50 describes a single failure as an occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure consistent with 10 CFR Part 50, Appendix A. DSRS Section 15.0.I.8.B, “Sequence of Events and System Operation,” states that the single failure criterion (SFC) applies to safety-related systems or components used to mitigate AOOs or PAs.

FSAR Section 15.0.0.5, states that a component that changes position or state to achieve its safety function is considered an “active” component, while a component that does not change position or state to achieve its safety function is considered a “passive” component. The applicant also considered failure of a check valve an active failure and subject to the SFC consistent with SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs,” dated March 28, 1994 (ML003708068).

The inadvertent actuation block (IAB) valve is a first-of-a-kind, safety-significant, active component integral to the NuScale ECCS. To meet the requirements for the ECCS in 10 CFR Part 50, an applicant must show that it has addressed the SFC. The SFC is defined in 10 CFR Part 50 Appendix K, “ECCS Evaluation Models,” and is derived from the definition of single failure in 10 CFR Part 50, Appendix A. During its review of the NPM-160 design, the staff noted that although the applicant assumed a single failure of a main ECCS valve to open, the applicant did not apply the SFC to the IAB valve in regard to the valve’s function to close. Because the NPM-20 design incorporates IAB valves, although only to the reactor recirculation valves (RRVs) and not also to the reactor vent valves (RVVs) as in the NPM-160 design, the following information regarding the decision on the application of the SFC to the IAB valves for the NPM-160 design also applies to the IABs present in the NPM-20 design.

For the NPM-160 design, NuScale disagreed with the staff’s application of the SFC to the IAB valve, which led the staff to request Commission direction to resolve this issue in SECY-19-0036 “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” dated April 11, 2019 (ML19060A081). In SECY-19-0036, the staff summarized the NRC’s historical practice for applying the SFC. Specifically, the staff summarized SECY-77-439, "Single Failure Criterion," dated August 17, 1977 (ML060260236), in which it informed the Commission how the staff then generally applied the SFC, and, SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994 (ML003708068), in which the staff requested Commission direction on applying the SFC in specified fact- or application-specific circumstances. In view of this historical practice, in SECY-19-0036, the staff requested Commission direction on applying the SFC to the IAB valve’s function to close.

In response to the paper, the Commission directed the staff in Memorandum (SRM)-SECY-19-0036, dated July 2, 2019 (ML19183A408), to “review Chapter 15 of the NuScale Design Certification Application without assuming a single active failure of the inadvertent actuation block valve to close.” The Commission further stated that “[t]his approach is consistent with the Commission's safety goal policy and associated core damage and large release frequency goals and existing Commission direction on the use of risk-informed decision-making, as articulated in the 1995 Policy Statement on the Use of Probabilistic Risk

Assessment Methods in Nuclear Regulatory Activities and the White Paper on Risk-Informed and Performance-Based Regulation (in SRM SECY- 98-0144 and Yellow Announcement 99-019)."

Based on the staff's historic application of the SFC and Commission direction on the subject, as described in SECY-77-439; SRM-SECY-94-084, dated June 30, 1994 (ML003708098); and SRM-SECY-19-0036, the NRC has retained some discretion, in fact- or application-specific circumstances, to decide when to apply the SFC. The Commission's decision in SRM-SECY-19-0036 provides direction regarding the appropriate application and interpretation of the regulatory requirements in 10 CFR Part 50 to the NuScale IAB valve's function to close. This decision is similar to those in previous Commission documents that addressed the use of the SFC and provided clarification on when to apply the SFC in other specific instances.

NuScale requested an exemption from the GDC 35 requirements for electric power. The staff evaluates this exemption request in Chapter 8 of this SER. NuScale's PDC 35 is functionally identical to GDC 35, except with respect to the discussion regarding electric power. Based on the above discussion, including redundancy of the ECCS to perform its design function assuming a single failure of a main ECCS valve to open, and the Commission determination that the SFC does not need to be assumed for the NuScale IAB valves' functions to close (applicable to the RRVs on the NPM-20 design), as well as the conclusion in SER Section 15.0.0.6.2 relative to the RVVs, the staff concludes that the ECCS system satisfies PDC 35 with respect to demonstrating that "the safety system function can be accomplished, assuming a single failure." The staff also discusses this issue in Section 6.3.4.2.7 of this SER. Section 6.3.4.1.6 of this SER contains the staff's evaluation of NuScale's PDC 35 with regard to sufficient core cooling in the ECCS system. On the same basis, the staff finds that the requirements of 10 CFR Part 50 Appendix K to assume "the most damaging single failure of ECCS equipment has taken place" for accident evaluations under the associated requirements of 10 CFR 50.46 are met for NuScale's ECCS system. Therefore, evaluation of a single failure of the NuScale IAB valve's function to close is not necessary for the evaluation of DBEs in Chapter 6, "Engineered Safety Features" or Chapter 15.

Passive failures can be initiating events, such as the assumed mechanical failure of an RVV/RRV in FSAR Section 15.6.6, "Inadvertent Operation of Emergency Core Cooling System," but passive failures of fluid systems are considered only on a long-term basis, which the applicant defines as greater than 24 hours after event initiation.

FSAR Section 15.0.0.5 states that active and passive failures are considered for electrical components. Protective actions must be accomplished in the presence of a single detectable failure. The effects of nondetachable failures are considered concurrently as part of the most-limiting single failure.

FSAR Chapter 15 evaluates a range of available electrical power assumptions. The range of electrical power assumptions, as discussed in Section 15.0.0.6.2 of this SER and evaluated for each event, is not considered a single failure but instead is used to determine if operation of electrical power systems that are not safety-related could impair the plant response to the event. Therefore, the worst single failure of a component is assumed for each electrical power system availability assumption.

Operator errors are considered as initiating events consistent with the guidance given in DSRS Section 15.0. The applicant stated that an error of omission is not relevant as the design is such that no operator action is necessary to mitigate a DBE for the first 72 hours. The applicant also stated that errors of commission, where no operator action is necessary, but an erroneous action is taken, are bounded by the worst case single failure assumption. During an audit, (ML24211A089) the staff confirmed that this statement is supported based on the following reasons:

- Single actions taken on safety-related SSCs are bounded by the SFC since a safety-related SSC cannot be affected by a single action from the main control room (MCR), with the exception of initiating an engineered safety feature (ESF).

- Control room operators cannot manipulate safety-related SSCs in the MCR except through the use of the module protection system (MPS) hard-wired manual actuation switches located at the standup panel for each unit. Operation of any of these switches is infrequent, is directed by procedure, and normally requires a prior peer-check.

- Operation of these switches is also expected to receive supervisory oversight, and because of the switches' location, their operation is conspicuous to the operating crew.

- An operator cannot override the MPS either before or after initiation, with the exception of containment isolation override to support either adding inventory to the reactor vessel using the CVCS or to the containment using the containment flooding and drain system, as well as sampling through the containment evacuation system (CES).

- Once an MPS setpoint is reached, the associated safety-related SSCs will transition to their single safety position. The containment isolation override function is required only during highly improbable BDBEs, which are addressed in FSAR Chapter 19.

- The containment isolation override function requires multiple deliberate steps, which are directed by procedures. The Conduct of Operations and generally accepted industry standards on human performance and use of error reduction tools ensure that a peer check and proper supervisory oversight would be provided to complete this action. To accidentally perform this action in error or to complete this action on the wrong unit is not deemed credible.

- Operator errors, including errors of commission, are considered when identifying initiating events. The Chapter 15 analysis models normal operation of systems that are not safety-related that increase the consequences of the event. With the exception of the DC power system described in Section 15.0.0.6.2 of this SER, normal operation of systems that are not safety-related that improve (decrease) the consequences of the event are not modeled. Therefore, an operator action of commission performed in error on SSCs that are not safety-related, that results in normal SSC operation or disables SSC operation and increases the consequences of the event, is bounded in the Chapter 15 analysis.

- Multiple operator errors or errors that result in common-mode failures are considered BDBEs. Chapter 19 analyzes these events.

The determination that operator errors of commission are bounded by a single failure of a safety-related SSC is dependent on the design of the manual operator capabilities and the likelihood of an erroneous operator action. Chapter 18 of this SER gives the staff's evaluation of human factors engineering, including an audit of Conduct of Operations defined by NuScale.

For a particular transient, the limiting single failure for one acceptance criterion may be different than the limiting single failure for a different acceptance criterion for the same DBE. The limiting single failures for Chapter 15 events are described with the event analysis and are identified in FSAR Table 15.0-9, "Referenced Topical and Technical Reports". The staff evaluates these failures as described in each of the design basis event subsections of FSAR Chapter 15, in the associated subsections of this SER chapter.

15.0.0.6 Equipment Response and Physical Parameter Assumptions

FSAR Section 15.0.0.6.1 through Section 15.0.0.6.3 address control rod assembly (CRA) insertion characteristics, decay heat, and ESF characteristics, respectively.

The time for inserting control rods directly affects the amount of heat that must be removed from the core in response to a DBE. SRP Section 4.3 describes the analytical basis for the CRA insertion rates and reactivity effect as a function of time. The analyses for the Chapter 15 DBEs apply additional conservatism to the reactivity insertion rate provided in FSAR Section 4.3, "Nuclear Design," to bound potential plant operating conditions. FSAR Figure 4.3-20, "Control Rod Position versus Time after Trip," shows the control rod position versus time, and Figure 15.0-1, "Reactor Trip Reactivity versus Time After Trip," shows the relative cumulative reactor trip reactivity worth versus time. The use of bounding insertion times provides conservative results for DBE analyses. GTS 3.1.4 specifies drop-time testing requirements.

Bounding values for decay heat are designated to represent the maximum decay heat of the core following an event and conservative minimum decay heats for the cooldown events. The NuScale Reactor Excursion and Leak Analysis Program, Version 5 (NRELAP5), uses the 1973 American Nuclear Society (ANS) 5.1, "Decay Heat Power in Light Water Reactors," decay heat standard to represent bounding decay heat. The LOCA methodology calculates fission product decay heat using a bounding form of the 1973 ANS decay heat standard with a 20 percent margin to address uncertainty added to the base value. The bounding form of the 1973 ANS standard in NRELAP5 is conservative relative to the 1971 ANS standard specified in 10 CFR Part 50, Appendix K. The applicant requested an exemption from 10 CFR Part 50, Appendix K, for certain phenomena not encountered in the NuScale NPM during a LOCA because of the NPM design. The staff's evaluation of this exemption is in Section 15.0.2, "Review of Transient and Accident Analysis Methods," of this SER.

For non-LOCAs, the model also uses the conservative 1973 ANS decay heat standard, which is varied by utilizing different decay heat multipliers and specifying whether to include the actinide contribution.

The following decay heat multipliers are used:

- minimum = use multiplier of 0.8 while excluding the actinide contribution
- maximum = use multiplier of 1.0 while including the actinide contribution

The staff evaluated the acceptability of the multipliers used for the non-LOCA analyses in the staff's SER (ML25051A200 (nonproprietary), ML24334A050 (proprietary)) of topical report (TR)-0516-49416-P, Revision 4, "Non-Loss-of-Coolant Accident Analysis Methodology," dated January 5, 2023 (ML23005A305 (nonproprietary), ML23005A306 (proprietary)).

For extended passive cooling phases following non-LOCA events, and boron transport and precipitation analyses, decay heat assumptions are based on the use of the Oak Ridge Isotope Generation (ORIGEN) computer code. The acceptability of the use of ORIGEN to calculate decay heat conditions is addressed in Section 15.0.5 of this SER and associated staff SER (ML25051A242 (nonproprietary), ML24355A062 (proprietary)) of TR-124587-P, Revision 0, "Extended Passive Cooling and Reactivity Control Methodology," dated January 5, 2023 (ML23005A308 (nonproprietary), ML23005A309 (proprietary)).

NuScale ESF systems include the containment systems (FSAR Section 6.2), ECCS (FSAR Section 6.3), and DHRS (FSAR Section 5.4.3), as described in the FSAR. The DHRS provides cooling for non-LOCA DBEs when normal secondary side cooling is unavailable or otherwise not used, and during LOCA events prior to ECCS actuation (particularly small LOCAs where the steam generators (SGs) are covered for a longer period of time). The DHRS is designed to remove post reactor trip residual and core decay heat from operating conditions and transition the NPM to safe-shutdown conditions without reliance on external power. FSAR Section 5.4.3 further describes the DHRS. In conjunction with the containment heat removal function of containment, the ECCS provides a means of core decay heat removal for LOCAs.

FSAR Section 15.0.0.6.3, "Engineered Safety Features Characteristics," states the that ECCS valves and the DHRS valves do not rely on electrical power or on non-safety-related support systems for actuation. After actuation, the valves do not require a subsequent change of state or continuous availability of power to maintain their intended safety functions. One RRV and one RVV are required for successful ECCS operation. If the redundant non-safety-related DC power to the MPS or the ECCS and DHRS valve actuators is lost, the RVVs actuate immediately and the RRVs open once differential pressure goes below the IAB valve operating threshold. The staff notes that actuation of the ECCS valves is not the only safety-related function performed by the system. For example, the function of the ECCS valves to remain closed when there is not a valid ECCS actuation signal is necessary to ensure RCPB integrity during normal and off-normal operating conditions, and safety limits are not exceeded during certain AOO transients (such as increase in heat removal and reactivity events). Accordingly, for initiating events that do not assume coincident loss of DC power to the ECCS valves or MPS, the continued operation of the DC power system is relied on in FSAR Chapter 15 safety analysis to prevent the RVVs from immediately opening thus ensuring the RCPB integrity is maintained and safety limits are satisfied. Section 6.3 of this SER contains the staff's detailed review of ECCS safety functions and Section 15.0.0.6.2 of this SER includes a more detailed discussion on the treatment of non-safety-electrical power systems.

The staff evaluates the ability to successfully mitigate non-LOCA and LOCA events using the DHRS or ECCS including into the long-term (up to 72 hours following the initiating event) as part of its overall FSAR Chapter 15 review.

15.0.0.6.1 Required Operator Actions

The applicant stated in FSAR Section 15.0.0.6.4, that no operator actions are credited to mitigate DBEs for at least 72 hours even with assumed failures. The staff evaluated the ability of the design to mitigate DBEs without operator action as part of the overall FSAR Chapter 15 review. None of the Chapter 15 events credit operator actions within the first 72 hours, and the staff's review of the design's capability to mitigate each DBE is contained in the Technical Evaluation section of each Chapter 15 subsection in this report.

15.0.0.6.2 Availability of Power

Normal alternating current (AC) power systems are not safety-related and not credited to mitigate Chapter 15 events. The normal AC power systems consist of the following:

- EHVS (high-voltage (13.8-kilovolt (kV)) AC electrical system and switchyard)
- EMVS (medium-voltage (4.16-kV) AC electrical distribution system)
- ELVS (low-voltage (480-volt (V) and 120-V) AC electrical distribution system)

The onsite DC power systems are not safety-related and are stated to not be credited to mitigate Chapter 15 events in most cases, as described further below. The DC power systems consist of the following:

- EDAS (augmented DC power system to supply essential loads) and
- EDNS (normal DC power system to supply nonessential loads).

The loss of normal AC power causes the MPS to initiate a reactor trip, actuate the DHRS, and close the containment isolation valves (CIVs). The loss of normal AC power also causes the loss of the EDAS chargers causing the EDAS to rely on backup batteries. If the augmented DC power system (EDAS) supply to the MPS or the ECCS and DHRS valves is lost, the ECCS valves open. Alternatively, at 8 hours after a loss of normal AC power to the EDAS battery chargers, the MPS actuates the ECCS valves causing them to open. If the 8-hour ECCS actuation is manually bypassed during the first 8 hours, the MPS load sheds the ECCS valves at 24 hours, causing them to open. When the EDAS supply is lost or shed or ECCS is actuated, RCS coolant is immediately discharged into containment through the RVVs, and subsequently through the RRVs when the IAB valve operating pressure threshold is reached. As such, the loss of EDAS at any time, including both normal power operation and during the progression of an AOO, results in high-pressure discharge of reactor coolant into containment and a loss of the RCS fission product barrier.

As no power systems in the design are designated as safety-related, several loss of power scenarios are evaluated to ensure that the FSAR Chapter 15 acceptance criteria are met. The applicant evaluated the following loss of power scenarios:

- Loss of normal AC either at the time of the initiating event or at the time of the turbine trip (TT). After 24 hours, the ECCS valves move to their fail-safe open position.

* Loss of normal DC power (EDNS) and normal AC. Power to the reactor trip breakers is provided via the EDNS, so this scenario is the same as a loss of normal AC with the addition of reactor trip at the time power is lost.

* Loss of the augmented DC power system (EDAS), EDNS, and normal AC at the time of the initiating event. This scenario results in a reactor trip, actuation of DHRS, and closure of CIVs. The RVVs move to their fail-safe open position when power is lost, and the RRVs move to their fail-safe open position when RCS pressure drops below the IAB valve operating pressure threshold.

Also evaluated are the scenarios in which power, AC or DC, remains, if the consequences of the event are more limiting.

The FSAR does not evaluate scenarios where EDAS is lost subsequent to an initiating event (after time zero) during the event progression. For AOO events where the system energy increases over either a short or extended period of time, a loss of EDAS can result in more severe consequences in terms of fuel and containment figures of merit than a loss of EDAS at the time of the initiating event. In these cases, staff determined that the EDAS system is relied on in the safety analysis to mitigate the consequences of the progression of those AOOs through maintaining the RCPB and enabling safe shutdown of the module (i.e., the safety analysis assumes EDAS functions to maintain the ECCS valves in the closed position). Examples of events where EDAS is assumed to remain functional during the entire design-basis period and perform these mitigating functions, includes, but is not limited to, decrease in feedwater temperature (FSAR Section 15.1.1), increase in feedwater flow (FSAR Section 15.1.2), increase in steam flow (FSAR Section 15.1.3), steam pipe failures (FSAR Section 15.1.5), and uncontrolled rod withdrawal at power (FSAR Section 15.4.2). Therefore, the staff concludes that the EDAS is needed to meet the Chapter 15 safety analysis acceptance criteria and similarly is relied on to protect two fission product barriers (i.e., fuel clad and RCS barriers). The staff audited engineering documentation to confirm these consequences.

The EDAS is subject to augmented quality requirements. The applicant stated that a loss of EDAS is not expected to occur during the life of a module. The staff did not validate this assertion, and the applicant did not quantify the potential degree of increased consequences if EDAS is not relied on to function during the Chapter 15 events. The staff review of the EDAS design, including its classification, is documented in Section 8.3.2 of this evaluation. The staff review of the EDAS modeling in the probabilistic risk assessment is in Section 19.1 of this SER.

15.0.0.6.3 Treatment of Systems that Are Not Safety-Related

Systems that are not safety-related are assumed to function if their normal operation is assumed to increase the consequences associated with the event and are not assumed to change state to lessen or mitigate the consequences of the event.

In FSAR Section 15.0.0.6.6, “Treatment of Non-Safety Related Systems,” the applicant stated that the treatment of non-safety-related systems for the DBEs, other than nonsafety-related power systems that are treated as described in Section 15.0.0.6.5, is as follows:

* Non-safety-related system normal operation that increases the consequences of the event is modeled.

- Non-safety-related system normal operation that improves (decreases) the consequences of the event is not modeled.

- Non-safety-related system normal operation that does not significantly alter the consequences of the event may be modeled.

- The failure of a non-safety-related system to a worst-state condition is not considered except as an event initiator.

- Non-safety-related equipment is evaluated to consider the licensing basis assumptions defining the events, including external events; environmental effects; and offsite and onsite power availability.

The above criteria are consistent with those discussed in DSRS Section 15.0.I.8.B, which specifies that only safety-related systems or components are used in mitigating AOOs and PAs.

DSRS Section 15.0.I.8.B states that the reviewer may consider the licensee’s technical justifications for the operation of systems or components that are not safety-related, for example, when used as backup protection and when not disabled, except by a detectable, random, and independent failure. The applicant’s design uses equipment that is not safety-related as a backup to safety-related equipment in the following three areas:

(1) The non-safety-related secondary MSIV serves as the backup isolation device to the safety-related MSIV for isolating the main steam piping penetrating containment when the safety-related MSIV is assumed to fail.

(2) The non-safety-related feedwater regulating valve (FWRV) serves as the backup isolation device to the safety-related feedwater isolation valve (FWIV) for isolating the feedwater system (FWS) piping penetrating the containment when the FWIV is assumed to fail.

(3) The non-safety-related feedwater check valve serves as the backup isolation device to the safety-related feedwater check valve for isolating the DHRS when reverse flow is experienced during a break in the FWS piping.

The non-safety-related secondary main steam isolation bypass valve (MSIBV) is credited as a backup isolation device to the safety-related MSIBV for isolation of the main steam piping penetrating containment. The basis for relying on a component that is not safety-related to serve as a backup to a safety-related component is described in NUREG-0138, “Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR to NRR Staff,” issued November 1976 (ML13267A423).

As discussed in NUREG-0138, Issue 1, the staff found it acceptable to credit the control and stop valves, which are not safety-related, during a main steamline break event because the consequences are less than those of a LOCA and the combined reliability of the turbine stop and control valves is similar to that of a safety-related component. NUREG-0138, Issue 1, states that valves that are not safety-related can be used as a backup to safety-related valves in the FWS, assuming that the same evaluation criteria (i.e., reliability and consequence) apply.

The staff finds the use of the non-safety-related main steam and feedwater valves identified above acceptable because their design and treatment are consistent with the staff position of NUREG-0138, Issue 1. Specifically, the non-safety-related main steam and feedwater valves will have a demonstrated reliability due to augmented design, quality, and testing requirements, and TS surveillance and operability requirements. The staff notes that crediting the secondary MSIV to mitigate a SG tube rupture is beyond the scope of NUREG-0138, Issue 1, because a tube rupture is a breach of the primary system pressure boundary not previously addressed by the staff. The staff evaluates crediting the non-safety-related secondary MSIV for a steam generator tube failure (SGTF) event in Section 15.6.3, "Steam Generator Tube Failure," of this SER. Section 3.9.6 of this SER contains the staff's review of the augmented quality and testing requirements applied to these components, and Section 10.3 of this SER contains the staff's review of the main steam system, including the non-safety-related main steam isolation valves. As systems that are not safety-related are not used to mitigate an AOO or PA, except to serve as a backup function to safety-related components, and with the exception of EDAS, the staff finds the applicant's use of equipment that is not safety-related acceptable.

15.0.0.7 Multiple Module Events

Chapter 15 DBEs are analyzed for a single NPM. FSAR Section 19.1.7, "Multiple Module Risk Evaluation," discusses consideration of multi-module events. The only safety-related shared system is the ultimate heat sink, which is evaluated in SER Section 9.2.5. Shared systems that are not safety-related are not credited in the NuScale transient and accident analyses; however, the failure of these systems is considered in the staff's evaluation presented in this chapter of the SER.

15.0.1 Radiological Consequence Analyses Using Alternative Source Terms

SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," is focused on the application of alternative source terms to operating reactors and is not directly applicable to the NuScale SDAA. The applicant used a modified version of the alternative source term methodology to evaluate radiological consequences of DBEs including the core damage event (CDE). Section 15.0.3 of this SER discusses the staff's evaluation of the NuScale DBA radiological consequence analyses.

15.0.2 Review of Transient and Accident Analysis Methods

15.0.2.1 Introduction

FSAR Section 15.0.2 summarizes the analysis methods and computer codes used in non-LOCA safety analyses, LOCA evaluations, and Extended Passive Cooling (XPC) evaluations. In addition, FSAR Table 15.0-9, lists the topical or technical reports used as the basis for analysis of each AOO and PA. These reports are also incorporated by reference into the FSAR as noted in Table 1.6-1, "NuScale Referenced Topical Reports," supplemented by ML24346A132**.** The following SER section summarizes the methodology and computer codes used for relevant DBEs. Several different license methodologies are required to provide the neutronic, thermal-hydraulic, and radiological responses of the plant to AOOs, PAs, and IEs. FSAR Table 15.0-1, lists the computer codes used for each DBE.

15.0.2.2 Summary of Application

FSAR Section 15.0.2.1, "Licensing Methodology (Evaluation Models)," describes the licensing methodology relevant to transient and accident analyses for non-LOCA, LOCA, and LTC events. FSAR Sections 15.1 through 15.6.4, and 15.6.6 through 15.9 describe safety analyses for non-LOCA events. FSAR Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," describes the safety analysis for LOCA events. FSAR Section 15.0.5 describes the safety analysis for LTC events which is also referred to as extended passive cooling (XPC). The NuScale LOCA evaluation model (EM) was designed to meet the applicable requirements of 10 CFR Part 50, Appendix K, with some exceptions as described in the related Appendix K exemption request evaluated in Section 15.0.2.4.1 of this SER, and the 10 CFR 50.46 acceptance criteria. The non-LOCA analysis methodology was then designed to build on the NRELAP5 LOCA EM to address specific phenomena important for each non-LOCA event. The XPC analysis methodology was designed to build on the LOCA EM and non-LOCA EM to address specific phenomena important for the period after short term LOCA and non-LOCA events.

In FSAR Section 15.0.2, the applicant described the analytical methods and computer programs used in non-LOCA safety analysis, LOCA evaluation, XPC evaluation, and other AOOs as detailed below.

15.0.2.2.1 Loss-of-Coolant Accident Methodology

FSAR Section 15.6.5 states that LOCA and ECCS valve opening event analyses are performed using NRELAP5, as described further below. FSAR Table 1.6-1, supplemented by ML24346A132, incorporates by reference TR-0516-49422-P, "Loss-of-Coolant Accident Evaluation Model," Revision 3, dated January 8, 2023 (ML23008A002 (nonproprietary), ML23008A003 (proprietary)). TR-0516-49422-P, Revision 3, uses the NuScale NRELAP5 systems computer code that is a modification and extension of the Idaho National Laboratory RELAP5-3D computer code. As described in TR-0516-49422-P, Revision 3, the applicant modified selected models and correlations to address unique features and phenomena of the NPM design and comply with the requirements of 10 CFR Part 50, Appendix K. In Section 10 of the SDAA Part 7, "Exemptions," the applicant requested exemptions from the requirements of 10 CFR Part 50, Appendix K, that the applicant stated are not applicable to the NuScale design.

NRELAP5 employs a combination of proven RELAP5 features, models, and components as well as new and advanced features and components requiring new correlations and models to simulate the needed operating conditions and component system behavior. Of particular importance is the use of the containment vessel (CNV) as an integral part of the ECCS, the modeling of condensation under high pressure conditions, and the addition of a new hydrodynamic component for the helical coil SGs.

TR-0516-49422-P, Revision 3, is based on the deterministic ECCS performance calculation approach detailed in Appendix K to 10 CFR Part 50 and 10 CFR 50.46(a)(ii). Since the NPM is designed so that there is no core uncovery or heatup for design-basis LOCAs, the applicant indicated significant margins to the peak cladding temperature (PCT) and the other criteria (10 CFR 50.46(b)(2) through (b)(4)), such that the relevant figures of merit are not PCT but (1) collapsed liquid water level in the core, (2) the critical heat flux ratio (CHFR), and (3) containment pressure and temperature. Additionally, since the NPM is designed not to reach core uncovery, the applicant's LOCA methodology does not address post-CHF heat transfer phenomena, including cladding oxidation, hydrogen production, or clad geometry changes such as swell and rupture. This is reflected in the requested exemption from the requirements in 10 CFR Part 50, Appendix K, as follows: I.A.4 (decay heat model), I.A.5 (Baker-Just equation for metal water reaction), I.B (cladding swelling and rupture), I.C.1.b (Moody model for two-phase discharge), and I.C.5.a (post-CHF heat transfer modeling).

The progression of the LOCA event for the NPM is slower than for conventional PWRs since break sizes are significantly smaller. In addition, the ECCS includes venting RVVs and recirculation RRVs that open to remove core decay heat by establishing stabilized recirculating flow from the containment back to the reactor pressure vessel (RPV) via boiling in the core and condensing in the CNV. TR-0516-49422-P, Revision 3, addresses the first four criteria of 10 CFR 50.46(b), and TR-124587-P, Revision 0, which is incorporated by reference in FSAR Table 1.6-1, supplemented by ML24346A132, addresses the LTC requirement of 10 CFR 50.46(b)(4) and (b)(5) for the first 72 hours of a LOCA. The long-term transient progression is analyzed using the methodology in TR-124587-P, Revision 0, which is incorporated by reference in FSAR Table 1.6-1, as supplemented by ML24346A132.

15.0.2.2.2 Non-Loss-of-Coolant Accident Methodology

FSAR Section 15.0.2.1 states that the non-LOCA analysis methodology builds on the NRELAP5 LOCA model. Additionally, FSAR Table 1.6-1, supplemented by ML24346A132, incorporates by reference TR-0516-49416-P, Revision 4. TR-0516-49416-P, Revision 4, the non-LOCA TR, describes event-specific methodologies for non-LOCA events including initial condition and parameter biases that present the greatest challenge to acceptance criteria. The report does not include evaluation of the minimum critical heat flux ratio (MCHFR) or radiological consequences, but it does describe the interface with the downstream subchannel and accident radiological analyses. The non-LOCA analysis methodology covers the short-term portion of the transient. The long-term transient progression is analyzed using the methodology in TR-124587-P, Revision 0, which is incorporated by reference in FSAR Table 1.6-1, as supplemented by ML24346A132.

FSAR Section 15.0.2.2.2, "Non-Loss-of-Coolant Accident Methodology" states that the rod ejection methodology is based on the use of CASMO5, SIMULATE5, SIMULATE-3K (S3K), NRELAP5, VIPRE-01, and the subchannel methods described in Section 15.0.2.2.4 of this SER. The CASMO5/SIMULATE5 code package for reactor core physics parameters is used with NRELAP5 for thermohydraulic input into the subchannel analysis. The nuclear analysis portion of the rod ejection transients is performed with 3D space-time kinetics code SIMULATE-3K. VIPRE-01 is used to perform the subchannel analysis and calculate the MCHFR and other figures of merit specific to the rod ejection transients. This methodology is presented and reviewed in TR-0716-50350-P, Revision 3, "Rod Ejection Accident Methodology," dated October 20, 2023 (ML23293A292), which is incorporated by reference in FSAR Table 1.6-1, as supplemented by ML24346A132.

15.0.2.2.3 Flow Stability

FSAR Section 15.0.2.2.3, "Flow Stability," states that the NuScale proprietary code, PIM, is used to demonstrate system stability at steady-state operation. Additionally, FSAR

Section 15.0.2.2.3 references TR-0516-49417-P-A, Revision 1 “Evaluation Methodology for Stability Analysis of the NuScale Power Module,”, issued March 2020, (ML20086Q664 (nonproprietary), ML20086Q668 (proprietary)), which is incorporated by reference in FSAR Table 1.6-1, supplemented by ML24346A132.

15.0.2.2.4 Subchannel Analysis

FSAR Section 15.0.2.3, “Subchannel Analysis,” states that subchannel analyses are performed using VIPRE-01 (Versatile Internals and Component Program for Reactors; Electric Power Research Institute, [EPRI]) with the NuScale-specific CHF correlations. Additionally, Section 15.0.2.3 references TR-0915-17564-A, “Subchannel Analysis Methodology,” Revision 2, issued February 2019 (ADAMS Accession No. ML19067A256), and NuScale Power, LLC, “Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2,” TR-108601-P-A, Revision 4 (ML24106A160/ML24106A161), which are incorporated by reference in FSAR Table 1.6-1, supplemented by ML24346A132. FSAR Section 15.0.2.3 also references FSAR Section 4.4.4, TR-0116-21012, “NuScale Power Critical Heat Flux Correlations,” Revision 1, issued December 2018 (ML18360A632) and NuScale Power, LLC, “Applicability Range Extension of NSP4 Critical Heat Flux Correlation, Supplement 1 to TR-0116-21012-P-A, Revision 1,” TR-107522-P-A, Revision 1 (ML23118A376), which are incorporated by reference in FSAR Table 1.6-1, as supplemented by ML24346A132 .

15.0.2.2.5 Radiological Consequence Analysis Methodology for the Design-Basis Accident

The computer codes used by the applicant to calculate DBA doses are described in TR-0915-17565-P-A, Revision 4 “Accident Source Term Methodology,” issued February 26, 2020 (ML20057G132 (nonproprietary), ML20057G133(proprietary)).

ITAAC: There are no inspection, test, analysis, and acceptance criteria (ITAAC) for this area of review.

Technical Specifications: The GTS associated with FSAR Section 15.0.2, are related to analytical methods used to determine core operating limits and are described in Part 4 of the applicant’s Standard Plant SDAA, Reporting Requirement 5.6.3, “Core Operating Limits Report [COLR].”

Technical Reports: The applicable technical reports associated with this section are listed in FSAR Table 15.0-9.

15.0.2.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 52.137 requires an FSAR to describe and analyze the design and performance of SSCs.
- The evaluation methodologies described in FSAR Section 15.0.2, form a partial basis for demonstrating compliance with the regulations identified in Section 15.0.0.3 of this SER.
- 10 CFR Part 50, Appendix K, provides the required and acceptable features of EMs.

15.0.2.4 Technical Evaluation

15.0.2.4.1 Loss-of-Coolant Accident Methodology

The staff's review of TR-0516-49422-P, Revision 3, supplemented by docketed markups to the TR to be incorporated into the final TR, is documented in the SER for the TR (ML25007A192 (nonproprietary), ML24312A004 (proprietary)). The staff reviewed NRELAP5 Version 1.7 and NPM base model Version 5, which NuScale provided by letter dated February 27, 2025 (ML25058A422), to confirm that the code, code modifications, and modeling adequately address the requirements of RG 1.203, "Transient and Accident Analysis Methods." The validation and modeling bases developed for the LOCA methodology is extended by the applicant to support modeling bases for NRELAP5 analysis used in non-LOCA and XPC transient analyses. The NRELAP5 code and input models used are key components of the applicant's methodology.

As part of its review, the staff reviewed the sections of the LOCA TR that cover the methodology for the EM for inadvertent opening of RPV valves (IORVs) and the methodology for containment response analysis. The staff's evaluation of the methodology relative to these requirements and the calculational framework established in RG 1.203 is documented in the SER for TR-0516-49422-P, Revision 3.

The staff noted that NRELAP5 models used to support the Chapter 15 accident analysis were built from a base model that used a computer aided design model to produce NPM geometry inputs. The modeling approach used was consistent with that used in the NIST-1/NIST-2 facility benchmark models and the base model was developed generically for non-LOCA application. The LOCA models include a similar modeling of secondary-side feedwater system components as the non-LOCA modeling, and the addition of a "hot assembly" which is hydraulically connected from the lower plenum to the riser. NRC staff reviewed the NuScale LOCA EM methodology described in TR-0516-49422-P, Revision 3, the analysis modeling approach outlined, and the NRELAP5 input options and models selected to represent the NuScale NPM transient behavior and determined it is suitable for performing LOCA safety analysis, ECCS valve opening analysis and containment response analysis, subject to the limitations and conditions (L&Cs) in the staff's SER.

After the short-term LOCA and non-LOCA events transition to the XPC period, the methodology outlined in TR-124587-P, Revision 0 is used to assess the long-term event progression and response. The staff's review of TR-124587-P, Revision 0 is documented in the SER for the TR (ML25051A242 (nonproprietary), ML24355A062 (proprietary)). The staff reviewed TR-124587-P, Revision 0 to confirm that the code, code modifications, and modeling adequately address the requirements of RG 1.203. The validation and modeling bases developed for the LOCA and non-LOCA methodologies is extended by the applicant to support modeling bases for NRELAP5 analysis used in XPC analyses.

NRC staff reviewed (1) the NuScale XPC EM methodology described in TR-124587, Revision 0, supplemented by docketed markups to the TR to be incorporated into the final TR, (2) the analysis modeling approach outlined, and (3) the inputs and models selected to represent the NuScale NPM-20 event behavior and determined it is suitable for performing Extended Passive Cooling safety analysis, subject to the L&Cs in the staff's SER.

Exemption from 10 CFR Part 50, Appendix K, Emergency Core Cooling System Evaluation Model

The staff reviewed the applicant's request for exemption from certain requirements of 10 CFR Part 50, Appendix K, as described in SDAA Part 7, Section 10, related to certain phenomena not encountered in the NuScale NPM during a LOCA because of the NPM design. The applicant stated that the underlying purpose of the rule is met because the NPM has been designed to avoid those phenomena, and the model conservatively calculates the consequences of postulated LOCAs. The specific requirements in 10 CFR Part 50, Appendix K, from which the applicant requested exemption are the following:

- I.A.4, as it relates to the heat generation rates from radioactive decay of fission products.
- I.A.5, as it relates to the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction.
- I.B, with respect to inclusion of a provision for predicting cladding swelling and rupture.
- I.C.1.b, with respect to calculation of the discharge rate for all times after the discharging fluid has been calculated to be two-phase.
- I.C.5.a, with respect to post-CHF correlations of heat transfer from the fuel cladding to the surrounding fluid.

<u>Evaluation for Meeting the Exemption Criteria of 10 CFR 50.12, Specific Exemptions</u>

Pursuant to 10 CFR 52.7, "Specific Exemptions," the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 52. As 10 CFR 52.7 further states, the Commission's consideration will be governed by 10 CFR 50.12, "Specific exemptions," which states that an exemption may be granted when: (1) the exemptions are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security; and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present for the NRC to consider granting an exemption request.

Authorized by Law

The NRC staff has determined that granting the applicant's proposed exemptions will not result in a violation of the Atomic Energy Act (AEA) of 1954, as amended, or the Commission's regulations because, as stated above, 10 CFR Part 52, allows the NRC to grant exemptions. The staff also determined that granting the applicant's proposed exemptions will not result in a violation of the AEA, or the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is authorized by law.

No Undue Risk to Public Health and Safety

The staff review of the exemption request to determine whether the exemption would present an undue risk to public health and safety (10 CFR 50.12(a)(1)) is described below. In its exemption request, the applicant stated that it has designed the NuScale Power Plant to avoid those phenomena covered by the requirements of 10 CFR Part 50, Appendix K, and has used updated decay heat models. The applicant further stated that the model conservatively calculates the consequences of postulated LOCAs from a spectrum of pipe break sizes and locations.

The staff reviewed TR-0516-49422-P, Revision 3, subject to the L&Cs in the staff's SER (ML25007A192 (nonproprietary), ML24312A004 (proprietary)), and confirmed that the EM conservatively calculates the consequences of postulated LOCAs.

Further, the staff reviewed the results of the LOCA analyses, discussed in Section 15.6.5 of this SER, to confirm that no post-CHF phenomena will occur for breaks within the LOCA design basis spectrum to invalidate the applicant's stated justification for the exemption request (i.e., certain Appendix K requirements are not modeled because they are precluded by the design of the NPM). Further, Table 15.0-4 in FSAR Section 15.0.0.2.2 specifies that the acceptance criteria for LOCAs for the NuScale design is CHF and collapsed liquid level (CLL). Since the FSAR acceptance criteria for LOCAs preclude any post-CHF phenomena, the use of an EM that does not model these aspects required by Appendix K will not result in an incomplete or non-conservative evaluation of the consequences of LOCAs. Further, the staff considered the updated decay heat models used in evaluating ECCS performance (1973 ANS-5.1 Standard plus 20 percent uncertainty) to be an acceptable alternative to the Appendix K requirement (1971 ANS-5.1 Standard plus 20 percent uncertainty) as discussed in the LOCA TR methodology. Section 15.6.5 of this SER contains the staff's evaluation of the calculational results of the consequences of LOCAs. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption poses no undue risk to public health and safety.

Consistent with Common Defense and Security

The proposed exemption does not affect the design, function, or operation of any structures or plant equipment that is necessary to maintain a secure plant status. In addition, the proposed exemption has no impact on plant security or safeguards procedures. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that this exemption will not impact the common defense and security.

Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of 10 CFR Part 50, Appendix K is to ensure that the LOCA EM conservatively calculates the consequences of postulated LOCAs. As described above, Table 15.0-4 in FSAR Section 15.0.0.2.2 specifies that the acceptance criteria for LOCAs for the NuScale design are CHF and CLL. Since the FSAR acceptance criteria for LOCAs preclude any post-CHF phenomena, the use of an EM that does not model these aspects identified in the exemption request and required by Appendix K will not result in an incomplete or nonconservative evaluation of the consequences of LOCAs. Therefore, the underlying purpose of the rule is met by the NuScale LOCA EM without inclusion of these model features.

The staff finds that the NuScale design meets the underlying purpose of these regulations because the LOCA EM can conservatively calculate the consequences of LOCAs for the phenomena present in the NuScale design and it is consistent with the acceptance criteria specified in Table 15.0-4 in FSAR Section 15.0.0.2.2.

The applicant stated in SDAA Part 7 that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(vi), related to any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption. However, if the staff finds that other special circumstances are present in accordance with another provision of 10 CFR 50.12(a)(2), a staff finding on whether special circumstances are present in accordance with 10 CFR 50.12(a)(2)(vi) is not necessary for the exemption to be granted. Because the staff finds that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), it makes no finding regarding the presence of special circumstances described in 10 CFR 50.12(a)(2)(vi).

Conclusion

For the reasons given above, as set forth in 10 CFR 50.12(a), the staff concludes that the proposed exemption requested in SDAA Part 7, Section 10, regarding requirements stated in 10 CFR Part 50, Appendix K, paragraphs I.A.4, I.A.5, I.B, I.C.1.b, and I.C.5.a is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. Also, the special circumstances in 10 CFR 50.12(a)(2)(ii) are present, in that the application of these portions of Appendix K in the particular circumstances is not necessary to achieve the underlying purpose of these rules. Therefore, the staff concludes that an exemption to the requirements of 10 CFR Part 50, Appendix K, paragraphs I.A.4, I.A.5, I.B, I.C.1.b, and I.C.5.a is justified and is approved.

15.0.2.4.2 Non-Loss-of-Coolant Accident Methodology

The staff reviewed TR-0516-49416-P, Revision 3, supplemented by docketed markups to the TR to be incorporated into the final TR revision, subject to the L&Cs in the staff's SER ((ML25007A192 (nonproprietary), ML24312A004 (proprietary))). The non-LOCA EM uses the code modeling bases developed for the LOCA methodology (TR-0516-49422-P, Revision 3) to apply to non-LOCA events.

The staff reviewed TR-0516-49422-P, Revision 3, subject to the L&Cs in the staff's SER (ML25007A192 (nonproprietary), ML24312A004 (proprietary)) and confirmed that the EM adequately applies to, and is capable of calculating, the consequences of non-LOCA events or boundary conditions used to determine the consequences of non-LOCA events. As described in SER Section 15.0.2.4.1 for LOCA, after the short-term LOCA and non-LOCA events transition to the XPC period, the methodology outlined in TR-124587-P, Revision 0, is used to assess the long-term event progression and response.

With respect specifically to L&C #4 regarding DHRS and SG heat transfer uncertainty, the staff reviewed information supplied by the applicant (ML25042A064) and determined that the treatment of uncertainty is justified for this application of the method based on margins to the non-LOCA figures of merit presented in the Chapter 15 analyses as well as the audited results of sensitivity studies on DHRS and SG heat transfer, such as those referenced by the applicant (ML25014A157).

With respect to L&C #8 requiring an assessment of the potential impact on the event-specific bias directions from changes to analytical limits and actuation delays for actuation of the reactor trip system and engineered safety features, the staff reviewed the specific analytical limits and actuation delays, as well as the applicant's safety analyses discussed in this SER chapter, and determined the event-specific bias directions applied were appropriate for the NPM-20 design described in the SDAA FSAR.

15.0.2.4.3 Flow Stability

The staff documents its review and approval of the stability methodology in the safety evaluation of TR-0516-49417-P-A, Revision 1. For application to the NPM-20 design, the staff confirmed that the L&Cs of the staff's prior approval are consistent with application to the NPM-20 design. Therefore, staff finds the PIM computer code, as described in TR-0516-49417-P-A, Revision 1, acceptable for performing stability analysis of the NuScale NPM-20 power module.

15.0.2.4.4 Subchannel Analysis

The staff documents itsreview and approval of the subchannel methodologies in the safety evaluations of TR-0915-17564-A and TR-108601-P-A, Revision 4 (ML19067A256 and ML24106A160). The staff documents its review and approval of the CHF correlations in the safety evaluations of TR-0116-21012-P-A, Revision 1, TR-107522-P-A, Revision 1 and TR-0516-49422-P, Revision 3 (ML18360A631, ML23118A377 and ML25007A192).

15.0.2.4.5 Radiological Consequence Analysis Methodology for the Design-Basis Accident

The computer codes used by the applicant to calculate DBA doses are described in the TR-0915-17565-P-A, Revision 4. The staff's evaluation and approval of TR-0915-17565-P-A, Revision 4, with L&Cs, is documented in its SER dated February 26, 2020 (ML20057G132).

15.0.2.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.0.2.

15.0.2.6 Conclusion

Based on the reviews discussed above, the staff determined that the NuScale analysis methods and computer codes used in non-LOCA safety analyses, LOCA evaluations, and extended passive cooling evaluations are consistent with the requirements in 10 CFR 52.137 and 10 CFR Part 50, Appendix K.

15.0.3 Radiological Consequences of Design-Basis Accidents

15.0.3.1 Introduction

This section of the SER describes the staff's review of the information provided in FSAR Chapter 15, that describes the evaluation of the design basis source terms (DBSTs) that are assessed for events projected to result in radiological consequences.

15.0.3.2 Summary of Application

In FSAR Chapter 15, the applicant performed radiological consequence assessments for reactor DBAs and the core damage event (CDE), using hypothetical site parameter atmospheric relative concentration (dispersion) values (χ/Q values) for accidents. Because all other aspects of the design are fixed, these χ/Q values help determine the required minimum distances to the exclusion area boundary (EAB) and the low population zone (LPZ) for a given site to provide reasonable assurance that the radiological consequences will be within the dose criteria given in the regulation, as identified below.

Although the term "design basis accident" is not explicitly defined in the regulations, the SRP defines DBAs as unanticipated conditions of operation (i.e., not expected to occur during the life of the nuclear power unit) that are used to set design criteria and limits for the design and sizing of safety-related systems and components. These are based upon the set of DBEs (as defined by NuScale), giving credit for fission product mitigation by safety-related SSCs. NuScale has provided DBSTs and radiological consequence analyses for DBAs and events. The events and surrogate source terms analyzed for radiological consequences include the following:

- failure of small lines carrying primary coolant outside containment (FSAR Sections 15.0.3.7.1 and 15.6.2)
- SGTF (FSAR Sections 15.0.3.7.2 and 15.6.3)
- main steamline break (MSLB) outside containment (FSAR Sections 15.0.3.7.3 and 15.1.5)
- rod ejection accident (REA) (FSAR Sections 15.0.3.7.4 and 15.4.8)
- fuel handling accident (FHA) (FSAR Sections 15.0.3.7.5 and 15.7.4)
- iodine spike design basis source term (DBST) (FSAR Section 15.0.3.7.6)
- CDE (FSAR Section 15.10)

The applicant provided information on the radiological consequences analysis methodology, assumptions, and results for the potential doses at the EAB, at the LPZ outer boundary, and in the MCR. The applicant also provided information on the radiological habitability in the NuScale design technical support center (TSC) to show compliance with the onsite emergency response facility regulatory requirements.

In FSAR Chapter 15, the applicant concluded that the NuScale design will provide reasonable assurance that the radiological consequences resulting from any of the above analyzed events will fall within the offsite dose criterion of 0.25 Sievert (Sv) (25 rem) total effective dose equivalent (TEDE), as given in 10 CFR 52.137(a)(2), and the control room operator dose criterion of 0.05 Sv (5 rem), as given in 10 GDC 19, as incorporated by reference in 10 CFR 52.137(a)(3). The applicant reached this conclusion by performing the radiological consequences analyses as follows:

- using reactor accident source terms based on NRC-approved TR-0915-17565-P-A, Revision 4.
- modeling removal of aerosols within the containment by natural phenomena using the methodology in the NRC-approved TR-0915-17565-P-A, Revision 4.

- crediting control of the pH of the water in the containment to prevent iodine evolution using the methodology in the NRC-approved TR-0915-17565-P-A, Revision 4.

- using a set of hypothetical atmospheric dispersion factor (χ/Q) values.

The χ/Q values are the relative atmospheric concentrations of radiological releases at the receptor point in terms of the rate of radioactivity release. In lieu of site-specific meteorological data, the applicant provided a reference set of site parameter short-term (accident) χ/Q values for the NuScale design using meteorological data that is expected to envelope offsite dispersion conditions at most potential plant site locations in the United States. The applicant gave the NuScale FSAR Accident χ/Q values used in the design basis dose analyses for the EAB, LPZ, MCR, and TSC receptors in NuScale FSAR Table 2.0-1, "Site Parameters."

FSAR Table 15.0-10 summarizes the offsite and MCR dose results from the design basis source terms and CDE radiological consequence evaluations and compares these results to the applicable dose acceptance criteria.

Technical and Topical Reports: The NuScale evaluation of the radiological consequences of the DBSTs and CDE relies upon the methodology in the NRC-approved NuScale TR-0915-17565-P-A, Revision 4.

15.0.3.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 52.137(a)(2)(iv), as it relates to the evaluation and analysis of the offsite radiological consequences of PAs with fission product release.

- 10 CFR 52.137(a)(2)(iv), as it relates to ability of plant systems to mitigate the radiological consequences of plant accidents.

- GDC 19, as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.

- 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," paragraph IV.E.8, as it relates to adequate provisions for an onsite TSC from which effective direction can be given and effective control can be exercised during an emergency.

The NRC staff notes that NuScale has proposed a PDC in place of GDC 19. The PDC proposed by NuScale is functionally identical to the GDC with respect to radiological control room habitability requirements.

The following document also provides additional criteria, or guidance in support of the acceptance criteria to meet the above requirements.

- RG 1.183, "Performance-Based Containment Leak-Test Program," as it provides guidance on acceptable methods to perform DBA radiological consequence analyses for LWRs.

15.0.3.4 Technical Evaluation

The applicant's analyses of the radiological consequences of the DBSTs and CDE reference the accident source term methodology described in TR-0915-17565, Revision 4. The staff's evaluation and approval of TR-0915-17565, Revision 4, with L&Cs is documented in the approved version of the TR.

The staff evaluated the applicant's calculated radiological consequences against the dose criteria, given in 10 CFR 52.137(a)(2)(iv), of 0.25 Sv (25 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release, and 0.25 Sv (25 rem) TEDE at the outer boundary of the LPZ for the duration of exposure to the radioactive release cloud.

The applicant incorporated accident-specific offsite dose acceptance criteria, which are derived from the regulatory criteria as either well within (0.065 Sv (6.5 rem)) or a small fraction of (0.025 Sv (2.5 rem)) the regulatory criteria based roughly on the likelihood of the accident.

The staff used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences from the DBSTs and CDE in the MCR of the NuScale design, pursuant to GDC 19. The staff evaluated the radiological habitability analysis for the NuScale design TSC against the onsite emergency response facility regulatory requirements in 10 CFR Part 50, Appendix E, paragraph IV.E.8 and 10 CFR 52.137(a)(8).

The radiological consequence analyses performed by the applicant evaluate the bounding radiological consequences DBEs and CDE described in FSAR Chapter 15, as applicable to the NuScale US460. The FSAR Chapter 15 DBSTs are analyzed for a single nuclear power module. FSAR Chapter 19 discusses the suitability of shared components and the design measures taken to ensure these components do not introduce multi-module risks.

The staff reviewed the radiological consequence analyses that were performed by the applicant using the hypothetical site parameter accident release χ/Q values given in FSAR Table 2.0-1. The staff's review found that the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria stated above for each of the analyses. To evaluate the applicant's analyses, the staff performed independent confirmatory radiological calculations as deemed necessary using the site parameter χ/Q values provided by the applicant and the RADionuclide Transport, Removal And Dose (RADTRAD) computer code, run within the Symbolic Nuclear Analysis Package (SNAP) suite of integrated applications for engineering analysis, developed for the NRC. Information on the SNAP/RADTRAD code is available from the NRC's Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) at https://ramp.nrcgateway.gov/content/snapradtrad-overview. The following sections describe the staff's findings.

The applicant followed the accident analysis guidance in FSAR Section 15.0.3, through implementation of the NuScale accident source term methodology TR. The NuScale radiological consequences analyses credit natural phenomena and safety-related SSCs to mitigate the radiological consequences of the DBSTs. The NuScale radiological consequence analyses of the DBSTs also take credit for selected non-safety-related fission product mitigation systems to ensure radiological habitability of the MCR. If the assumption resulted in a more limiting radiological consequence, non-safety-related SSCs were assumed operational. FSAR Section 15.0.0.6.6 describes the treatment of non-safety-related systems in the evaluation of DBEs. In its radiological consequence analyses, the applicant evaluated the DBEs considering a single, active failure that maximizes the radiological consequences. No credit was taken for operator actions in the initial 72 hours of the accident. Due to NuScale's passive safety feature design, loss of offsite power (LOOP) does not affect the analyses with respect to estimation of the radioactive release. Scenarios with and without loss of normal AC power were evaluated with respect to the MCR and TSC ventilation systems capabilities to protect occupants during DBEs and the CDE.

The staff used guidance in NuScale DSRS Section 15.0.3, "Design Basis Accident Radiological Consequence Analysis for NuScale SMR Design," and RG 1.183, as applicable, in its review of the NuScale radiological consequence analyses of the DBSTs. Although RG 1.183 was written to apply to currently operating LWRs, its guidance on radiological acceptance criteria, formulation of the source term, and DBA modeling is useful in the review of the NuScale design, which is an LWR design. The applicant's radiological consequence analyses of the DBSTs and CDE use the guidance in RG 1.183 as far as applicable, as described in the referenced accident source term methodology TR. The applicant stated that deviations from the RG 1.183 guidance are due to differences between the NuScale design and large LWRs. The staff's review of the methodology used to evaluate the radiological consequences of the DBSTs is dependent on the acceptability of the referenced accident source term methodology TR. This SER discusses the staff's evaluation of the applicant's implementation of the accident source term methodology in the FSAR Chapter 15, radiological consequence analyses, including the bases for analysis input values and comparison of results to the regulatory requirements listed above and the NuScale DSRS 15.0.3 accident-specific dose acceptance criteria.

The staff applied a graded review approach to evaluate those areas that have an impact on safety in more detail. The staff's review concentrated on assumptions and analysis inputs related to post-accident operation of SSCs that based on uncertainty may increase the projected dose to reduce the margin to the applicable dose criteria. For example, although the staff did assess the applicant's analysis of the dose to MCR operators for all potential pathways as discussed below, the staff did not do a detailed evaluation of the applicant's calculations of direct dose in the MCR due to streaming from a power module CNV during the CDE. The staff made this decision to place less emphasis on review of the details of the applicant's direct dose analysis for this pathway because of the NuScale plant configuration which includes a large pool of water and several concrete walls between the CNV and the MCR area to provide radiation shielding. Conversely, the staff placed more emphasis on understanding the dose analysis inputs and assumptions that model the MCR, including operation of the non-safety related MCR ventilation systems that provide post-accident MCR habitability to comply with applicable requirements. Uncertainty in ventilation isolation and filtration capability with respect to exclusion and removal of airborne radioactive materials can have a large effect on the projected dose to the MCR occupants. The discussion of the staff's review below will generally be consistent with the emphasis placed on the topic of review.

15.0.3.4.1 Selection of Design Basis Accidents

The applicant evaluated the radiological consequences of the FSAR Chapter 15 DBEs that would result in radiological releases to the atmosphere and potentially could result in offsite doses or dose to MCR operators. The events include postulated deterministic accidents similar to those accidents other than the LOCA for evaluation of radiological consequences, described in SRP Chapter 15, "Transient and Accident Analysis," and RG 1.183, as applicable to a PWR.

The applicant also evaluated a postulated CDE in which significant core damage occurs with release to the environment through design leakage from the intact containment in order to give a CDST that is consistent with the source term description in 10 CFR 52.137(a)(2)(iv). NuScale DSRS Section 15.0.3 identifies the same accidents as being applicable to the NuScale design, with the exception that the NuScale FSAR evaluates the CDE and iodine spike DBST instead of the LOCA.

15.0.3.4.2 Site Characteristic Short Term Atmospheric Dispersion Factors

Because no specific site is associated with the NuScale design, the applicant defined the offsite boundaries only in terms of hypothetical atmospheric relative concentration (χ/Q) values at fixed EAB and LPZ distances as site parameters. The applicant assumed that the EAB and LPZ outer boundary are both located at the same analytical distance of 369 feet, which is the closest distance to the EAB/LPZ boundary from any potential release point. The applicant also provided hypothetical site parameter χ/Q values for each pairing of accident release point and receptor for the MCR and TSC, both for the ventilation system intake and the assumed MCR envelope inleakage location. FSAR Table 2.0-1 lists the site parameter accident release χ/Q values used in the radiological consequence analyses for the NuScale design. Section 2.3.4 of this SER discusses the staff's review of the hypothetical atmospheric dispersion factors.

An applicant that references the NuScale US460 design will provide short-term (less than or equal to 30 days) site-specific atmospheric dispersion factors for potential radiological consequence analyses based on the location of the EAB and the LPZ outer boundary using onsite meteorological data. If the applicant's site characteristic atmospheric dispersion factors exceed the NuScale site parameter values used in this evaluation (i.e., poorer dispersion characteristics), an applicant may have to consider compensatory measures, such as increasing the size of the site or providing additional ESF systems to reduce radiological releases to meet the relevant dose criteria.

15.0.3.4.3 Radiological Consequences

The NuScale US460 is an integral PWR design, with up to six nuclear power modules included in the plant. Although the NuScale US460 is an LWR, the plant design includes passive features to mitigate accidents, unlike currently operating domestic power reactors. In SECY-16-0012, "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors," dated February 7, 2016 (ML15309A319), the staff described the history of the potential policy and licensing issues for small modular reactors (SMRs) and non-LWRs with respect to the determination of accident source terms and the resulting dose calculation and siting evaluations. The staff has found it acceptable to permit use of mechanistic source terms to account for design-specific accident scenarios and accident progression in developing radiological source terms to meet the regulatory requirement in 10 CFR 52.137(a)(2), provided that the accident analyses include a DBA resulting in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products. TR-0915-17565-P-A, Revision 4, includes a methodology to determine a surrogate accident source term to meet the analysis requirements in 10 CFR 52.137(a)(2) in lieu of following prior NRC guidance regarding LOCA radiological consequence analyses for PWRs.

The NRC issued RG 1.183 in July 2000 to provide guidance to licensees of operating power reactors on acceptable applications of alternative source terms pursuant to 10 CFR 50.67, “Accident source term.” This RG provides guidance based on insights from NUREG-1465 “Accident Source Terms for Light-Water Nuclear Power Plants,” issued February 1995, and significant attributes of other alternative source terms that the staff may find acceptable for operating LWRs. It also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted alternative source term for operating power reactors. Although 10 CFR 50.67 is not applicable to new reactor reviews, the guidance in RG 1.183 may be applicable to LWR designs. In SRP Section 15.0.3, “Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors,” the staff’s review procedures direct the use of RG 1.183 regulatory positions, as applicable, to the plant design under review. The applicant followed the relevant guidance in RG 1.183 for PWRs, as applicable.

Core Radionuclide Inventory

The NuScale DBST and CDE radiological consequence analyses are based on 102 percent of rated core thermal power. The 2 percent power uncertainty assumption is consistent with the guidance in RG 1.183 on accounting for power uncertainty in the core fission product inventory. Accordingly, the applicant calculated the core fission product isotopic inventory at 102 percent of the core rated thermal power (RTP) of 250 megawatts thermal (MWt), which is 255 MWt.

The applicant used the Standardized Computer Analyses for Licensing Evaluation (SCALE) 6.1 code to develop the reactor core fission product inventory. The ORIGEN-S isotope generation and depletion computer code was used to calculate the core isotopic inventory shown on FSAR Table 11.1-1, “Maximum Core Isotopic Inventory.” The SCALE code suite is a comprehensive modeling and simulation suite for nuclear safety analysis and design developed and maintained by Oak Ridge National Laboratory (ORNL) under contract with the NRC, U.S. Department of Energy, and the National Nuclear Security Administration to perform reactor physics, criticality safety, radiation shielding, and spent fuel characterization for nuclear facilities and transportation/storage package designs. More information on the SCALE code suite is available at https://www.ornl.gov/scale. ORIGEN-S and ORIGEN-ARP are widely used in the nuclear industry to calculate fission product production and depletion. RG 1.183 includes guidance on use of an appropriate code to calculate the core fission product inventory and cites versions of the ORIGEN code (ORIGEN 2 and ORIGEN-ARP) as being appropriate. ORIGEN-S is the most up-to-date version of the ORIGEN code, while ORIGEN 2 is no longer supported by ORNL. ORIGEN-ARP is a fast version of the code sequence to perform point-irradiation calculations with the ORIGEN-S code using problem-dependent cross sections for fuel characterization.

Coolant Activity Concentrations

For DBEs other than the CDE and the FHA, the source of radioactive materials available for release are the primary and secondary coolant. FSAR Section 11.1, “Source Terms,” discusses the estimation of the radionuclide activity concentrations in the primary and secondary coolant. For use in the radiological consequence analyses of the DBSTs, the applicant adjusted the primary coolant radionuclide activity concentrations to the NuScale GTS 3.4.8 specific activity limits of 5.8E-02 microcuries per gram (µCi/gm) Dose Equivalent (DE) I-131 and 16 µCi/gm DE Xe-133 for the equilibrium initial condition. For the pre-incident iodine spiking scenario, the applicant assumes that the primary coolant iodine activity concentration is at the GTS 3.4.8 maximum specific activity limit of 3.5 µCi/gm DE I-131.

The applicant's analyses do not model the radionuclide activity concentration in the secondary coolant, which may be present by leakage of primary coolant through the SG tubes to the secondary coolant. TR-0915-17565-P-A, Revision 4, discusses a sensitivity study for the SGTF and MSLB accidents which assumed that the initial condition secondary coolant radionuclide activity concentrations were equivalent to the primary coolant radionuclide activity concentrations. The sensitivity study demonstrated that the dose results are not sensitive to the secondary coolant activity concentration initial conditions.

Reactor Building Pool Boiling Radiological Consequences

FSAR Section 15.0.3.6.3, "Reactor Building Pool Boiling Radiological Consequences," states that without available power for the active cooling systems, the addition of makeup water, or operator action, the sensible and decay heat from the NPMs and spent fuel would heat the ultimate heat sink (UHS) pool water and could eventually cause the water to boil. FSAR Table 9.2.5-2 indicates that it takes 2.3 days for the reactor building (RB) pool to reach boiling after a loss of normal AC power event. The applicant states that the estimated dose associated with this condition would be less than 1 milli Sievert (mSv) (0.1 rem) TEDE at each of the onsite and offsite receptor locations.

The applicant provided clarifying information in response to an RAI on the NuScale US600 design (ML17213B267) stating that the postulated reactor building pool boiling event is not a DBA, and the results are not added to the dose results of each of the DBSTs and CDE. The applicant further stated in the RAI response that projected dose from a reactor pool boiling event is discussed in Design Control Document, Tier 2, Chapter 15 to provide additional information concerning the potential event. Because the reactor building pool boiling event is not a DBA, and the doses are provided only for additional information, the staff did not evaluate the applicant's calculation of the reactor building pool boiling radiological consequence and makes no specific finding on this analysis. This clarifying information provided for the US600 also applies to the SDAA for the US460. The staff's conclusions regarding the reactor building pool boiling event for the US600 design are applicable to the US460 design as well.

Control Room and Technical Support Center Radiological Habitability

FSAR Section 6.4, "Control Room Habitability", describes the control room habitability system (CRHS), which provides breathable air to the control room for 72 hours without reliance on electrical power if the normal control room heating, ventilation, and air conditioning (HVAC) system (CRVS) is unavailable. After 72 hours, the CRVS, if restored, provides filtered HVAC service to the control building (CRB) for the remainder of an event recovery period. The CRVS is described in FSAR Section 9.4.1.

FSAR Sections 15.0.3.6.1, "Main Control Room Design," and 15.0.3.6.2, "Technical Support Center Design," describe the modeling of the MCR and TSC in the radiological consequence analyses. The dose pathways considered include intake and inleakage to the MCR envelope, direct shine, skyshine and shine from filters. Occupancy factors are taken from RG 1.183. The MCR dose analyses are performed for two cases, based on the MCR emergency mode. The staff evaluated the applicant's analyses of the radiological consequences discussed in the FSAR for each of the following MCR emergency mode cases.

The first case models the operation of the nonsafety-related CRVS inlet filters removing 99 percent of iodine providing continuous filtered airflow to the control room envelope for the duration of the event. This emergency mode assumes an uninterrupted power supply. Because the TSC is included in the MCR envelope for this mode, the TSC radiological habitability evaluation uses this model. The MCR and TSC dose analyses include the delay time for the isolation and initiation of the system. The MCR dose analyses assume 5 cubic feet per minute (cfm) unfiltered inleakage for ingress and egress through the airlock, plus 10 cfm unfiltered inleakage through the MCR envelope. The TSC dose analyses assume 10 cfm unfiltered inleakage for ingress and egress through the TSC door, plus 10 cfm unfiltered inleakage through the ventilation envelope.

The second case models the operation of the CRHS to provide clean bottled air for 72 hours. The CRHS is a nonsafety-related system that provides emergency breathing air to the MCR envelope and maintains a positive control room pressure for habitability and control of radioactivity when conditions prohibit the CRVS from fulfilling these functions. After 72 hours, the CRVS provides continuous filtered airflow to the MCR envelope for the remainder of the event duration. The CRHS is assumed to be initiated automatically to isolate and pressurize the MCR, based upon immediate loss of power. During the operation of the CRHS, the MCR dose analyses assume 5 cfm unfiltered inleakage for ingress and egress through the airlock, plus 10 cfm unfiltered inleakage through the MCR envelope. After 72 hours, the assumptions on filtered intake and unfiltered inleakage during operation of the CRHS are the same as in the first case. The TSC is not supplied by the CRHS, therefore TSC doses are not calculated for this case. The TSC function can be moved if the TSC is uninhabitable.

The following sections discuss the staff's evaluation of each DBA and the finding that the NuScale US460 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of the postulated DBAs will meet the applicable regulatory requirements for the MCR and the TSC.

Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

FSAR Section 15.0.3.7.1, "Failure of Small Lines Carrying Primary Coolant Outside Containment," describes the radiological consequence analysis for the failure of small lines carrying primary coolant outside containment and FSAR Section 15.6.2, "Failure of Small Lines Carrying Primary Coolant Outside Containment," contains additional information on the event. The event is a postulated break in the CVCS injection and discharge lines, which does not result in fuel damage. Therefore, the radiological consequence analysis assumes release of the primary coolant through the break, with a coincident iodine spike that raises the equilibrium appearance rate by a factor of 500 for 8 hours. The containment isolates, and the assumed single failure is a stuck open CIV in the line that has the break. The analysis does not credit holdup or retention in the reactor building.

The staff determined that the analysis method described in FSAR Section 15.0.3.7.1 is consistent with the methodology in TR-0915-17565-P-A, Revision 4, and uses NuScale design specific inputs to determine the radiological consequences of the failure of small lines carrying primary coolant outside containment. The dose results are less than the applicable accident specific dose criteria. Based upon its review of the FSAR implementation of the TR-0915-

17565-P-A, Revision 4, methodology, the staff finds that the applicant's analysis of the failure of small lines carrying primary coolant outside containment is acceptable.

The staff finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the offsite radiological consequences of a postulated failure of small lines carrying primary coolant outside containment will be a small fraction of dose criteria set forth in 10 CFR 52.137(a)(2). The staff also finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the postulated failure of small lines carrying primary coolant outside containment. Therefore, the staff finds there is reasonable assurance that the radiological consequences in the MCR and TSC following a postulated failure of small lines carrying primary coolant outside containment meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

Radiological Consequences of Steam Generator Tube Failure

FSAR Section 15.0.3.7.2, "Steam Generator Tube Failure," describes the SGTF radiological consequence analysis. The analysis in FSAR Section 15.6.3 shows that the SGTF does not result in fuel damage. Therefore, the radiological consequence analyses assume two primary coolant iodine spiking cases: a coincident iodine spike that raises the equilibrium appearance rate by a factor of 335 for 8 hours, and a pre-incident iodine spike, where the primary coolant iodine activity concentration is elevated to the GTS maximum. The SGTF radiological analysis assumes a single failure of an MSIV on the faulted SG to maximize the radiological release. Primary coolant flows into the secondary coolant in the faulted SG through the failed SG tube at a rate and duration specified in FSAR, Section 15.6.3. Primary coolant leaks into the secondary side of the intact SG at the maximum leak rate allowed by design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure. The analysis does not credit holdup or retention in the reactor building.

The staff determined that the analysis method described in FSAR Section 15.0.3.7.2 is consistent with the methodology in TR-0915-17565-P-A, Revision 4, and uses NuScale design specific inputs to determine the radiological consequences of a postulated SGTF. The dose results are less than the applicable accident-specific dose criteria for both iodine spiking cases. Therefore, based upon its review of the FSAR implementation of the TR-0915-17565-P-A, Revision 4, the staff finds that the applicant's analysis of the SGTF is acceptable.

The staff finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the offsite radiological consequences of a postulated SGTF with coincident iodine spiking will not exceed a small fraction of the dose criteria set forth in 10 CFR 52.137(a)(2) and that the offsite radiological consequences of a postulated SGTF with pre-incident iodine spiking will not exceed the dose criteria set forth in 10 CFR 52.137(a)(2).

The staff also finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the postulated SGTF, for both iodine spiking cases. Therefore, the staff finds there is reasonable assurance that the radiological consequences in the MCR and TSC following a postulated SGTF meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

Radiological Consequences of a Main Steamline Break Outside Containment

FSAR Section 15.0.3.7.3, "Main Steamline Break Outside Containment Accident," describes the radiological consequence analysis for the MSLB outside containment. The analysis in FSAR Section 15.1.5, "Steam Piping Failures Inside and Outside of Containment," shows that the MSLB outside containment does not result in fuel damage. Therefore, the radiological consequence analyses assume two primary coolant iodine spiking cases; a coincident iodine spike that raises the equilibrium appearance rate by a factor of 500 for 8 hours, and a pre-incident iodine spike, where the primary coolant iodine activity concentration is elevated to the GTS maximum. Primary coolant leaks into the secondary side of the intact steam generators at the maximum leak rate allowed by design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure. The analysis does not credit holdup or retention in the reactor building.

The staff determined that the analysis method described in FSAR Section 15.0.3.7.3 is consistent with the methodology in TR-0915-17565-P-A, Revision 4, and uses NuScale design specific inputs to determine the radiological consequences of a postulated MSLB outside containment. The dose results are less than the applicable accident-specific dose criteria for both iodine spiking cases. Therefore, based upon its review of the FSAR implementation of the TR-0915-17565-P-A, Revision 4, methodology, the staff finds that the applicant's analysis of the MSLB outside containment is acceptable.

The staff finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the offsite radiological consequences of a postulated MSLB outside containment with coincident iodine spiking will not exceed a small fraction of the dose criteria set forth in 10 CFR 52.137(a)(2) and that the offsite radiological consequences of a postulated MSLB outside containment with pre-incident iodine spiking will not exceed the dose criteria set forth in 10 CFR 52.137(a)(2).

The staff also finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the postulated MSLB outside containment, for both iodine spiking cases. Therefore, the staff finds there is reasonable assurance that the radiological consequences in the MCR and TSC following a postulated MSLB outside containment meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

Radiological Consequences of a Rod Ejection Accident

The analyses described in FSAR, Section 15.4.8, show that there is no fuel damage predicted for the limiting REA. Therefore, the radiological consequences of the REA are bounded by other events, as stated in FSAR Section 15.0.3.7.4.

Section 3.2.1, "Rod Ejection Accident," of TR-0915-17565-P-A, Revision 4, provides a source term and dose analysis method that assumes failure of one full assembly. The TR methodology goes on to state that in accordance with the guidance in RG 1.183, Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," no radiological consequence analysis is required if no fuel damage is indicated in the REA analysis. In that case, the radiological consequences of the REA are bounded by the LOCA, MSLB and steam generator tube rupture (SGTR). For the NuScale design, the CDE is the equivalent analysis to the RG 1.183 LOCA, while the SGTF is the equivalent to the SGTR.

The staff finds that the applicant's analysis of the REA is consistent with the methodology in TR-0915-17565-P-A, Revision 4, and RG 1.183. Therefore, because the REA does not result in fuel damage, the staff finds that the radiological consequences of the REA are bounded by other events analyzed in the NuScale FSAR.

Radiological Consequences of a Fuel Handling Accident

FSAR Section 15.0.3.7.5, "Fuel Handling Accident," describes the FHA radiological consequence analysis with supporting information in FSAR Section 15.7.4. The FHA is postulated for dropping a fuel assembly onto the spent fuel racks, in the reactor vessel, in a spent fuel cask during loading, or on the weir wall between the reactor pool and the spent fuel pool. The analysis assumes damage to all rods in the dropped fuel assembly, with release to the reactor pool and subsequent release to the environment. In accordance with the methodology in TR-0915-17565-P-A, Revision 4, the analysis assumed a conservative pool scrubbing effective iodine decontamination factor of 200 based on 23 feet of water above the fuel, although the pool is much deeper. The analysis assumed no holdup or retention in the reactor building. No single failure affects this accident.

The staff determined that the analysis method described in FSAR Section 15.0.3.7.5 is consistent with the methodology in TR-0915-17565-P-A, Revision 4, and uses NuScale design specific inputs to determine the radiological consequences of the FHA. The dose results are less than the applicable accident-specific dose criteria. Therefore, based upon its review of the FSAR implementation of the TR-0915-17565-P-A, Revision 4, methodology the staff finds that the applicant's analysis of the FHA is acceptable.

The staff finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the offsite radiological consequences of a postulated FHA are well within the dose criteria set forth in 10 CFR 52.137(a)(2).

The staff also finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the postulated FHA. Therefore, the staff finds there is reasonable assurance that the radiological consequences in the MCR and TSC following a postulated FHA meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

Radiological Consequences of an Iodine Spike Design Basis Source Term

As described in FSAR Section 15.0.3.7.6, "Radiological Analysis of an Iodine Spike Design-Basis Source Term," the iodine spike DBST is a postulated surrogate source term to bound a spectrum of events that result in release of primary coolant to the intact containment, without damage to the fuel. The NuScale US460 has DBEs that result in primary coolant being released to the intact containment.

Neither DSRS Section 15.0.3 nor RG 1.183 describes a DBA with such characteristics, including the offsite dose criteria that should apply. However, the referenced TR-0915-17565-P-A, Revision 4, methodology states that the assumptions for the coolant activity and modeling of iodine spiking are consistent with the assumptions for modeling a PWR MSLB accident from RG 1.183, Appendix E. The iodine spike DBST is analyzed for two iodine spiking cases, with the same assumptions as in the SGTF and the MSLB.

The primary coolant is instantaneously released to the CNV through a non-specific release point and homogenously mixed as a vapor into the CNV free volume. In accordance with RG 1,183, the activity is then assumed to leak into the environment at the design basis containment leak rate for the first 24 hours, then at 50 percent of the design leak rate for the remainder of the release. The release from containment is assumed to end at 30 hours, when the reactor is depressurized.

The staff determined that the analysis method described in FSAR Section 15.0.3.7.6 is consistent with the methodology in TR-0915-17565-P-A, Revision 4, and uses NuScale design specific inputs to determine the radiological consequences of the iodine spike DBST. The dose results are less than the applicable accident-specific dose criteria. Therefore, based upon its review of the FSAR implementation of the TR-0915-17565-P-A, Revision 4, methodology, the staff finds that the applicant's analysis of the iodine spike DBST is acceptable. The staff finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the offsite radiological consequences of a postulated iodine spike DBST will not exceed the dose criteria set forth in 10 CFR 52.137(a)(2).

The staff also finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the postulated iodine spike DBST. Therefore, the staff finds there is reasonable assurance that the radiological consequences in the MCR and TSC following a postulated iodine spike DBST meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

Radiological Consequences of the Core Damage Event

The applicant postulates a CDE to show compliance with the regulatory requirements cited above in Section 15.0.3.3 of this SER. This postulated event closely follows accident characteristics described in 10 CFR 52.137(a)(2) and the related footnote and is not a single specific accident scenario. The applicant applied the methodology in TR-0915-17565-P-A, Revision 4, to develop a surrogate CDST to evaluate the CDE for the NuScale design, based on severe accident scenarios. The CDST associated with the CDE is composed of key radiological release and transport parameters, derived from a range of accident scenarios that result in significant damage to the reactor core with subsequent release of appreciable quantities of fission products into the containment. The CDST is used as input to radiological consequence assessments. The SER for TR-0915-17565-P-A, Revision 4 contains additional information on the staff's evaluation of the CDE radiological consequence analysis methodology, including determination of the CDST..

FSAR Section 15.10, "Core Damage Event," describes the applicant's CDE radiological consequence analysis, including the parameters for the CDST. The key parameters that describe the CDST release from the core to the containment are the magnitude of the release, expressed as a fraction of the core radionuclide inventory, and the timing of the release. The core radionuclide inventory is in accordance with the methodology in TR-0915-17565-P-A, Revision 4. The CDST release timing and magnitude are derived from a spectrum of severe accident scenarios based on the NuScale PRA. The CDST is developed based on five accident scenarios derived from intact-containment internal events in the Level 1 PRA. Each of the five surrogate scenarios involves various failures of the ECCS, with the DHRS assumed available. The applicant used the MELCOR severe accident code to estimate the release timing and core release fractions for each of the five surrogate accident scenarios, which were then used to determine the CDST release timing and core release fractions using the methodology in the TR. The parameters describing the CDST release from the core to the containment are given in FSAR, Table 15.10-1, “Core Damage Source Term Release Timing,” and Table 15.10-2, “Core Inventory Release Fractions.”

One consideration that bears upon the assumptions on potential re-evolution of iodine removed from the containment atmosphere is the post-accident temperature-dependent pH (pH_T) of the liquid inside containment. TR-0915-17565-P-A, Revision 4, Section 4.4, “Post-Accident pH_T,” describes the methodology used for evaluating post-accident pH_T in coolant water following a significant core damage event that results in the CDST and to present a method to show compliance with the radiological consequence evaluation factors in 10 CFR 52.137(a)(2)(iv)(A) and 10 CFR 52.137(a)(2)(iv)(B) for offsite doses; GDC 19 for MCR radiological habitability; and the requirements related to the TSC in 10 CFR 52.137(a)(8) and paragraph IV.E.8 of Appendix E to 10 CFR Part 50. Section 4.4 of the TR also includes a summary of acids and bases that are expected to enter the coolant and influence the pH_T during a significant CDE.

The staff also reviewed FSAR Section 15.10.1.8, “pHT and Iodine Re-Evolution,” which describes the calculation and results for the post-accident pH_T for input to the TR methodology to develop the CDST. The results of the applicant’s analyses show that the post-accident pH_T inside the containment is between 6.0 and 7.0, and iodine re-evolution is not explicitly included in the CDST, in accordance with the methodology in TR-0915-17565-P-A, Revision 4. Therefore, the staff finds the assumption that iodine re-evolution in the containment is negligible for the CDST to be acceptable.

The applicant’s analysis of aerosol natural deposition in containment during the CDE used the methodology in TR-0915-17565-P-A, Revision 4. The applicant’s analysis method calculates time-dependent airborne aerosol mass and removal rates within the NuScale containment by modeling the effects of aerosol deposition by natural processes. The staff found that the applicant’s analysis used design-specific inputs consistent with the event-specific information in FSAR Section 15.10 to calculate the time-dependent aerosol removal rates listed in FSAR Table 15.10-3, “Containment Aerosol Removal Rates.” Because the applicant used the methodology in TR-0915-17565-P-A, Revision 4, along with design-specific inputs, to determine the aerosol removal rates used in the CDE radiological consequence analysis, the staff finds the modeling of aerosol natural deposition in containment to be acceptable.

The staff’s review of the applicant’s analysis of the radiological consequences of the CDE determined that the analyses were performed using the methodology described in TR- TR-0915-17565-P-A, Revision 4, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. The doses calculated by the applicant are within the dose criteria given in NuScale DSRS Section 15.0.3 for the LOCA, which include events such as the CDE. Therefore, based upon its review of the FSAR implementation of the TR-0915-

17565-P-A, Revision 4 methodology, the staff finds that the applicant's analysis of the CDE is acceptable. The staff finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the offsite radiological consequences of a postulated CDE will not exceed the dose criteria set forth in 10 CFR 52.137(a)(2).

The staff also finds that the NuScale design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the estimated doses in the MCR and TSC are less than 0.05 Sv (5 rem) TEDE for the postulated CDE. The staff finds there is reasonable assurance that the radiological consequences in the MCR and TSC following a postulated CDE meet the dose criterion given in GDC 19 and the TSC habitability regulatory requirements, respectively.

15.0.3.5 Conclusion

The staff has reviewed the radiological consequences analyses of the DBSTs and CDE described in FSAR Chapter 15, for the NuScale design. Based on the evaluation discussed above, the staff finds that the NuScale design meets 10 CFR 52.137(a)(2)(iv) dose criteria and the accident-specific offsite dose acceptance criteria given in NuScale DSRS Section 15.0.3.

Based on the evaluation discussed above, the staff finds reasonable assurance that the MCR habitability systems, as described in FSAR Sections 6.4 and 9.4.1, can mitigate the dose in the MCR following DBEs including the CDE to meet the dose criterion specified in GDC 19.

In addition, the staff finds reasonable assurance that the MCR habitability systems can mitigate the dose in the TSC following DBEs and the CDE to be within 0.05 Sv (5 rem) TEDE, to meet the TSC habitability requirements in paragraph IV.E.8 of Appendix E to 10 CFR Part 50, and 10 CFR 52.137(a)(8).

15.0.4 Safe, Stabilized Condition

15.0.4.1 Introduction

Safety analyses of DBEs are performed from event initiation until a safe, stabilized condition is reached. A safe, stabilized condition is reached when the initiating event is mitigated, the acceptance criteria are met, and system parameters (for example, inventory levels, temperatures, and pressures) are trending in the favorable direction.

15.0.4.2 Summary of Application

In FSAR Section 15.0.4, "Safe Stabilized Condition," the applicant stated, "for events that involve a reactor trip, system parameters continue changing slowly as decay and residual heat are removed and the RCS continues to cool down." Post-event actions with respect to boron distribution, from both LOCA and non-LOCA events, are important to ensure that a core dilution event is prevented and coolable geometry is maintained. FSAR Section 15.0.4 states the following:

> Boron concentrations and distributions are shown to be acceptable during XPC for decay and residual heat removal in Section 15.0.5. Boron concentration and distribution in the NPM are important considerations when exiting passive cooling and must be

accounted for to ensure subcriticality and coolable geometry are maintained during post-event recovery actions.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: There are no technical specifications associated with this section.

Technical Reports: There are no technical reports associated with this section.

15.0.4.3 Technical Evaluation

The applicant stated, and the staff agrees, that no operator action is required to reach or maintain a safe, stabilized condition within the first 72 hours based on meeting each of the specific event Chapter 15 acceptance criteria. The staff notes that the post-event recovery actions referred to in FSAR Section 15.0.4 are outside the scope of the standard design review but are important to capture in the development of operating procedures with respect to ensuring coolable geometry (remaining below the boron solubility limit for precipitation) and subcriticality are maintained during post-event recovery actions. The applicant included COL Item 13.5-3 in FSAR Section 13.5.2, “Operating and Maintenance Procedures,” for the development of operating procedures at a future licensing stage.

15.0.4.4 Conclusion

Section 15.0.5 of this SER contains the staff’s conclusions as to whether the events have been appropriately evaluated until a safe, stabilized condition is reached.

15.0.5 Extended Passive Cooling for Decay and Residual Heat Removal

15.0.5.1 Introduction

Two systems perform the safety-related function of decay and residual heat removal from the NPM-20 following a DBE. The DHRS, described in FSAR Section 5.4.3, provides decay and residual heat removal, while RCS inventory is retained inside the RPV. The ECCS, described in FSAR Section 6.3, is the other system and provides for decay heat and residual heat removal when RCS inventory is redistributed between the RPV and CNV after the RVVs and RRVs are opened. NuScale describes four long-term heat removal scenarios:

(1) DHRS with ECCS actuation blocked by operators after verifying acceptable reactivity conditions

(2) DHRS with the RVVs and RRVs opening 24 hours after a loss of normal AC power

(3) DHRS with the RVVs and RRVs opening 8 hours after a reactor trip (if ECCS actuation not blocked by operator action)

(4) ECCS actuation following a LOCA, inadvertent ECCS operation or loss of normal AC and EDAS power

In the first scenario, the DHRS provides long-term decay heat removal for the entire event without ECCS actuation, FSAR Section 5.4.3 discusses the ability of the DHRS to remove residual and decay heat. Section 5.4.3 of this SER gives the staff's evaluation of the DHRS. The design of the upper riser holes in the NPM-20 is intended to allow sufficient mass flow rate between the downcomer and riser to support continued decay heat removal through RPV primary flow without a maldistribution of boron occurring between the downcomer and riser prior to ECCS actuation due to condensation forming on the steam generator tubes during extended DHRS operation with RPV water level below the top of the riser.

Scenarios 2 through 4 utilize the ECCS, which either begins with ECCS valves opening, or following successful initiation of the DHRS. The ECCS valves open, either at the initiation of the DBE (such as a LOCA) or as a result of a transition from DHRS heat removal during a non-LOCA event. The opening of the ECCS valves establishes recirculation flow between the RPV and containment.

For events in which ECCS is actuated, the ECCS supplemental boron (ESB) function provides a passive source of boron that compensates for the positive reactivity added by the cooldown. For events that result in a reactor trip and DHRS actuation, the ECCS actuates 8 hours after a reactor trip unless the operators bypass the ECCS timer. Operators bypass the ECCS timer if reactivity conditions indicate the additional negative reactivity provided by the ESB function is not needed to maintain subcriticality under cold conditions. The ESB function is described in Section 6.3 of the FSAR.

The XPC configuration is reached through both LOCA and non-LOCA initiating events. The non-LOCA initiating events generally involve the DHRS cooldown and eventual ECCS initiation, much later than for LOCA cases. Consequently, compliance with the long-term core cooling requirements of 10 CFR 50.46(b)(4) and (b)(5) must be demonstrated for LOCA or non-LOCA events. The XPC analyses are intended to demonstrate that decay heat removal and reactor cooldown via the reactor pool ultimate heat sink are effective such that the reactor module(s) will remain in a safe, stable condition for up to 72 hours following a postulated event. The XPC analysis also demonstrates that boron transport and redistribution is adequate to maintain the reactor subcritical and remains below the solubility limit for precipitation.

TR-124587, Revision 0, (hereafter referred to as the XPC TR) provides the methodology for demonstrating that XPC using DHRS and ECCS is adequate to remove decay and residual heat after the RPV and containment pressures have approximately equalized in pressure, and stable flow through the RRVs is established. The XPC TR also describes the methodology for calculating boron transport and redistribution to maintain subcriticality and remain below the solubility limit for precipitation. FSAR Sections 6.3 and 15.6.5 discuss the short-term ability of the ECCS to remove residual and decay heat. Sections 6.3 and 15.6.5 of this SER provide the staff's evaluation of this capability.

The staff's evaluation in this SER section is for XPC events, which use either the DHRS for long term heat removal or that transition to ECCS. This SER section evaluates the application of the XPC TR methodology and results from applying the methodology to the design that show that the LTC acceptance criteria are met

15.0.5.2 Summary of Application

XPC and residual and decay heat removal can be accomplished by either (1) the DHRS, (2) a combination of the DHRS and transition to ECCS, or (3) early actuation of the ECCS with minimal assistance from DHRS. The XPC TR provides the EM and methodology for LTC and boron transport for calculations for the LTC configuration. The XPC TR methodology addresses all DBEs that evolve to a LTC configuration using either DHRS or a combination of both DHRS and ECCS. The results of applying the methodology to the design are contained in the FSAR. The NPM-20 is intended to remain in a safe, stable condition out to 72 hours without operator action. As stated in the XPC TR, the figures of merit are that the collapsed liquid level (CLL) in the RPV remains above the top of the active fuel, the core remains subcritical assuming the highest worth control rod stuck out, and boron concentrations in the core region remain below the boron solubility limit for precipitation. The applicant maintained that CHF is limiting during the short-term LOCA and non-LOCA transient phase, and maintaining the CLL above the top of active fuel is sufficient to demonstrate that the MCHFR limit is met in the long term.

The applicant also described its evaluation results relative to the design of the riser holes and the potential for downcomer boron dilution. For non-LOCA events, following a reactor trip and actuation of the DHRS, the subsequent cooldown may cause RPV water level to drop below the top of the riser (referred to as riser uncovery). Once the level drops below the top of the riser via DHRS cooling (or potential leakage), the upper riser holes are designed to allow sufficient mass flow rate between the downcomer and riser to support continued decay heat removal through RPV primary flow. Continued decreases in the RPV level through decay heat removal (or leakage) would lead to an ECCS actuation on the low-low level signal (some other ECCS actuation may also occur, i.e. ECCS actuation timer) before the upper riser holes are uncovered. Additionally, the upper riser holes are intended to reduce downcomer dilution of the boron content due to condensation forming on the steam generator tubes during extended DHRS operation with RPV water level below the top of the riser, such that an unacceptable positive reactivity insertion upon opening of the ECCS valves is precluded. The upper riser holes are intended to allow a sufficient mass flow rate of borated liquid between the downcomer and riser to maintain downcomer boron concentrations above the critical boron concentration prior to ECCS operation. After ECCS actuation, the design of the lower riser holes is intended to allow recirculation of liquid from the riser into the downcomer to maintain a mixed boron concentration in the RPV.

To determine the riser hole design adequacy, the applicant evaluated the boron concentration using an acceptance criterion that it remains above the critical concentration out to 72 hours. Based on the 72 hour trends, the applicant also evaluated out to 7 days. Section 19.3.4 and Section 6.3 of this SER contains the evaluation of boron concentration out to 7 days.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The following GTS are applicable to this area of review:

- The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.
- LCO 3.5.4, “Emergency Core Cooling System Supplemental Boron (ESB)”.

Technical Reports: There are no technical reports associated with this area of review.

15.0.5.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

- GDC 27, as it relates to controlling the rate of reactivity changes to ensure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 34, states that a system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded.

- GDC 35, states that a system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal water reaction is limited to negligible amounts.

- For coolable geometry, 10 CFR 50.46(b)(4) requires that, calculated changes in core geometry shall be such that the core remains amenable to cooling.

- For LTC, 10 CFR 50.46(b)(5) requires that, after any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period required by the long-lived radioactivity remaining in the core.

The staff notes that the applicant provided the PDC for GDC 34 and 35, which are functionally identical to GDC 34 and 35.

15.0.5.4 Technical Evaluation

15.0.5.4.1 Evaluation Model

The applicant used the XPC TR analysis methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to analyze the reactor thermal-hydraulic response to the event. In addition, the applicant used the XPC methodology to evaluate the boron transport and distributions analyses for subcriticality and boron precipitation. Section 15.0.2 of this SER describes the staff's evaluation of the EM.

15.0.5.4.2 Input Parameters, Initial Conditions, and Assumptions

FSAR Table 15.0-22, "Input Parameters for Emergency Core Cooling System Extended Passive Cooling Analysis - Limiting Boron Cases," and FSAR Table 15.0-23, "Input Parameters for Emergency Core Cooling System Extended Passive Cooling Analysis - Limiting Level and Temperature Cases," present the initial conditions and key input parameters evaluated for LTC and boron transport calculations. The specific values chosen are a function of which figure of merit is being evaluated. Initial conditions and biasing are determined based on the XPC TR methodology to provide conservative results with respect to the figure of merit being evaluated except **{{**

}}. Biasing the non-condensable gas **{{ }}** is acceptable for the FSAR analysis because the amount of non-condensable gas is insignificant based on: (1) FSAR Table 6.2-1, "Containment Design and Operating Parameters," shows that less than 1 psia of non-condensable gas is in the CNV, (2) FSAR Figure 5.2-2, " Containment Leakage Detection Acceptability, " shows lower CNV allowable operating pressures at lower pool temperatures and (3) the design has a safety related passive autocatalytic recombiner as described in FSAR section 6.2.5, "Combustible Gas Control in the Containment Vessel" and related subsections.

The staff finds these input parameters, initial conditions and assumptions acceptable as they are either supported by the GTS (e.g., minimum pool temperature), MPS setpoints or were chosen by NuScale to be conservative assumptions for the respective figure of merit consistent with the XPC TR methodology.

15.0.5.4.3 Evaluation of Analysis Results

<u>DHRS Heat Removal Cooldown Capability: Riser Hole Evaluation</u>

FSAR Section 15.0.5.2.1, "Riser Flow Characteristics," describes the flow characteristics of the riser flow holes during DHRS operation while the riser remains covered and when the riser is uncovered. During DHRS operation, heat removal is achieved by flow going over the top of the riser. As DHRS continues to remove heat, the RCS pressure and temperature conditions trend lower as decay heat is removed. During continued decay heat removal, the level in the RPV reduces potentially below the top of the riser. Once the level drops below the top of the riser via DHRS cooling, the upper riser holes are designed to allow sufficient mass flow rate between the downcomer and riser to support continued decay heat removal through RPV primary flow.

NuScale provides the design values for the upper and lower riser holes in FSAR Section 15.0.5.3, "Emergency Core Cooling System Extended Passive Cooling"**.** Given (1) the design values and the conservatisms for the riser holes, and (2) behavior and trends observed in XPC TR Section 5.5, which the staff confirmed is consistent with the SDAA design via audit, the staff finds that DHRS with the riser holes are adequate to provide the ability to continue to remove decay heat during the DHRS heat removal phase for LTC evaluations.

<u>DHRS Extended Passive Cooling and Boron Transport</u>

LTC and residual and decay heat removal can be accomplished by the DHRS. FSAR Section 15.0.5.2.2, "Decay Heat Removal System Extended Passive Cooling," states that during DHRS cooling, RCS inventory is retained inside the RPV and therefore maintaining collapsed liquid level above the top of the core is not challenged. The staff confirmed that liquid level above the core is not challenged during DHRS cooling because the RCS inventory is retained in the RPV and sufficient inventory loss to the CNV would cause the ECCS to be actuated if the level during DHRS cooling gets below the low-low level ECCS setpoint. In which case, the ECCS would maintain collapsed liquid level above the top of the core.

FSAR Section 15.0.5.2.2 states that the non-LOCA transient analyses presented in Chapter 15 demonstrate that adequate decay heat removal is established and that the pool liquid level remains above the top of the DHRS condensers for at least 72 hours. Additionally, the FSAR states that after stable DHRS cooling exceeds decay heat and downward trends in primary pressure and temperature are established during the short-term transient response, RCS pressure and temperatures remain acceptable, and cooling continues for 72 hours. The NRC staff confirmed that the trends from the limiting short-term analyses show that cooling is maintained, and heat removal exceeds decay heat at the end of the short-term events. The staff finds that the DHRS is capable of providing adequate heat removal for LTC given that cooling exceeds the decay heat at the end of short-term events and that the DHRS tubes remain covered.

The FSAR states that for DHRS extended passive cooling boron transport subcriticality analyses (1) the upper riser flow paths provide borated water from the riser to mix with the condensate in the downcomer and primary coolant flow through the riser flow paths exceeds condensation generation; (2) the downcomer concentration remains higher than the critical concentration for beginning of cycle (BOC) and middle of cycle (MOC) over a range of cooling conditions, including accounting for the highest-worth CRA stuck out of the core and leakage; and (3) subcriticality is demonstrated for conditions where boron distribution effects are less important (end of cycle (EOC) conditions) and temperature effects are maximized. The staff confirmed in audit that the BOC and MOC DHRS extended passive cooling boron transport subcriticality analyses show downcomer and core boron concentrations remain above the critical boron concentration. DHRS extended passive cooling boron transport subcriticality analyses with EOC conditions are evaluated as part of the ECCS extended passive cooling boron transport subcriticality analyses and the results are presented in FSAR Figures 15.0-7, “Boron Transport Analysis Concentrations - Reactor Component Cooling Water Line Break Case (Slow Bias Supplemental Boron),” and 15.0-9, “Boron Transport Analysis Concentrations - Reactor Component Cooling Water Line Break Case (Fast Bias Supplemental Boron)”. The ECCS extended passive cooling boron transport subcriticality analyses include DHRS cooling then a transition to ECCS cooling after 8 hours. FSAR Figures 15.0-7 and 15.0-9 show that the boron concentration in the downcomer and core remains above the critical boron concentration for the DHRS cooling portion of the events. The analyses are performed at EOC conditions, which correspond to minimal initial boron concentrations in the RPV and include conservative inputs for DHRS cooldown and would therefore bound BOC and MOC conditions. Therefore, the staff finds that during the DHRS cooling period the core remains subcritical during BOC, MOC and EOC conditions with consideration for a stuck rod.

The FSAR states that ECCS cooling conditions bound extended DHRS cooling with respect to boron precipitation evaluations because ECCS cooling results in lower coolant temperatures with a higher boron concentration. The staff confirmed that ECCS conditions bound extended DHRS cooling conditions because ECCS cooling results in lower coolant temperatures and adds boron to the coolant due to the function of the ESB system.

<u>Boron Transport During ECCS Cooldown</u>

LTC and residual and decay heat removal can be accomplished by early ECCS initiation or with DHRS that transitions to ECCS cooling. FSAR Section 15.0.5.3.1, “Boron Transport during Emergency Core Cooling System Cooldown,” discusses ECCS extended passive cooling boron transport subcriticality analyses during an emergency core cooling system cooldown. The FSAR states that an ECCS cooldown occurs following a LOCA or inadvertent ECCS operation events or can occur following a non-LOCA event if required to maintain subcriticality, where a timer actuates ECCS. The setpoint of the timer is 8 hours after a reactor trip if not bypassed by the operator. The operator may bypass the timer if reactivity conditions indicate the additional negative reactivity provided by the ESB is not needed to maintain subcriticality under cold conditions. Additionally, the FSAR states that there are two factors that determine limiting cases for maintaining adequate boron concentration in the core to preclude criticality which are (1) ensuring subcriticality for an injection line break inside containment that results in a continual flow path for boron to transport from the core and riser into containment and (2) ensuring subcriticality for events that transition from DHRS cooling to ECCS cooling on the 8-hour timer. The staff audited the underlying calculations that support ECCS extended passive cooling boron transport subcriticality analyses and confirmed that the reactor component cooling water system break with 8 hours of DHRS cooling and a transition to ECCS is the liming event (LOCA and non-LOCA) with respect to boron transport for demonstrating subcriticality.

The FSAR evaluates a break in piping connected to the reactor component cooling water system and considers this the limiting event because any release of liquid into the CNV can potentially dissolve the ESB boron oxide into a liquid that collects in the bottom of the CNV before ECCS actuation and the additional unborated liquid entering the system from the RCCW provides an additional dilution volume. The eight-hour timer actuates ECCS in the analysis. The mass of reactor component cooling water added to the CNV is given as 6840 kg, which sets the limit for the maximum allowable water available in the RCCW that can flood the containment. The staff confirmed that this is the limiting event with respect to boron transport and maintaining subcriticality in the core. The ESB provides the means of adding boron to the system used in the analysis. The staff notes that the minimum amount of boron oxide is 25 kg in each ESB dissolver. This represents the maximum amount boron available to be added to the system and the boron quantity is required to be within the limits specified in the core operating limits report per technical specification limiting condition for operation 3.5.4 and associated bases.

The FSAR states that ESB parameters used in the boron transport analysis are conservative and includes conservative biasing of mixing tube losses, length, elevation. Additionally, FSAR Table 6.3-4, “Classification of Structures, Systems, and Components,” provides the ESB minimum collector rail capacity and FSAR Table 15.0-22, “Input Parameters for Emergency Core Cooling System Extended Passive Cooling Analysis - Limiting Boron Cases” provides the CNV wall condensation collection area assumed in the boron transport analyses. The XPC TR methodology describes how these assumptions and inputs are used including the considerations for containment wall heat transfer and condensate collection. The CNV related transport terms assumed in the analyses are dependent on the as-built plant design response with respect to condensate flow rates, mixing tube flow rates, boron dilution and transport. FSAR Section 14.2.3.3, “Testing of First-of-a-Kind Design Features,” and Table 14.2-40, “Test #40 Emergency Core Cooling System,” describes testing for the as-built design for the ESB and containment response to confirm the boron dissolution, mixing and transport. The NRC staff reviewed the ESB and containment parameters and finds them to be adequate given the design values in combination with the testing and XPC TR for use in the boron transport analyses.

The FSAR states that the lower riser hole flow rates calculated by NRELAP5 are assessed as described in the XPC TR methodology. The staff reviewed the results of the assessment presented in the FSAR and audited supporting calculations. The results from the assessment were verified in accordance with 10 CFR 50 Appendix B. The NRC staff finds that the lower riser hole flow rates used are adequate for boron transport subcriticality and precipitation analyses because the assessment is consistent with the methodology and shows adequate margin to the calculated flow rates.

The staff finds that the analyses were performed adequately because the analyses were performed consistent with XPC TR methodology, and inputs and assumptions were used that lead to conservative results for the figure of merit. This includes accounting for pre-transient operational histories including reduced-power operations and rapid power ascensions and their impacts on short term xenon changes and the potential for low decay heat by establishing operational limits for minimum boron concentration and the corresponding power ascension rate, as shown in Figure 15.0-16 and technical specification controls specified in LCO 3.5.4 and corresponding bases.

FSAR Table 15.0-18, “Margin to Critical Boron Concentration for Emergency Core Cooling System Cooldown Events,” gives the results for the limiting cases for the ECCS extended passive cooling boron transport criticality analysis. FSAR Figures 15.0-5, “Boron Transport Analysis Concentrations - Injection Line Break Case,” through Figure 15.0-10, “Boron Transport Analysis Masses - Reactor Component Cooling Water Line Break Case (Fast Bias Supplemental Boron),” show the boron concentrations and boron masses in the RPV and CNV for the limiting cases. The data in FSAR Table 15.0-18, “Margin to Critical Boron Concentration for Emergency Core Cooling System Cooldown Events,” show the safety margins in terms of equivalent soluble boron concentration for the reactor to reach criticality. The limiting case shows a minimum margin of 28 ppm to the critical boron concentration. The results of the analysis show that there is at least a 25 ppm margin that can account for **{{** **}}** to account for the consideration of the density difference between the borated and unborated liquid as required by the XPC TR. The staff finds that subcriticality is maintained because the boron concentration in the core during ECCS extended passive cooling boron transport subcriticality events remain above the critical boron concentration.

FSAR Section 15.0.5.3.1 describes the ECCS extended passive cooling boron transport precipitation analyses. The boron precipitation analysis is performed in a similar manner to the ECCS extended passive cooling boron transport subcriticality analysis but using assumptions that maximize the boron in the core. The FSAR states that the limiting event is an inadvertent opening of an RVV. The staff notes that the maximum amount of boron oxide is 30 kg in each ESB dissolver. This represents the maximum amount of boron available to be added to the system. The staff finds that the analyses were performed adequately because the analyses were performed consistent with the XPC TR methodology, and inputs and assumptions were used that lead to conservative results for the figure of merit.

FSAR Table 15.0-19, “Margin to Boron Precipitation Limit for Emergency Core Cooling System Cooldown Event,” gives the results for the ECCS extended passive cooling boron transport precipitation analysis. FSAR Figures 15.0-11, “Boron Precipitation Analysis Concentrations – Inadvertent Opening of a Reactor Vent Valve Case,” and Figure 15.0-12, “Boron Precipitation Analysis Masses – Inadvertent Opening of a Reactor Vent Valve Case,” show the boron concentrations and boron masses in the RPV and CNV for the limiting case. FSAR Table 15.0-19 shows the safety margins in terms of boron precipitation limits and a minimum margin of 6250 ppm to the precipitation limit for the limiting case and therefore the staff finds that boron precipitation is precluded during the event and coolable geometry is therefore maintained.

Long Term Minimum Collapsed Liquid Level

FSAR Section 15.0.5.3.2, "Minimum Collapsed Liquid Level during Emergency Core Cooling System Cooldown," states that a series of transient events are evaluated with minimum level biasing applied consistent with the XPC TR methodology. The biasing focused on maximizing the differential pressure across the ECCS valves and minimizing the initial RCS inventory. The staff notes that Section 3.9.6.4.6.1 of the SE for the XPC TR specifies that the Xc value used to calculate effects of compressibility through the RVVs is part of the QME-1 program to be consistent with the analyses. The applicant evaluated a wide range of events and conditions including LOCAs of varying sizes, inadvertent valve openings, breaks outside containment, leakage scenarios, and SGTF.

FSAR Section 15.0.5.3.2, states that the most limiting factor for collapsed liquid level is the mass retained in the RCS and CNV. Additionally, the FSAR states that breaks outside the CNV and the SGTF lose RCS fluid that is not available for ECCS cooling. The staff confirmed that events that cause an initial loss of inventory from the system lead to the most limiting results for the collapsed liquid level figure of merit. The FSAR states that the limiting loss of inventory event is a SGTF. The RPV inventory loss corresponds to the mass of water loss through the secondary side. The applicant stated that once the secondary system isolates, additional mass is transferred as the pressure equalizes between the primary and the failed secondary loop. Once the pressures equalize, some mass drains back from the failed secondary into the primary, but a portion of the mass transferred is retained in the secondary piping.

The staff confirmed that the SGTF is the limiting event with regards to the lowest collapsed liquid inventory. Breaks outside containment also resulted in low levels in comparison to events where inventory is not lost from the RCS. The staff reviewed the results and supporting calculations during audit for other cases analyzed and found them not limiting relative to the acceptance criteria in comparison to the SGTF case.

The staff reviewed the selected event specific inputs to conservatively evaluate the respective acceptance criteria. The staff finds that the analyses were performed adequately because the analyses were performed consistent with the XPC TR methodology, and inputs and assumptions were used that lead to conservative results for the figure of merit. Additionally, the staff performed confirmatory analysis for the SGTF case which indicated the SDA analysis was performed conservatively.

FSAR Table 15.0-20, "Limiting Collapsed Liquid Level above Top of Active Fuel during Emergency Core Cooling System Cooldown," gives the results for the limiting cases for the collapsed liquid level analysis. FSAR Figure 15.0-4, "Collapsed Liquid Level for Limiting Steam Generator Tube Failure Case," shows the collapsed liquid level for the limiting event. FSAR Table 15.0-20 shows the safety margins in terms of collapsed liquid level and the minimum collapsed liquid level for the limiting case is 1.8 ft above the top of the fuel.

The staff finds that the top of active fuel remains covered for the limiting SGTF event and the LTC requirement for 10 CFR 50.46(b)(5) is met.

Maximum Temperature During ECCS Cooldown

FSAR Section 15.0.5.3.3, "Maximum Temperature during Emergency Core Cooling System Cooldown" states that analyses were performed that evaluated extended ECCS cooling with maximum temperature biasing applied using the methodology described in the XPC TR. The FSAR states that results of this analysis demonstrate that during extended ECCS cooling, vessel integrity is maintained, and module pressures and temperatures decrease and are maintained at low values. The FSAR states that the limiting case is the inadvertent opening of an RVV with loss of AC and EDAS power with one RVV failing to open.

The staff reviewed the applicant selected event specific inputs to conservatively evaluate the respective acceptance criteria. The staff finds that the analyses were performed adequately because the analyses were performed consistent with the XPC TR methodology, and inputs and assumptions were used that lead to conservative results for the figure of merit, such as accounting for multimodule impacts, high pool temperature and low pool level.

FSAR Figure 15.0-15, “Moderator Temperature for Inadvertent Opening of a Reactor Vent Valve Event (Maximum Temperature Analysis)” gives the result for the limiting case for maximum temperature analysis. The staff finds that the NPM-20 heat removal capacity for the high temperature case is adequate as demonstrated by the temperature and pressures are adequately reduced over the long-term period. Additionally, the short-term MCHFR limit would not be exceeded given the ability for decay heat to be adequately removed.

15.0.5.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.0.5.

15.0.5.6 Conclusion

Based on analysis results reviewed, the staff determined that the requirements in 10 CFR 50.46 and 52.137(a); PDCs 34 and 35, GDCs 26, and 27; 10 CFR Part 50, Appendix K; have been met throughout the 72-hour period following a LOCA or non-LOCA event with DHRS or ECCS cooldown. Therefore, the staff has determined that the analysis of extended passive cooling resulting from non-LOCA and LOCA events contain margin to the acceptance criteria, including (1) collapsed liquid level in the reactor vessel remains above the top of the core, (2) subcriticality is maintained, (3) boron concentration remains below the solubility limit for precipitation, and (4) margins to MCHFR are maintained.

15.1 Increase in Heat Removal by the Secondary System

15.1.1 Decrease in Feedwater Temperature

15.1.1.1 Introduction

The staff reviewed FSAR Section 15.1.1, “Decrease in Feedwater Temperature,” to ensure that the event was analyzed appropriately and meets the acceptance criteria outlined in Section 15.1.1.3 of this SER. A decrease in feedwater temperature is an AOO that causes an increase in heat transfer from the primary to the secondary system. The negative MTC and the cooldown of the RCS cause an increase in core reactivity. Automatic control rod motion to maintain RCS temperature also adds positive reactivity. As a result, core power increases, and the MCHFR decreases. Continued overcooling leads to a reactor trip and subsequent actuation of the DHRS.

15.1.1.2 Summary of Application

A decrease in feedwater temperature may result from a failure in the FWS. The increase in heat transfer from the primary to the secondary system causes a reduction in RCS temperature, leading to positive reactivity insertion from the negative MTC feedback. The event is analyzed with and without the reactivity control system in the automatic mode to determine the more limiting scenario. If the reactivity control system is in the automatic mode, it inserts positive reactivity by pulling the regulating control rods from the core to maintain RCS temperature. The limiting MCHFR scenario is with automatic control of the regulating control bank disabled, which delays the reactor trip. The core power increase leads to a reactor trip signal on high core power for the limiting MCHFR case. The secondary system isolation (SSI) and DHRS actuation isolate feedwater, ending the cooldown, and transition the NPM to a stable condition.

The applicant analyzed this event using the NRELAP5 computer code to obtain the NPM time-dependent thermal-hydraulic response and VIPRE-01 to obtain the time-dependent MCHFR and the statistical subchannel analysis methodology developed in TR-108601-P-A, Revision 4, is used to calculate the uncertainties associated with the MCHFR using the VIPRE-01 computer code. The applicant stated that the chosen initial conditions result in a conservative calculation. The applicant stated that it did not credit operator actions and stated that no single failure produces a more limiting result for MCHFR.

The applicant concluded that the MCHFR remains above the 95/95 analysis limit for the NSP4 CHF correlation. The applicant further stated that the primary and secondary peak pressures for this event meet the respective acceptance criteria and are less limiting than the pressures described in FSAR Section 15.2, “Decrease in Heat Removal by the Secondary System.”

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.1.1.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS design with appropriate margin so that SAFDLs are not exceeded during normal operations, including AOOs.
- GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
- GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so that the pressure boundary is not breached during normal operations, including AOOs.
- GDC 20, as it relates to the reactor protection system being designed to automatically initiate appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs.

- GDC 26, as it relates to the control of reactivity changes so that SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

FSAR Sections 15.1.1 through 15.1.4 list the following acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections:

- The most limiting moderate-frequency initiating events that result in increased heat removal are identified.

- For the most limiting initiating events, it is verified that the plant responds to the transients in such a way that the criteria for fuel damage and system pressure are met.

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

- Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR based on acceptable correlations (see DSRS Section 4.4, "Thermal and Hydraulic Design").

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

- The guidance provided in RG 1.105, "Setpoints for Safety-Related Instrumentation," can be used to analyze the effect of instrument spans and setpoints on the plant response to the type of transient addressed in DSRS Sections 15.1.1 through 15.1.4, to meet the requirements of GDC 10, 13, 15, 20, and 26.

- The most limiting plant system's single failure, as defined in "Definitions and Explanations," in 10 CFR Part 50, Appendix A, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53, "Application of the Single-Failure Criterion to Safety Systems."

- The applicant's analysis of transients caused by excessive heat removal should be performed using an acceptable analytical model and approved methodologies and computer codes. The values of the parameters used in the analytical model should be suitably conservative.

15.1.1.4 Technical Evaluation

15.1.1.4.1 Evaluation Model

The applicant used the non-LOCA analysis methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to analyze the reactor thermal-hydraulic response to the event. In addition, the applicant performed a subchannel analysis using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.1.1.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the transient analysis model. The applicant modeled the decrease in feedwater temperature as a linear decrease from the nominal temperature of 121 degrees Celsius (C) (250 degrees Fahrenheit (F) to the minimum feedwater temperature of 26.7 °C (80 °F). The applicant conducted a sensitivity study to identify the limiting rate of temperature decrease by varying the length of time over which the temperature decreases and found that the limiting MCHFR occurs when the decrease occurs over 100 seconds, or a rate of about 0.94 °C (1.7 °F) per second. (ML24305A000). The staff finds this modeling approach acceptable because it considers an acceptable range of temperature decrease rates, and the applicant uses the limiting cooldown rate in conjunction with limiting initial conditions in the limiting event analysis.

The staff reviewed the initial parameter values and biases to ensure that the applicant selected conservative values for the analysis. The staff notes that the applicant assumed suitably conservative parameters that maximize the consequences of the event, including a 102 percent initial core power level, and maximum time delay to scram with the most reactive rod held out of the core. The use of the most negative (i.e., EOC) moderator reactivity feedback with the most positive Doppler temperature feedback is acceptable because it maximizes the core power response and results in a more limiting MCHFR. (ML24348A130 (nonproprietary), ML24348A132 (proprietary)).

The applicant credited several MPS signals for a decrease in feedwater temperature event. For the limiting event for MCHFR, the high core power signal trips the reactor. The high main steam pressure signal actuates SSI and DHRS. For the limiting event for primary and secondary pressures, the high pressurizer pressure signal trips the reactor.

The applicant based high power level and high power rate trips on an effective power level (specified in FSAR Table 15.0-7, "Analytical Limits and Time Delays") to account for the temperature decalibration effect on ex-core detectors because of cooler, denser water in the downcomer affecting ex-core detector signals. (ML24291A085) The staff audited the calculation for the impact on the accuracy of the ex-core detector caused by the downcomer temperature decrease (ML24211A089). Footnote 2 on FSAR Table 15.0-7 indicates that the effective core power is reduced by a factor of 7 percent when downcomer temperature decreases by less than 10 °F, and this factor is scaled upward from 7 percent by approximately 0.6 percent/°F for the amount that the downcomer temperature decreases beyond 10 °F. While this adjustment is larger than indicated by the calculation the staff audited, the NRC staff expects that this overestimation is conservative because it will cause RTS actuation on the high power level and high power rate signals to occur later and at a higher power level during overcooling transients (ML24353A333). The staff finds that the methodology used for calculating this impact is acceptable because the result is reasonable.

The staff notes that technical report TR-122844, Revision 0, "NuScale Instrument Setpoint Methodology Technical Report," issued December 2022 (ML23304A349 (nonproprietary), ML23304A350 (proprietary)), describes how the applicant's setpoints conform to RG 1.105. The staff's review of TR-122844, Revision 0, is documented in Chapter 7 of this SER.

The applicant did not credit operator action to mitigate the decrease in feedwater temperature event. Although the limiting case disabled the automatic rod control system, the applicant considered cases where the rod control system operates as designed to ensure the limiting case is properly identified. The feedwater controls are conservatively disabled to provide a constant feedwater flow rate rather than the normal response of reducing feedwater flow.

The applicant did not assume a loss of power in the limiting event analysis. The staff notes that loss of AC power would trip the feedwater pumps, terminating the cooldown event. The staff notes that the DC power system, EDAS, is relied on to remain functional during this event. Section 15.0.0.6.2 of this SER gives the detailed evaluation of EDAS treatment in the Chapter 15 safety analysis. The applicant also stated that no single failure causes a more limiting MCHFR. Although a single failure of an MSIV or FWIV may contribute to overcooling of the RCS, the staff notes that these valves close after a valid SSI signal. Because MCHFR occurs before SSI, failure of an MSIV or FWIV to close would not produce more limiting results for MCHFR.

The staff audited (ML24211A089) the applicant's sensitivity studies that investigated the most limiting initial conditions, single failures, and loss of power assumptions to confirm that they led to the most limiting results**.** The audited material supports the discussions in the FSAR, and the staff finds that the input parameters and initial conditions listed in the FSAR are suitably conservative and result in the most limiting conditions for MCHFR. As discussed in FSAR, Section 15.1, "Increase in Heat Removal by the Secondary System," the non-LOCA EM in TR-0516-49416-P, Revision 4, does not require performing extensive sensitivity studies to maximize RCS or SG pressure for this event category because cooldown events do not significantly challenge system pressures. In addition, the applicant stated that the pressure responses to cooldown events are less severe than those of AOOs in FSAR Section 15.2.The staff agrees that the cooldown AOOs analyzed in the FSAR are not limiting with respect to system pressures. Because of the bounding nature of the inherently pressurizing FSAR Section 15.2 events, the staff finds the applicant's treatment and calculation of maximum RCS and SG pressures for FSAR Section 15.1, events acceptable.

15.1.1.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.1.1, to ensure that they meet the DSRS acceptance criteria. The staff reviewed the sequence of events table for the decrease in feedwater temperature event and finds that it is consistent with the event description and assumptions about protective system actuation and delay times. In addition, the staff reviewed the figures showing the transient progression and finds that they support the applicant's event description and assertion that the acceptance criteria are met. Furthermore, the figures show that the NPM reaches a stable condition, indicating that the events do not lead to a more serious plant condition.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. For this event, the reported analytical results relative to the MCHFR and RCS pressure figures of merit are the results without the adjusted DHRS modeling, which the staff finds acceptable as MCHFR and RCS pressure are not significantly impacted by this modeling adjustment. The analytical results for secondary pressure figures of merit reported in FSAR Table 15.1-3, "Decrease in Feedwater Temperature (15.1.1) - Limiting Analysis Results,"are the NRELAP5 results obtained after adjusting the DHRS modeling.

FSAR Table 15.1-3presents the limiting analysis results for this event for MCHFR. The limiting MCHFR for this event, 1.71, remains above the 95/95 limit described in the DSRS. The applicant performed sensitivity studies consistent with TR-0516-49416-P, Revision 4, which the staff audited, and reported the maximum system pressures. The limiting sequence of events for the maximum system pressures results in an MPS trip on high pressurizer pressure with calculated maximum RCS and SG pressures of 2,121 pounds per square inch, absolute (psia) and 1454 psia, respectively, which are below 110 percent of the design values (2,420 psia). These values are tabulated in FSAR Table 15.1-3.

By demonstrating that the DSRS AOO acceptance criteria are met for the most limiting decrease in the feedwater temperature scenario, the applicant satisfied the requirements associated with GDC 10, 13, 15, 20, and 26 for this transient.

15.1.1.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.1.1.

15.1.1.6 Conclusion

The staff reviewed the decrease in feedwater temperature event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. The staff concludes that the applicant's analysis of this event is acceptable and meets the requirements of GDC 10, 13, 15, 20, and 26 with respect to this event because it satisfies the acceptance criteria in the DSRS.

15.1.2 Increase in Feedwater Flow

15.1.2.1 Introduction

The staff reviewed FSAR Section 15.1.2, "Increase in Feedwater Flow," to ensure that the event was analyzed appropriately and meets the acceptance criteria discussed in Section 15.1.2.3 of this SER. An increase in feedwater flow causes an increase in heat transfer from the primary to the secondary system. The negative MTC, the cooldown of the RCS, and the automatic control rod withdrawal (if the reactivity control system is in an automatic mode) to maintain RCS temperature cause an increase in core reactivity. Reactor power increases, while MCHFR decreases. Continued overcooling leads to a reactor trip and subsequent SSI and actuation of the DHRS.

15.1.2.2 Summary of Application

An increase in feedwater flow may result from a failure in the FWS. The increased heat transfer from the primary to the secondary system reduces RCS temperature, leading to positive reactivity insertion from the negative MTC and from the automatic control rod withdrawal attempting to maintain a programmed RCS temperature, if the reactivity control system is in an automatic mode. Core power increases and MCHFR decreases until the reactor trips on one of several MPS signals. Feedwater flow into the SG ends through either isolation or when SG pressure reaches the FW pump cutoff pressure, which ends the overcooling event. The high pressurizer (PZR) pressure or high main steam pressure MPS signal actuates DHRS, which transitions the NPM to a stable condition.

The applicant analyzed this event using NRELAP5 to obtain the NPM time-dependent thermal-hydraulic response and VIPRE-01 to find the time-dependent MCHFR. The applicant stated that the chosen initial conditions result in a conservative calculation. The overcooling event analyses account for delays in power-based reactor trips due to decreasing downcomer temperature. The applicant did not credit operator action and stated that no single failure produces a more limiting result for MCHFR. The applicant considered a single failure of an FWIV to investigate possible SG overfill scenarios.

The applicant concluded that the limiting increase in feedwater flow event meets the DSRS acceptance criteria. MCHFR remains above the 95/95 analysis limit for the NSP4 CHF correlation. The applicant further stated that the primary and secondary peak pressures for this event meet the respective acceptance criteria because they are less limiting than the pressures described in FSAR Section 15.2.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.1.2.3 Regulatory Basis

The regulatory basis described in SER Section 15.1.1.3 also applies to SDAA Section 15.1.2.

15.1.2.4 Technical Evaluation

15.1.2.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to analyze the thermal-hydraulic response to the event. In addition, the applicant performed a subchannel analysis using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. The statistical subchannel analysis methodology developed in TR-108601-P-A, Revision 4 is used to calculate the uncertainties associated with the MCHFR using the VIPRE-01 computer code. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.1.2.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the transient analysis model. To identify the limiting increase in feedwater flow scenario, the staff audited the applicant's calculations and confirmed that the applicant analyzed a spectrum of feedwater flow increases up to the maximum flow that can be provided by the number of feedwater pumps in service at a given power level, up to three feedwater pumps at full power**.** The applicant assumed that the feedwater pump speed increases over 1.0 second. (ML24305A000) The staff notes that the analyzed spectrum of flow increases bounds possible causes of an increase in feedwater flow and is therefore acceptable. In addition, varying the magnitude of flow increase is sufficient to adequately identify the limiting rate of increase. The applicant concluded that the limiting event for MCHFR results from an increase in FW pump speed corresponding to an approximately 60 percent increase in feedwater flow.

The staff reviewed the initial parameter values and biases to ensure that the applicant selected conservative values for the analysis. The staff notes that the applicant assumed suitably conservative parameters that maximize the consequences of the event, including a 102 percent initial core power level, maximum time delay to scram with the most reactive rod held out of the core, and the most limiting moderator reactivity feedback caused by the most negative MTC.

NRC staff audited the applicant's calculations and noted the applicant analyzed cases with both the most-negative and least-negative DTC. In some instances where the least-negative DTC was assumed, reactor trip was induced by the high power rate signal. NRC staff finds varying the DTC to identify the limiting MCHFR acceptable.

The applicant credited several MPS signals for an increase in feedwater flow event. The limiting event for MCHFR results in a reactor trip on the high core power signal, SSI on the low PZR pressure signal, and DHRS actuation on the high main steam pressure signal. (ML24305A000) The applicant generated high power level and high power rate trips based on an effective power level (specified in FSAR Table 15.0-7) t to account for the temperature decalibration effect as a result of cooler, denser water in the downcomer affecting ex-core detector signals. The staff audited the calculation for the impact on the accuracy of the ex-core detector caused by the downcomer temperature decrease (ML24211A089). Footnote 2 on FSAR Table 15.0-7 indicates that the effective core power is reduced by a factor of 7 percent when downcomer temperature decreases by less than 10 °F, and this factor is scaled upward from 7 percent by approximately 0.6 percent/°F for the amount that the downcomer temperature decreases beyond 10 °F. While this adjustment is larger than indicated by the calculation the staff audited, the NRC staff considers that this overestimation is conservative because it will cause RTS actuation on the high power level and high power rate signals to occur later and at a higher power level during overcooling transients. The staff finds that the methodology used for calculating this impact is acceptable because the result is reasonable. The limiting SG overfill case credits the reactor trip and secondary side isolation on the low main steam superheat signal.

TR-122844, Revision 0, describes how the applicant's setpoints conform to RG 1.105. The staff's review of TR-122844, Revision 0, is in Chapter 7 of this SER.

The applicant did not credit operator action to mitigate the increase in feedwater flow event.

The applicant assumed normal AC power is available for this event. A loss of AC power is not conservative because it would cause the feedwater pumps to stop and terminate the event progression. In addition, the staff notes that EDAS is relied on to remain functional during this event. Section 15.0.0.6.2 of this SER contains the detailed evaluation of EDAS treatment in the Chapter 15 safety analysis.

The applicant assumed that the rod control system operates as designed, which is conservative because it withdraws control rods when the RCS temperature drops, inserting positive reactivity. In addition, the applicant did not assume a single failure for the limiting MCHFR case. For the reasons stated in Section 15.1.1.4.2 of this SER, which also apply to this event, the staff finds these assumptions acceptable.

The staff audited the applicant's sensitivity studies (ML24211A089) that investigated the most limiting initial conditions and assumptions to confirm that they led to the most limiting results for MCHFR. The audited material supports the discussions about the limiting MCHFR case in the SDAA, and the staff finds that the input parameters, initial conditions, and assumptions listed in the SDAA are suitably conservative and result in the most limiting conditions for MCHFR.

The applicant considered a single failure of an FWIV to close to analyze the potential for SG overfill, which could degrade performance of the related DHRS train. The applicant provided initial conditions and biases for the DHRS performance case (SG overfill case) in SDAA Table 15.1-6, "Table 15.1-6: Increase in Feedwater Flow – Inputs," and FSAR Section 15.1.2.3.3 and assumptions in FSAR Section 15.1.2.3.2. The NRC staff reviewed these initial conditions and assumptions and the details of the calculations made available for audit, and finds that they maximize SG level and minimize margin to DHRS performance. These assumptions include biased-high decay heat, biased-high pool temperature, and **{{** **}}**. The applicant concluded **{{**

}}. The staff audited the underlying calculation and confirmed that the analyses therein support the docketed information and conclusions regarding SG overfill.

15.1.2.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.1.2, to ensure that they meet the DSRS acceptance criteria. The staff reviewed the sequence of events table in FSAR for the increase in feedwater flow event and finds that it is reasonably similar to the event description, assumptions, and results regarding protective system actuation and delay times in the supporting calculation documents that were audited. The applicant confirmed (ML24346A270) **{{**

}}. (ML24346A270) In addition, the staff reviewed the figures showing the transient progression for the limiting MCHFR case and finds that they support the applicant's event description and assertion that the acceptance criteria are met. Furthermore, the figures for the MCHFR case and audited figures for the limiting decay heat removal cases show that the NPM reaches a stable condition, indicating that the event does not lead to a more serious plant condition.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. For this event, the reported analytical results relative to the MCHFR and RCS pressure figures of merit are the results without the adjusted DHRS modeling, which the staff finds acceptable as MCHFR and RCS pressure are not significantly impacted by this modeling adjustment. The analytical results for secondary pressure figures of merit reported in Table 15.1-7 of the FSAR, as well as the time required for DHRS heat removal to exceed decay heat reported in FSAR Table 15.1-5, "Sequence of Events for Limiting Steam Generator Overfill Case (15.1.2 Increase in Feedwater Flow)," are the NRELAP5 results obtained after adjusting the DHRS modeling.

FSAR Table 15.1-7 presents the limiting analysis results for this event. The applicant calculated maximum RCS and SG pressures of 2,188 psia and 1471 psia, respectively. Although the applicant did not perform sensitivities to maximize system pressures, for the reasons discussed in Section 15.1.1.4.2 of this SER, which are also applicable to this analysis, the staff has reasonable assurance that the RCS and SG pressures will remain below 110 percent of the design values (2,420 psia) even if maximized for this event. The limiting MCHFR for this event, 1.69, remains above the 95/95 limit.

By demonstrating that the DSRS AOO acceptance criteria are met for the increase in feedwater flow events, the applicant satisfied the requirements associated with GDC 10, 13, 15, 20, and 26 for this transient.

15.1.2.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.1.2.

15.1.2.6 Conclusion

The staff reviewed the increase in feedwater flow event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. As documented above, the applicant's analysis shows that the AOO and DSRS acceptance criteria are met for this event. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of an increase in feedwater flow event meet the relevant requirements set forth in GDC 10, 13, 15, 20, and 26.

15.1.3 Increase in Steam Flow

15.1.3.1 Introduction

The staff reviewed SDAA Section 15.1.3, "Increase in Steam Flow," to ensure that the event was analyzed appropriately and meets the acceptance criteria discussed in Section 15.1.3.3 of this SER. This event is postulated to result from a spurious opening of the turbine bypass valve. The increased steam flow increases heat transfer from primary to secondary, cooling the RCS and causing a positive reactivity insertion. If the reactivity control system is in an automatic mode, positive reactivity is also added as control rods withdraw to maintain moderator temperature, resulting in increasing power. A reactor trip, SSI, and DHRS actuation mitigate the event. An increase in steam flow is classified as an AOO as discussed in FSAR Section 15.0.

15.1.3.2 Summary of Application

An increase in steam flow event may result from a spurious opening of the turbine bypass valve or the main steam safety valves (MSSVs). The increased steam flow increases heat transfer from the primary to the secondary system, decreasing moderator temperature and inserting positive reactivity. The control rods withdraw to maintain a programmed RCS temperature, inserting more positive reactivity. The reactivity addition increases core power and decreases MCHFR. A reactor trip, SSI, and DHRS mitigate the transient. No operator action is credited to mitigate the effects of an increase in steam flow event.

The applicant analyzed this event using NRELAP5 to obtain the NPM time-dependent thermal-hydraulic response and VIPRE-01 to find the time-dependent MCHFR. The applicant stated that the chosen initial conditions result in a conservative calculation. The applicant did not credit operator action and stated that no single failure produces a more limiting result for MCHFR.

The applicant concluded that the MCHFR remains above the 95/95 analysis limit for the NSP4 CHF correlation. The applicant further stated that the primary and secondary peak pressures for this event meet the respective acceptance criteria because they are less limiting than the peak pressures calculated in FSAR Section 15.2.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.1.3.3 Regulatory Basis

The regulatory basis described in SER Section 15.1.1.3 is also applicable to FSAR Section 15.1.3.

15.1.3.4 Technical Evaluation

15.1.3.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including the NRELAP5 code, to analyze the thermal-hydraulic response to the event. In addition, the applicant performed a subchannel analysis using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. The statistical subchannel analysis methodology developed in TR-108601-P-A, Revision 4, is used to calculate the uncertainties associated with the MCHFR using the VIPRE-01 computer code. Section 15.0.2 of this SER describes the staff’s evaluation of these codes.

15.1.3.4.2 Input Parameters, Initial Conditions, and Assumptions

To identify the limiting increase in steam flow scenario, the applicant analyzed a spectrum of steam flow increases up to a 100 percent opening of the turbine bypass valve. In addition, the staff finds that varying the magnitude of flow increase is sufficient to reliably identify the limiting rate of increase. The applicant found that the limiting event for MCHFR results from the TBV opening to 52 percent of its full-open area, resulting in an increase in steam flow.

The staff reviewed the initial parameter values and biases for the increase in steam flow event to ensure that the applicant selected conservative values for the analysis. The staff notes that the applicant assumed suitably conservative parameters that maximize the consequences of the event, including a 95 percent initial core power level, maximum time delay to scram with the most reactive rod held out of the core, and limiting EOC moderator reactivity feedback and the most-positive DTC. In addition, the RCS initial conditions are consistent with the most challenging conditions for MCHFR established in TR-0915-17564-A, Revision 2 and TR-108601-P-A, Revision 4.

The applicant credited several MPS signals for an increase in steam flow event, but the limiting event does not result in a reactor trip. For the limiting case for MCHFR, automatic control of the regulating control rod bank is disabled. The fuel temperature increase inserts negative reactivity and the reactor reaches a new equilibrium power level (approximately 110 percent) below the high power MPS trip setpoint. The staff notes that a slower initial increase in steam flow could also be postulated, which would similarly avoid the MPS trip setpoints and stabilize at a higher equilibrium power level above 110 percent (up to approximately 120 percent), but below the MPS trip setpoint. However, the staff agrees this would not result in a lower MCHFR value than the cases analyzed by NuScale.

In general, the high power, high power rate, and high PZR pressure signals result in reactor trips. SSI actuation generally occurs on the high PZR pressure and low pressurizer pressure signals, and DHRS actuation generally occurs on the high pressurizer pressure and low pressurizer level signals. (ML24346A273) The applicant generated high power level and high power rate trips based on an effective power level (specified in FSAR Table 15.0-7) to account for the temperature decalibration effect as a result of cooler, denser water in the downcomer affecting ex-core detector signals. The staff audited the calculation for the impact on the accuracy of the ex-core detector caused by the downcomer temperature decrease (ML24211A089). Footnote 2 on FSAR Table 15.0-7 indicates that the effective core power is reduced by a factor of 7 percent when downcomer temperature decreases by less than 10 °F, and this factor is scaled upward from 7 percent by approximately 0.6 percent/°F for the amount that the downcomer temperature decreases beyond 10 °F. While this adjustment is larger than indicated by the calculation the staff audited, the NRC staff expects that this overestimation is conservative because it will cause RTS actuation on the high power level and high power rate signals to occur later and at a higher power level during overcooling transients (ML24353A333 (nonproprietary)/ ML24353A334 (proprietary)). The staff finds that the methodology used for calculating this impact is acceptable because the result is reasonable. TR-122844, Revision 0, describes how the applicant's setpoints conform to RG 1.105. Chapter 7 of this SER presents the staff's review of TR-122844, Revision 0.

The applicant stated that no single failure causes a more limiting MCHFR. The applicant stated that although single failures of an MSIV or FWIV are possible, they were not analyzed because, for cases that result in a reactor trip, they would only occur after secondary side isolation/DHRS actuation signals, which occur after the limiting MCHFR has taken place. For cases that do not trip (such as the limiting MCHFR case), the staff notes that a single failure of an MSIV or an FWIV to the closed position would decrease the severity of the overcooling event that arose from an increase in steam flow by abating the increase in steam flow; such single failure would lead to a secondary side steam parameter change (e.g. steam superheat, steam pressure) that could then result in a reactor trip. The applicant assumed normal AC power is available for this event. A loss of AC power is not conservative because it would cause the feedwater pumps to stop and reduce the overcooling event. In addition, the staff notes that EDAS is relied on to remain functional during this event. Section 15.0.0.6.2 of this SER contains the detailed evaluation of EDAS treatment in the Chapter 15 safety analysis.

The staff audited the applicant's sensitivity studies that investigated the most limiting initial conditions and assumptions to confirm that they led to the most limiting MCHFR (ML24211A089). The audited material supports the discussions about the limiting MCHFR case in the FSAR, and the staff finds that the input parameters, initial conditions, and assumptions listed in the FSAR are suitably conservative and result in the most limiting conditions for MCHFR.

15.1.3.4.3 Evaluation of Analysis Results

The staff reviewed the analyses and the results presented in FSAR Section 15.1.3 to ensure that they meet the DSRS acceptance criteria. The staff reviewed the sequence of events table for the increase in steam flow event and finds that it is consistent with the event description and assumptions regarding protective system actuation and delay times. In addition, the staff reviewed the figures showing the transient progression and finds that they support the applicant's event description and assertion that the acceptance criteria are met. Furthermore, the figures show that the NPM reaches a stable condition, indicating that the event does not lead to a more serious plant condition.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. The analytical results for the secondary pressure figures of merit reported in Table 15.1-10, "Increase in Steam Flow (15.1.3) - Limiting Analysis Results," of the FSAR are the NRELAP5 results obtained from NRELAP5 without the adjusted DHRS condenser headers modeling, but with a bounding additional 50 psia added to the result to account for the DHRS modeling adjustment. This 50 psia is a value the applicant calculated as a conservative bound beyond the SGS pressure increase **{{ }}** due to adjusting the DHRS NRELAP5 input from the limiting loss of AC (LOAC) non-LOCA-methodology event, which the staff agrees is a bounding change to add to the previously calculated SGS pressure values.

FSAR Table 15.1-10 presents the limiting analysis results for this event. For the spectrum of increase in steam flow events, the applicant's calculated maximum RCS and SG pressures are 2,129 psia and 1253 psia, respectively, which is below 110 percent of the design values (2,420 psia). The limiting MCHFR for this event, 1.55, remains above the 95/95 limit. The staff also noted in the audited calculations that sensitivity analyses were performed to demonstrate the fuel centerline melt figure of merit was not exceeded. The case that maximized linear heat generation rate (LHGR) resulted in an **{{ }}** and subchannel analysis calculated a maximum LHGR of **{{ }}**, which is within the **{{ }}** limit for FCM (and 1 percent cladding strain).

Plant response for cases that stabilize at a new, higher power level show that design basis event acceptance criteria are met. The applicant states that no operator action is required to mitigate this design basis event. The NRC staff understands that, in reference to cases that stabilize at a new operating condition, the analysis shows that no operator action is needed to mitigate the event because the transient analysis reaches a steady-state without violating any acceptance criteria. The NRC staff finds this to be an acceptable means to demonstrate that design basis event acceptance criteria are met because it demonstrates that acceptance criteria are met for 72 hours in the absence of additional failures, consistent with Commission policy. However, although these analyses do not credit operator action for 72 hours, this should not be construed to mean that continued operation in the perturbed state for 72 hours is appropriate in the course of normal operation. The applicant has stated in audit discussions that operators will promptly follow procedures to restore the plant to prescribed operating conditions upon diagnosis of an event, and that inaction for a 72 hour period would violate operating procedures and technical specifications. For the initiating events discussed in this section, where the reactor does not trip, the indications available to the operators, such as core power, provide operators a range of variables sufficient to monitor and diagnose the event and ensure that fission product barriers provide adequate safety within prescribed operating ranges, consistent with GDC 13.

By demonstrating that the DSRS AOO acceptance criteria are met for the increase in steam flow event, the applicant satisfied the requirements associated with GDC 10, 13, 15, 20, and 26 for this transient.

15.1.3.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.1.3.

15.1.3.6 Conclusion

The staff reviewed the increase in steam flow event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. As documented above, the applicant's analysis shows that the AOO and DSRS acceptance criteria are met for this event. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of an increase in steam flow event meet the relevant requirements set forth in GDC 10, 13, 15, 20, and 26.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Introduction

FSAR Section 15.1.4, "Inadvertent Opening of Steam Generator Relief or Safety Valve," states that the NPM design does not have SG relief or safety valves but does include two MSSVs downstream of the MSIVs. The FSAR further states that an inadvertent opening of an MSSV is bounded by the increase in steam flow event analyzed in FSAR Section 15.1.3.

15.1.4.2 Summary of Application

Technical Specifications: There are no technical specifications associated with this section.

Technical Reports: There are no technical reports associated with this section.

15.1.4.3 Regulatory Basis

The regulatory basis described in SER Section 15.1.1.3 is also applicable to FSAR Section 15.1.4.

15.1.4.4 Technical Evaluation

The staff evaluated the applicant's claim that this event is bounded by an increase in steam flow event. FSAR Section 15.1.4 states that the two MSSVs together are sized to accommodate 100 percent of the full power steam flow. Therefore, a spurious opening of one MSSV would result in a steam flow of at least 50 percent (per FSAR Table 10.3-2, "Main Steam System Failure Modes and Effects Analysis (Isolation Functions),") but less than 100 percent of steam flow at full power. A spurious opening of the turbine bypass flow valve, also located downstream of the MSIVs, could result in a 100 percent increase in steam flow. For this reason, the applicant concluded that the spectrum of possible steam flow increases due to an inadvertent full or partial opening of the turbine bypass valve bounds the steam flow increase caused by a spurious opening of an MSSV and did not analyze an inadvertent MSSV opening event. Based on its review of the design information, the staff agrees with and confirms the applicant's conclusion that the inadvertent opening of an MSSV is bounded by the increase in steam flow event analyzed in FSAR Section 15.1.3.

15.1.4.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.1.4.

15.1.4.6 Conclusion

For the NPM design, the staff concludes that the inadvertent opening of a SG relief or safety valve event is bounded by the increase in steam flow event, which is discussed in Section 15.1.3 of this SER. Because the increase in steam flow event meets the DSRS acceptance criteria and GDC 10, 13, 15, 20, and 26, the staff finds that the inadvertent opening of a steam generator relief or safety valve also meets these requirements.

15.1.5 Steam System Piping Failures Inside and Outside of Containment

15.1.5.1 Introduction

A break in steam piping inside or outside of containment can cause an increase in the heat removal rate from the RCS resulting in a reduction of RCS temperature and pressure.

15.1.5.2 Summary of Application

The applicant provided an event description in FSAR Section 15.1.5.

The applicant analyzed a spectrum of steamline break (SLB) locations with varied core and plant conditions, both inside and outside containment, to determine the scenarios with the most severe results. An SLB inside the CNV would increase the pressure inside containment, reaching the high containment pressure analytical limit. The high containment pressure signal trips the reactor, isolates the CNV and CVCS, deenergizes PZR heaters, and actuates SSI and the DHRS. The break flow would decrease due to SG depressurization. This process continues until SG dryout is reached after feedwater isolation. The containment pressure is sensitive to an SLB, so the protection system detects the break sooner than a comparable break outside of containment. FSAR Section 6.2 describes the peak containment pressure response associated with an SLB event. The applicant stated that aside from containment pressure, the plant conditions for an SLB inside containment are bounded by the analysis for an SLB outside of containment because an SLB inside the CNV increases the pressure inside containment, reaching the high containment pressure MPS setpoint and the high containment pressure signal actuates RTS, isolates the CNV and CVCS, deenergizes PZR heaters, and actuates SSI and DHRS to mitigate the event.

An SLB outside the CNV would cause an increase in steam flow event that could either cause a low SG pressure signal or a high core power or power rate trip due to the reactor power response from the decreased RCS temperature. The break flow would be stopped by the closure of the MSIV and depressurization of the steam system piping or SG dryout from FW isolation. The applicant concluded that a small SLB outside of containment is the most limiting type because it provides the longest event progression before detection by the protection system. The applicant stated that it evaluated cases with both turbine control enabled and disabled to identify the most limiting configuration. Similarly, it evaluated cases with both automatic control of the regulating bank enabled and disabled to identify the most limiting configuration.

The applicant stated that the SLB accident was evaluated against the 95/95 MCHFR AOO acceptance criteria.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this FSAR section

15.1.5.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 27, as it relates to controlling the rate of reactivity changes to ensure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 28, as it relates to limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other RPV internals to impair significantly the capability to cool the core.

- GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

The DSRS Section 15.1.5, “Steam System Piping Failures Inside and Outside of Containment,” lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

15.1.5.4 Technical Evaluation

The following sections discuss the staff’s technical evaluation of the applicant’s SLB analysis.

15.1.5.4.1 Causes

The staff reviewed FSAR Section 15.1.5 to assess the applicant’s identification of causes leading to this event. The staff notes, from FSAR Section 15.1.5.1, “Identification of Causes and Accident Description,” that because a steam line break in the NuScale plant can result from various mechanisms, the applicant considered a spectrum of SLB sizes and locations, with varying core and plant conditions, to determine the SLB scenarios with the most limiting results. The applicant states that SLB is assumed to occur inside or outside the CNV.

The applicant determined that for each of the acceptance criteria, a different SLB case was most limiting. Therefore, the applicant presented various limiting cases in this FSAR section for each of the acceptance criteria. The limiting cases presented in FSAR Section 15.1.5 are the following: limiting RCS pressure case, limiting SG pressure case, limiting MCHFR case. The staff also notes that the applicant classifies this event as an accident, which is consistent with the DSRS. Although the applicant classifies the event as an accident, FSAR Section 15.1.5.1 states that small size steam line breaks are more likely and evaluates these against AOO criteria. The NRC staff finds this acceptable as AOO acceptance criteria are more restrictive than accident acceptance criteria. The staff finds the applicant's assessment of causes leading to the event acceptable because it considers a spectrum of SLBs in different locations throughout the system.

FSAR 15.0.3 presents the radiological analysis of the SLB accident. This analysis does not credit holdup or dilution in the secondary system but assumes that primary coolant is released directly to the environment at the maximum primary-to-secondary release rate allowed by design-basis limits.

15.1.5.4.2 Methodology

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER to analyze the thermal-hydraulic response to the event.

The applicant performed a subchannel analysis using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. Section 15.0.2 of this SER presents the staff's evaluation of these codes.

15.1.5.4.3 Model Assumptions, Input, and Boundary Conditions

The staff reviewed the applicant's modeling assumptions, analysis input, and boundary conditions to assess the adequacy of the transient analysis model. The staff noted that some input assumptions varied slightly among the limiting SLB cases presented because some of these parameters would affect different aspects of the transient. In any case, the staff reviewed each limiting case to determine if the applicant's proposed parameters were conservatively chosen.

For all cases presented as part of the FSAR Section 15.1.5 analysis, the staff confirmed that initial parameters, such as power level, RCS temperature, PZR pressure, PZR level, RCS flow, scram characteristics (including the assumption of a stuck rod), Doppler reactivity feedback, moderator temperature reactivity feedback, SG characteristics, and DHRS characteristics, were chosen conservatively. The staff also confirmed that NuScale considered instrument inaccuracies and credited no operator action to mitigate the consequences of an SLB. The staff confirmed through an audit (ML24211A089) that the applicant's modeling assumptions, analysis input, and boundary conditions were supported by sensitivity analyses.

The staff reviewed the applicant's single-failure assumptions for this event. The staff confirmed that the applicant considered and analyzed single failures for each limiting case of the SLB event.

For the limiting RCS pressure case and limiting MCHFR case, the limiting break is between the containment upper head and the first (primary) MSIV. This means the break is unisolable and leads to a loss of DHRS inventory and functionality on one of two DHRS trains. For these cases, the applicant modeled a single failure of the FWIV on the affected train. The staff confirmed that a failure of the FWIV on the affected train will maximize mass and energy release after an isolation signal occurs, which is conservative for the RCS pressure case. Failure of the FWIV to close does not affect the limiting CHFR as secondary side isolation occurs after the time of MCHFR. For the limiting SG pressure case, the staff noted that the applicant modeled a single failure of the FWIV on the affected SG. The staff confirmed that a failure of the FWIV, which allows additional FW into the affected SG, is conservative for the SG pressure case.

The staff reviewed the applicant's assumptions regarding break size and location. The staff noted that, as with other model assumptions, the limiting break size and location were dependent on which figure of merit (e.g., RCS pressure, MCHFR, or SG pressure) was being analyzed. Nevertheless, the staff confirmed through an audit (ML24211A089) that the applicant's assumed limiting break sizes and locations were supported by sensitivity analyses.

The staff reviewed the applicant's assumptions regarding the availability or unavailability of electric power systems. The staff confirmed that, for each limiting case, the applicant's power assumptions were conservatively determined. The potential loss of the normal direct current power system (EDNS) is assessed based on the discussion in FSAR Section 15.0.0The applicant determined that the limiting SLB case is when EDNS is available. If a loss of EDAS is not assumed at event initiation, EDAS is relied on to remain functional during the remainder of the event. Section 15.0.0.6.2 of this SER contains the detailed evaluation of EDAS treatment in the Chapter 15 safety analysis.

The staff finds the applicant's SLB analysis-specific assumptions, input, and boundary conditions acceptable because they were selected conservatively and demonstrate acceptable mitigation of this event and protection of fission product barriers.

15.1.5.4.4 Evaluation of Analysis Results

The staff reviewed the results presented in FSAR Section 15.1.5, to determine if they meet the DSRS acceptance criteria. The staff reviewed the transient behavior of several parameters by evaluating plots of the parameters as a function of time. The staff specifically reviewed the calculated reactor power, reactor and SG pressures, core temperatures, break flow rates, MCHFR, and reactivity.

As part of its review of these transient parameters, the staff verified that the sequence of events was reasonable given the automatic actuations of protection systems at their analytical setpoints.

The staff reviewed the applicant's SLB case that resulted in a limiting RCS pressure. The staff confirmed that for the worst RCS pressure case, the RCS pressure remained below 110 percent of the design pressure. The staff finds this acceptable because this meets the DSRS acceptance criteria.

The staff reviewed the applicant's SLB case that resulted in a limiting SG pressure. The staff confirmed that for the worst SG pressure case, the SG pressure remained below 110 percent of the design pressure. The staff finds this acceptable because this meets the DSRS acceptance criteria.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers NRELAP5 input. For this event, the reported analytical results relative to the MCHFR and RCS pressure figures of merit are the results with the adjusted DHRS modeling. The analytical results for secondary pressure figures of merit reported in FSAR Table 15.1-15, “Steam Piping Failure (15.1.5) - Limiting Analysis Results,” of the FSAR are the NRELAP5 results obtained after adjusting the DHRS modeling.

The staff notes that for an SLB accident, the secondary side naturally depressurizes, and thus, in any given case, the main steam system pressure is of no concern. However, the staff notes that, in general, when the DHRS is actuated, a large pressurization of the DHRS loops, which each include one steam generator will occur. While the DHRS loops pressurize far beyond the normal SGS operating pressures, which vary with primary side power, this pressurization is understood and expected and the reason the SGS is designed to be comparable to the primary side nominal operating pressure. Nothing about a SLB accident would give reason to expect the SGs in the DHRS loops to be challenged at their pressure limits. The staff audited supporting calculations and found that the liming case of SGS pressure resulted from a one percent break on the turbine inlet piping with and failure of one FWIV to close, contemporaneous with normal DHRS actuation. As expected, main steam system pressure is not challenged during an SLB.

The staff reviewed the applicant’s SLB case that resulted in a limiting MCHFR. The staff confirmed that for the worst MCHFR case, the MCHFR remained above the 95/95 DNBR limit described in the DSRS.

The staff notes that FSAR Section 15.0.3 presents the radiological analysis of the SLB accident . Section 15.0.3 of this SER documents the staff’s review of the radiological consequences occurring from an SLB. NRC staff audited (ML24211A089) the calculations supporting the SLB evaluation and noted that NuScale considered releases from limiting break sizes, locations, and loss of power scenarios to confirm that the release analyzed in FSAR 15.0.3 is indeed bounding. The limiting release scenario considered by the applicant is a steamline break inside containment with a failure of EDAS at time zero, which results in some of the steam from the RVVs releasing through the MSL up to the time of isolation. The staff finds the applicant’s evaluation acceptable because it considers limiting break sizes, locations, and loss of power scenarios and confirms that the licensing basis radiological consequence analysis remains bounding.

The staff confirmed that the DHRS is safety-related, and under the worst break locations for this event, which could render one train of the DHRS completely inoperable, the second train of the DHRS automatically actuates and provides an adequate amount of heat removal, in conjunction with pressure relief from the RSVs, to cool the core during and after the accident.

15.1.5.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.1.5.

15.1.5.6 Conclusion

The staff concludes that the consequences of postulated SLBs meet the relevant requirements set forth in the GDC 13, 27, 28, and 31 with respect to this event. As documented above, for this event sequence, the staff confirmed that the applicant's analysis demonstrates that the DSRS acceptance criteria are met.

15.1.6 Loss of Containment Vacuum and Containment Flooding

15.1.6.1 Introduction

In the NuScale design, the CNV, which is partially submerged in the reactor pool, is normally kept at a low vacuum internal pressure. This serves to insulate the RPV from the relatively cooler pool water during normal operations. An event resulting in a loss of containment vacuum conditions, whether through air or water ingress into the containment, would degrade the insulation function provided by the containment vacuum and thereby increase the heat transfer from the RPV to the pool, with an effect similar to an overcooling event. Overcooling events have the potential to decrease moderator temperature, which increases core reactivity, and can therefore lead to both an increase in reactor power and RCS pressure, and reductions in MCHFR. This event is expected to be of moderate frequency when compared to a pipe break, as it could result from operator action or equipment malfunction and is therefore classified as an AOO.

15.1.6.2 Summary of Application

FSAR Section 15.1.6, "Loss of Containment Vacuum and Containment Flooding," summarizes the analyses performed by the applicant related to the loss of containment vacuum or containment flooding event. During normal operation, the CES maintains the containment vacuum conditions. FSAR Section 9.3.6, "Containment Evacuation System," and Section 9.3.7, "Containment Flooding and Drain System," discuss the CES. As stated by the applicant, a failure of this system could lead to an increase in containment pressure. Another means of losing containment vacuum is a pipe break or leakage of sufficient quantity to overwhelm the CES, such that containment begins to flood. In both scenarios, more severe transients exist that could cause similar effects but have other impacts and are analyzed in other sections of the FSAR; for example, in FSAR Section 15.2.8, "Feedwater System Pipe Breaks Inside and Outside Containment," or FSAR Section 15.1.5.

The applicant stated that the loss of vacuum event is bounded by a containment flooding event. The most severe containment flooding event that is not bounded by another existing event analyzed in Chapter 15 is initiated as a break or leak in the reactor component cooling water system (RCCWS). In the event of a RCCWS leakage, the fluid leaking from the RCCWS would not immediately boil because the fluid temperature is below the CNV saturation temperature and would begin filling the containment. This scenario allows for the containment to partially fill with water, increasing the heat losses from the RPV, while not generating substantial vapor so that containment pressure remains below the trip setpoint. As with other overcooling events, if a containment flooding event trips the reactor, the subsequent pressure rises in the primary and secondary systems are a result of the isolation functions and not a direct consequence of the containment flooding.

The limiting containment flooding event is based on an RCCWS break inside containment, with one RCCWS pump operating following the break event. Additionally, a CES pump operates at capacity, which delays the onset of the containment pressure trip, with no trip occurring during the period of analysis. The applicant stated that the most severe sequence for this scenario results from retaining all power, with no single failures resulting in a more limiting response with respect to the acceptance criteria. Other combinations of available power (AC, non-safety-related DC, and highly reliable DC) are not analyzed.

For the analyses, the applicant used initial conditions intended to maximize RCS power over the longest duration such that the entire RCCWS volume flows into the CNV, which increases heat losses from the RPV and therefore increases the magnitude of the overcooling. FSAR Section 15.1.6.3.2, "Input Parameters and Initial Conditions (Limiting Minimum Critical Heat Flux Ratio Case)," and Table 15.1-16, "Loss of Containment Vacuum/Containment Flooding—Inputs," summarize these input parameters. FSAR Figures 15.1-44, "Reactor Power (15.1.6 Containment Flooding)," through 15.1-48, "Figure 15.1-48: Critical Heat Flux Ratio (15.1.6 Containment Flooding)," show the results for the limiting case. The applicant stated that this event results in a small overcooling transient with a slightly degraded MCHFR that is bounded by the other overcooling transients.

The FSAR further states that for containment flooding cases in which the high containment pressure MPS setpoint is reached, reactor trip, containment isolation, and DHRS actuation ensure a return to a stable condition with no operator actions.

Technical Specifications: The following GTS are applicable to this area of review:

- LCO 3.4.7, "RCS Leakage Detection Instrumentation."
- the GTS listed in Section 15.0.0 of this SER.

Technical Reports: There are no technical reports related to FSAR Section 15.1.6.

15.1.6.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that SAFDLS are not exceeded during normal operations, including AOOs.
- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
- GDC 15, as it relates to the RCS being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.
- GDC 20, as it relates to the reactor protection system being designed to automatically initiate appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs.

- GDC 26, as it relates to the ability of the design to reliably control reactivity changes to ensure that SAFDLs are not exceeded, including during AOOs.

DSRS Section 15.1.6, “Loss of Containment Vacuum,” lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

15.1.6.4 Technical Evaluation

A loss of containment vacuum event is unique to the NuScale design. Because of the coupling and relative size of the RPV, the CNV, and the reactor pool, what would be a relatively benign liquid release in a traditional PWR could have an appreciable impact on the primary-side heat balance for the NuScale design. The containment is normally kept at vacuum conditions (less than 6900 Pa (1.0 psia)), and therefore normally serves to insulate the RPV from the cooler reactor pool. An event in which the vacuum condition is not maintained, whether because of the ingress of water vapor or air, or the flooding of containment, degrades that insulation function. This results in an event sequence that falls outside of the traditional Chapter 15 scope but requires analysis to confirm that safety acceptance criteria are met.

15.1.6.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including the NRELAP5 code, to model the thermal-hydraulic response to the event. In addition, the applicant performed a subchannel analysis using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. The statistical subchannel analysis methodology developed in TR-108601-P-A, Revision 4, is used to calculate the uncertainties associated with the MCHFR using the VIPRE-01 computer code. Section 15.0.2 of this SER describes the staff’s evaluation of these codes.

15.1.6.4.2 Input Parameters, Initial Conditions, and Assumptions

One means of losing containment vacuum involves a failure or degradation in CES capability such that a small amount of air or water vapor is allowed to exist within the containment. In the long term, this is similar to operating at a higher containment pressure while under the trip setpoint. The staff audited loss of containment vacuum and containment flooding transient analysis and confirmed that loss of containment vacuum was evaluated for the NPM-20 design and figures of merit are more limiting for the containment flooding event.

As stated in the FSAR, another cause of a loss of containment vacuum is a pipe break or leakage of sufficient volume to overwhelm the CES pump flow rate. Non-RCS fluid sources include the feedwater line, the main steamline, the RCCWS, which cools the control rod drive system. RCS breaks could also cause containment flooding events, but most would have additional consequences and are analyzed in other Chapter 15 sections. Some of these events are analyzed separately: feedwater pipe breaks are evaluated in FSAR Section 15.2.8, “Feedwater System Pipe Breaks Inside and Outside of Containment”; SLBs are evaluated in Section 15.1.5; and CVCS line breaks are evaluated in Section 15.6.5, “Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary.”

The applicant also stated that the RCCWS break bounds the non-safety-related RCS high-point vent line with regards to the containment flooding event because of the subcooled nature of the break, which results in additional fluid being released into containment before the high-pressure trip setpoint is reached. The NRC staff finds this conclusion to be consistent with the basic physical phenomena, especially given that other effects associated with the RCS high-point vent line break are assessed as part of the break spectrum analyzed in FSAR Section 15.6.5. Therefore, the only containment flooding event analyzed in Section 15.1.6 is the RCCWS line break.

With respect to the transient initial conditions, the applicant selected values that maximize RCS power and thereby maximize the amount of overcooling. The RCCWS is assumed to operate at minimum temperature so that liquid conditions in the containment are most conducive to transferring heat from the RPV to the reactor pool. The number of operating RCCWS pumps is varied to identify the most limiting case. Initial power is maximized, and the most positive DTC and most negative MTC are used to produce a limiting power response. The NRC staff reviewed these conditions, as well as the related analyses of other input conditions in calculations audited by the staff (ML24211A089) and determined that the values selected by NuScale represent a conservative set of values for this transient, as they produce the most limiting power response.

In the event of an RCCWS line break inside containment, the CNV slowly begins to fill with a mixture of liquid water and water vapor as the result of the low vapor pressure inside the CNV. The CES then acts to remove vapor as the containment pressure increases. For the purposes of this transient, this assumption is conservative because maintaining containment pressure under the trip setpoint prolongs the transient as the CNV fills with water. During the transient, the RCCWS line break is assumed to flood containment with a 30,500 lb mass, which bounds the RCCWS inventory (ML24346A276). As documented in FSAR Section 9.2.2, “Reactor Component Cooling Water System,” and supplemented by ML24346A276, the volume of RCCWS fluid is limited by the available inventory in the RCCWS system, as makeup to the system requires operator action. Leaks from the RCCWS can be detected through a variety of means, including those associated with RCS leakage.

FSAR Section 15.1.6.2, “Sequence of Events and Systems Operation,” notes that there are no single failures that would result in more severe calculated values with respect to the acceptance criteria. The staff reviewed the potential for single failure and determined that this is true, provided single failures leading to other transients analyzed in Chapter 15 are also considered, as these events would bound the scenario discussed here. The staff simulated scenarios related to RCCW flooding, in particular the limiting CNV peak pressure event, and determined that the RCCW injection increases condensation cooling and reduces peak pressure although CNV free volume is reduced. Section 6.2 of this SER contains the staff’s evaluation of events relative to the containment pressure figure of merit.

15.1.6.4.3 Evaluation of Analysis Results

The results from the limiting MCHFR sequence discussed previously are displayed in FSAR Figures 15.1-44 through 15.1-48. These results include the uncertainties determined using statistical subchannel analysis methodology described in TR-108601-P-A, Revision 4. Figure 15.1-44 shows reactor power during the transient; as stated previously, the reactor does not trip, and thus, reactor power stabilizes about 0.3 percent higher than the initial conditions.

Figure 15.1-45, “Reactor Component Cooling Water System Break Flow Rate (15.1.6 Containment Flooding),” displays the RCCW break flow, which is a constant based on the system flow rate until the assumed inventory is depleted. Figure 15.1-46, “Reactor Coolant System Average Temperature (15.1.6 Containment Flooding),” provides the average RCS coolant temperature. It shows that coolant temperatures decrease early in the transient due to additional RPV heat losses, then increase as power increases later in the transient. RCS average temperature remains within 1 °F of the initial RCS temperature. Figure 15.1-48 shows the figure of merit for the analysis, MCHFR. Figure 15.1-47, “Liquid Levels above Top of Active Fuel (15.1.6 Containment Flooding,” shows the liquid levels above the top of active fuel and in the CNV. Because the applicant assumed a limited inventory in the RCCW system, the CNV level value rapidly increases to level off at a little over 13 feet in the CNV.

From the information provided in the FSAR, in conjunction with the material audited (ML24211A089) by the NRC staff, this is a transient sequence different from most of the Chapter 15 events, as this transient does not involve a reactor trip for the limiting MCHFR sequence. This results in minor differences in determining what constitutes the transient end state, with the applicant choosing to terminate the transient upon reaching stable conditions at a higher reactor power level.

Because no trip setpoint is reached for the limiting MCHFR sequence, and the transient approaches an equilibrium state, the staff accepts this choice as reasonable for the loss of containment vacuum event, especially given that the figures of merit for the relevant acceptance criteria, discussed in the following paragraph, are bounded by the values calculated for other Chapter 15 events. The applicant did perform sensitivity analyses for primary and secondary pressures, and the staff confirmed in its audit of the applicant’s calculations that those event sequences result in MPS trip setpoints for high containment pressure being reached. The values for the peak pressures for those events are tabulated in Table 15.1-17, “Loss of Containment Vacuum/Containment Flooding (15.1.6) – Limiting Analysis Results,” of the FSAR and pressure in the reactor coolant and main steam systems is maintained below 110 percent of the design values. Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. The analytical results for the secondary pressure figures of merit reported in Table 15.1-17 of the FSAR are the NRELAP5 results obtained from NRELAP5 without the adjusted DHRS condenser headers modeling, but with a bounding additional 50 psia added to the result to account for the DHRS modeling adjustment. This 50 psia is a value the applicant calculated as a conservative bound beyond the SGS pressure increase **{{ }}** due to adjusting the DHRS NRELAP5 input from the limiting LOAC non-LOCA-methodology event, which the staff agrees is a bounding change to add to the previously calculated SGS pressure values.

The acceptance criteria for AOOs include RCS pressure, steam pressure, containment pressure, MCHFR and fuel centerline temperature. In addition, an AOO should not progress into a more serious event without other faults occurring independently. By demonstrating that these acceptance criteria are met, the applicant satisfied the requirements associated with GDC 10, 15, 20, and 26 for this transient. For the limiting MCHFR event sequence, the applicant’s analysis demonstrates that the minimum DNBR of 2.50 remains above the 95/95 limit. Fuel centerline temperature and containment pressure are considered nonlimiting for this event, with no challenge to limits expected and other events substantially bounding the values calculated for this transient. For this sequence, the reactor does not trip but stabilizes at a new, higher power level. Since the transient does not progress into a more severe event, the staff agrees with the applicant's conclusion that this AOO meets the defined thermal-hydraulic acceptance criteria, and the radiological barriers will not be challenged. Therefore, the staff finds that GDC 10, 15, 20, and 26 are met for this transient.

Plant response for cases that stabilize at a new, higher power level show that DBE acceptance criteria are met. The applicant stated that no operator action is required to mitigate this design basis event. The NRC staff understands that, in reference to cases that stabilize at a new operating condition, the analysis shows that no operator action is needed to mitigate the event because the transient analysis reaches a steady-state without violating any acceptance criteria. The NRC staff finds this to be an acceptable means to demonstrate that DBE acceptance criteria are met because it demonstrates that acceptance criteria are met for 72 hours in the absence of additional failures, consistent with Commission policy. However, although these analyses do not credit operator action for 72 hours, this should not be construed to mean that continued operation in the perturbed state for 72 hours is appropriate in the course of normal operation. The applicant has stated in audit discussions that operators will promptly follow procedures to restore the plant to prescribed operating conditions upon diagnosis of an event, and that inaction for a 72 hour period would violate operating procedures and technical specifications. For the initiating events discussed in this section, where the reactor does not trip, the indications available to the operators, particularly those indications relied on to monitor reactor leakage as stipulated by TS LCO 3.4.5, "RCS Operational Leakage," provide operators a range of variables sufficient to monitor and diagnose the event and ensure that fission product barriers provide adequate safety within prescribed operating ranges, consistent with GDC 13. The plant response for this transient in the event of a reactor trip, as discussed previously, is enveloped by other transients evaluated in Chapter 15.

15.1.6.5 Combined License Information Items

There are no COL information items related to FSAR Section 15.1.6.

15.1.6.6 Conclusion

The staff concludes that the analysis of transients resulting from loss of containment vacuum or flooding of the containment analyzed here is acceptable and meets the requirements of GDC 10, 13, 15, 20, and 26 with respect to this event. As documented above, the staff found for this event sequence, the applicant's analysis showed that the AOO acceptance criteria are met. Other events analyzed elsewhere (FSAR Section 15.1.5 and15.2.8) in Chapter 15 substantially bound the values calculated for this transient for more severe containment flooding events. For this transient, the plant stabilizes at new, higher power level and does not trip, and therefore no escalation to a more serious plant condition will occur.

15.2 Decrease in Heat Removal by the Secondary System

15.2.1 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum

This SER section documents the staff's review of FSAR Sections 15.2.1 through 15.2.3, "Loss of External Load," "Turbine Trip," and "Loss of Condenser Vacuum," respectively. These events are discussed together because the transient responses are highly similar, and the applicant presented a single set of bounding results that envelops these three events.

15.2.1.1 Introduction

The staff reviewed the events in FSAR Sections 15.2.1 through 15.2.3 to ensure that they are analyzed appropriately and meet the acceptance criteria outlined in Section 15.2.1.3 of this SER. The loss of external load (LOEL), turbine trip (TT), and loss of condenser vacuum (LOCV) events all result in a decrease in heat removal by the secondary system and a corresponding temperature and pressure increase in the RCS. Secondary pressure also increases.

15.2.1.2 Summary of Application

The information provided by the applicant in FSAR Sections 15.2.1 through 15.2.3 is summarized below.

An LOEL is an AOO caused by loss of most or all of the turbine generator load. The LOEL generates a TT that results in isolating the steam flow from the SGs to the turbine because of the closure of the turbine control valves (TCVs). The NPM design includes a turbine bypass valve that opens to allow the reactor to remain in operation in the event of a TT by transferring the main steam flow to the condenser. However, the events in FSAR Sections 15.2.1 through 15.2.3, do not credit the turbine bypass system. Therefore, the primary and secondary-side temperatures and pressures increase until a reactor trip is issued, SSI is initiated, and the DHRS is actuated. The DHRS transfers decay heat to the reactor pool. The severity of the event is ultimately determined by how long it takes to initiate and establish a steady cooldown via the DHRS.

A TT is an AOO that may result from many different conditions that cause the turbine generator control system to initiate a TT signal or a failure in the control system itself. A TT initiates closure of the turbine stop valves (TSVs), which terminates steam flow from the SGs to the turbine. The TT event proceeds similarly to the LOEL event.

An LOCV is an AOO that may occur because of a reduction in condenser cooling, failure of the main condenser evacuation system to remove noncondensable gases, or in-leakage of air. An LOCV is similar to the LOEL and TT events, except that a loss of feedwater also occurs at event initiation as a result of the loss of net positive suction head for the condensate pumps. Therefore, a similar system response results.

The LOEL, TT, and LOCV events potentially challenge the RCS pressure, SG pressure, and MCHFR acceptance criteria. Therefore, the applicant evaluated the limiting cases with respect to each of the acceptance criteria in the FSAR. The applicant stated that the LOEL, TT, and LOCV events resulted in almost identical responses and therefore included a single bounding set of figures for the three events. The applicant concluded that the limiting cases meet all acceptance criteria.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.2.1.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS being designed with appropriate margin so SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so the RCPB is not breached during normal operations, including AOOs.

- GDC 26, as it relates to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

NuScale DSRS Sections 15.2.1 through 15.2.5 list the following acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections:

- The most limiting moderate-frequency event that results in an unplanned decrease in secondary system heat removal is identified, in particular as to primary pressure, secondary pressure, and long-term decay heat removal.

- The predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

- Fuel cladding integrity must be maintained by the minimum DNBR (for NuScale, MCHFR) remaining above the 95/95 limit based on acceptable correlations (see FSAR Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.

- An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

- Plant protection systems' setpoints assumed in the transient analyses are selected with adequate allowance for measurement inaccuracies as delineated in RG 1.105.

- Event evaluations consider single failures, operator errors, and performance of non-safety-related systems consistent with RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)" (June 2007).

- The applicant should analyze these events using an acceptable analytical model. Any other analytical model proposed by the applicant will be evaluated by the staff for acceptability.

15.2.1.4 Technical Evaluation

15.2.1.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to model the NPM thermal-hydraulic response to the LOEL, TT, and LOCV events. The applicant performed subchannel analyses using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR as stated in FSAR Section 4.4.2.5. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.2.1.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the transient analysis model. As stated previously, the LOEL, TT, and LOCV events are similar. The main difference is that an LOEL causes TCV closure, while a TT initiates TSV closure. The applicant models TCV and TSV closure as an instantaneous closure in the calculation audited by the staff. Therefore, plant responses are conservatively modeled identically for TT and LOEL. An LOCV is similar to a TT except that it also includes a loss of feedwater at event initiation, which is typically limiting for RCS pressure and MCHFR responses.

The conditions, biases, and assumptions used to determine limiting values for each acceptance criterion (RCS pressure, SG pressure, or MCHFR) may be different to maximize the consequences for the acceptance criterion being considered. Therefore, some differences occur in the event progressions for each acceptance criterion. However, one common assumption is that the applicant does not credit operator action. In addition, each of the analyses assumes that plant control systems perform as designed, with allowance for instrument inaccuracy, unless their action mitigates the events. Furthermore, the applicant did not assume loss of DC power in any case, which the staff finds acceptable because it would result in actuation of the ESFs earlier in the transient, thus leading to less severe consequences.

The limiting RCS pressure case is a TT event with a concurrent loss of AC power. The immediate loss of feedwater leads to reduced heat removal by the secondary and RCS pressurization. The high PZR pressure signal actuates reactor trip, SSI, and DHRS. The SSI closes the FWIVs, MSIVs, and the non-safety-related FWRVs and secondary MSIVs, and DHRS actuation opens the DHRS valves. The high PZR pressure trip also deenergizes the PZR heaters, though for this case, they are already deenergized because of the loss of AC power. A reactor safety valve (RSV) opens to mitigate the RCS pressure increase. The applicant determined that no single failure is limiting for the RCS pressure case.

The limiting SG pressure scenario is a TT event with an assumed failure of an FWIV to close, which causes feedwater flow to continue to the affected SG until the non-safety related FWRV closes. This maximizes SG pressure and leads to a reactor trip as well as SSI and DHRS actuation on high PZR pressure. A loss of AC power is not limiting for SG pressure because it would terminate feedwater flow at the start of the transient.

The limiting MCHFR case is a TT event with a concurrent loss of AC power, which leads to a reactor trip as well as SSI and DHRS actuation on high PZR pressure. The immediate loss of feedwater leads to reduced heat removal by the secondary side. An RSV lifts to mitigate the pressure increase. The applicant indicated that no single failure results in a more limiting MCHFR. The staff agrees no single failure would be more limiting.

The staff reviewed the input parameters and initial conditions for each of the limiting cases to ensure that the applicant selected conservative values for the analyses. The staff notes that the applicant assumed suitably conservative parameters as described in DSRS Sections 15.2.1 through 15.2.5, including a 102 percent initial core power level, maximum time delay to scram with the most reactive rod held out of the core, and the most limiting reactivity feedback. The BOC reactivity feedback is conservative for RCS overheating events because the reactivity coefficients are least negative and therefore minimize the negative reactivity feedback resulting from temperature increases. In addition, the assumptions of maximum decay heat and reactor pool temperature are limiting for overheating events since they present the greatest challenge to heat removal.

The staff audited the applicant's sensitivity studies that investigated the most limiting initial conditions, single failures, and loss of power assumptions to confirm that they led to the most limiting results (ML24211A089). The audited material supports the discussions in the FSAR, and the staff finds that the input parameters and initial conditions listed in the FSAR are suitably conservative and result in the most limiting conditions for each of the respective acceptance criteria.

15.2.1.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Sections 15.2.1 through 15.2.3, to ensure they meet the DSRS acceptance criteria. The staff reviewed the sequence of events tables for the LOEL, TT, and LOCV events and notes that they are consistent with the event descriptions and assumptions regarding protective system actuation and delay times. In addition, the staff reviewed the figures showing the transient progressions and finds that they support the applicant's event descriptions and assertion that the acceptance criteria are met. Furthermore, the figures show that the NPM reaches a stable condition, indicating that the events do not lead to more serious plant conditions.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. For this event, the reported analytical results relative to the MCHFR and RCS pressure figures of merit are the results without the adjusted DHRS modeling, which the staff finds acceptable as MCHFR and RCS pressure are not significantly impacted by this modeling adjustment. The analytical results for secondary pressure figures of merit reported in Table 15.2-7 of the FSAR are the NRELAP5 results obtained after adjusting the DHRS modeling.

FSAR Table 15.2-7, "Loss of External Load -Turbine Trip - Loss of Condenser Vacuum – Limiting Analysis Results," presents the limiting analysis results for these events. The staff finds that the predicted plant response for the most limiting events satisfies the AOO acceptance criteria because the analysis demonstrates that the limiting MCHFR of 2.49 is above the 95/95 limit of 1.43; the maximum RCS pressure of 2,300 psia remains below 110 percent of the RCS design pressure (2,420 psia); and the maximum peak secondary pressure of 1451 psia remains below 110 percent of the secondary system design pressure (2,420 psia).

By demonstrating that the AOO acceptance criteria are met for the most limiting LOEL, TT, and LOCV events, the applicant satisfied the requirements associated with GDC 10, 13, 15, and 26 for this transient.

15.2.1.5 Combined License Information Items

There are no COL information items associated with FSAR Sections 15.2.1 through 15.2.3 of.

15.2.1.6 Conclusion

The staff reviewed the LOEL, TT, and LOCV events, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transients. As documented above, for these events, the applicant's analyses show that the AOO and DSRS acceptance criteria are met. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of an LOEL, TT, and LOCV meet the relevant requirements set forth in GDC 10, 13, 15, and 26 with respect to these events.

15.2.2 Turbine Trip

The staff's review of this event is documented in SER Section 15.2.1.

15.2.3 Loss of Condenser Vacuum

The staff's review of this event is documented in SER Section 15.2.1.

15.2.4 Closure of Main Steam Isolation Valve(s)

15.2.4.1 Introduction

The staff reviewed FSAR Section 15.2.4, "Closure of Main Steam Isolation Valve(s)," to ensure that the event was analyzed appropriately and meets the acceptance criteria outlined in Section 15.2.1.3 of this SER. The inadvertent closure of one or both MSIVs is an AOO resulting from a steamline or reactor system malfunction or inadvertent operator actions. The event results in rapid pressurization in the primary and secondary systems, as well as an RCS temperature increase.

15.2.4.2 Summary of Application

The information provided by the applicant in FSAR Section 15.2.4 is summarized below.

The MSIV closure event is initiated by the closure of one or both MSIVs because of a spurious closure signal or operator error. One MSIV could also fail to close on a valid MSIV closure signal. The closure of one or more MSIVs results in an increase in secondary temperature and pressure in the affected SG(s) and consequent increases in primary temperature and pressure as the result of reduced heat removal. A reactor trip, actuation of DHRS, and SSI on the high PZR pressure signal are credited to mitigate the event. The RSV lifts for a short time to limit the RCS pressure increase for the cases that reach the RSV setpoint.

The MSIV closure event potentially challenges the RCS pressure, SG pressure, and MCHFR acceptance criteria. Therefore, the applicant evaluated the limiting cases for each of the acceptance criteria as defined in the FSAR Table 15.0-2 for AOOs. The applicant concluded that the limiting cases meet all acceptance criteria.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.2.4.3 Regulatory Basis

The regulatory basis described in SER Section 15.2.1.3 is also applicable to FSAR Section 15.2.4.

15.2.4.4 Technical Evaluation

15.2.4.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to model the NPM thermal-hydraulic response to the MSIV closure event. The applicant performed subchannel analyses using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.2.4.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the transient analysis model. The staff noted that the applicant identified three different limiting cases to maximize the consequences for the acceptance criterion being considered. However, the applicant did not credit operator action in the analyses for any of these cases. In addition, each of the cases assumes that plant control systems perform as designed, with allowance for instrument inaccuracy, unless their action mitigates the events.

The applicant stated that the closure of two MSIVs is limiting for all acceptance criteria. This agrees with the staff's expectations, as closing two MSIVs leads to isolation of both SGs and the maximum loss of heat removal from the RCS. The applicant also stated that no single failures resulted in more severe results for the RCS pressure and MCHFR acceptance criteria.

For the SG pressure figure of merit, no single failures resulted in more severe results either. However, the applicant states that although sensitivity analyses demonstrated that it was not the case for these analyses, the single failure of an FWIV can lead to the limiting SG pressure as the feedwater system continues filling the SG until the FWRV closes, which is the non-safety related backup.

The MSIV closure event progresses as described in Section 15.2.4.2 of this SER for each of the limiting cases. The RCS pressure and MCHFR cases assume a LOOP at event initiation. This causes a feedwater pump trip, exacerbating the RCS heatup. Loss of AC power was not limiting for SG pressure. Furthermore, the applicant did not assume loss of DC power in any case, which the staff finds acceptable because it would result in ESF actuation sooner in the transient, thus leading to less severe consequences.

The staff reviewed the input parameters and initial conditions for each of the limiting cases to ensure that the applicant selected conservative values for the analyses. The staff notes that the applicant assumed suitably conservative parameters as described in DSRS Sections 15.2.1 through 15.2.5, including a 102 percent initial core power level, maximum time delay to the most reactive rod held out of the core, and the most limiting reactivity feedback (which occurs at BOC for this event). In addition, the assumptions of biased-high reactor pool temperature and maximum DHRS valve opening time are limiting for overheating events since they present the greatest challenge to decay heat removal.

The staff audited the applicant's sensitivity studies that investigated the most limiting initial conditions, single failures, and loss of power assumptions to confirm that they led to the most limiting results (ML24211A089). The audited material supports the discussions in the FSAR, and the staff finds that the input parameters and initial conditions listed in the FSAR are suitably conservative and result in the most limiting conditions for each of the respective acceptance criteria.

15.2.4.4.3 Evaluation of Analysis Results

The staff reviewed the results presented in FSAR Section 15.2.4 to ensure that they meet the DSRS acceptance criteria. The staff reviewed the sequence of events tables for the MSIV closure event and notes that they are consistent with the event description and the assumptions for protective system actuation and delay times. In addition, the staff reviewed the figures showing the transient progressions and finds that they support the applicant's event descriptions and assertion that the acceptance criteria are met. Furthermore, the figures show that the NPM reaches a stable condition, indicating that the event does not lead to more serious plant conditions.

FSAR Table 15.2-12, "Main Steam Isolation Valve Closure - Limiting Analysis Results," presents the limiting analysis results for this event. The staff finds that the predicted plant response for the most limiting events satisfies the AOO acceptance criteria because the analysis demonstrates that the limiting MCHFR of 2.40 is above the 95/95 limit of 1.43; the maximum RCS pressure of 2,307 psia remains below 110 percent of the RCS design pressure (2,420 psia); and the maximum peak secondary pressure of 1,523 psia remains below 110 percent of the secondary system design pressure (2,420 psia).

By demonstrating that the AOO acceptance criteria are met for the MSIV closure event, the applicant satisfied the requirements associated with GDC 10, 13, 15, and 26 for this transient.

15.2.4.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.2.4.

15.2.4.6 Conclusion

The staff reviewed the MSIV closure event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. As documented above, for this event, the applicant's analysis shows that the AOO and DSRS acceptance criteria are met. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of an MSIV closure event meet the relevant requirements set forth in GDC 10, 13, 15, and 26 with respect to this event.

15.2.5 Steam Pressure Regulator Failure (Closed)

FSAR Section 15.2.5, "Steam Pressure Regulator Failure (Closed)," states that the NuScale design does not use a steam pressure regulator, and therefore the applicant did not evaluate the steam pressure regulator failure event. The staff examined design information and drawings to confirm that there is no steam pressure regulator in the NuScale design and determined that this event is not applicable to the NuScale design.

15.2.6 Loss of Non-Emergency Alternating Current Power to the Station Auxiliaries

15.2.6.1 Introduction

The staff reviewed FSAR Section 15.2.6, "Loss of Non-Emergency Alternating Current Power to the Station Auxiliaries," to ensure that the event was analyzed appropriately and meets the acceptance criteria outlined in Section 15.2.6.3 of this SER. A loss of AC power event is an AOO that can result from failures in the electrical grid or the onsite AC distribution system and causes rapid primary and secondary pressurization.

15.2.6.2 Summary of Application

The information provided by the applicant in FSAR Section 15.2.6 is summarized below.

Failures in the electrical grid or plant or switchyard equipment, as well as external weather events, may lead to a loss of AC power to the station auxiliaries. A loss of AC power event causes a TT and a loss of pumps in the secondary system, leading to increasing primary temperature and pressure. The reactor trips and SSI and DHRS actuate on the high PZR pressure signal for all limiting cases. Secondary pressure increases until stable DHRS operation is established to transfer decay heat from the RCS to the reactor pool. The applicant concluded that the event meets the DSRS acceptance criteria.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The TS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.2.6.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS being designed with appropriate margin so SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so the RCPB is not breached during normal operations, including AOOs.

- GDC 26, as it relates to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

DSRS Section 15.2.6 lists the following acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

- Fuel cladding integrity must be maintained by keeping the minimum DNBR (for NuScale, MCHFR) above the 95/95 limit based on acceptable correlations (see DSRS Section 4.4).

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

- The positions of RG 1.105 are considered with respect to plant protection system setpoints assumed in this event.

- The most limiting plant system single failure, as defined in 10 CFR Part 50, Appendix A, must be assumed in the analysis and must satisfy the positions of RG 1.53.

- The applicant should analyze this event using an acceptable analytical model. The staff will evaluate any other analytical model proposed by the applicant.

- The parameter values in the analytical model should be suitably conservative.

15.2.6.4 Technical Evaluation

15.2.6.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to model the NPM thermal hydraulic response to the loss of AC power event. The applicant performed subchannel analysis using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. Section 15.0.2 of this SER describes the staff's evaluation of these codes is described in.

15.2.6.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the transient analysis model. For the loss of AC power analysis, the applicant assumed a loss of the low voltage AC power distribution system (ELVS), which powers DC battery chargers, plant motors, heaters, and packaged equipment.

The applicant assumed that both the EDAS and EDNS DC power remains available for this event. The staff audited the applicant's subchannel analysis of loss of non-emergency AC power (ML24211A089). The staff finds assuming DC power available acceptable because loss of DC power would cause a reactor trip or ESF actuation, or both, earlier in the transient than if the DC power were not available.

Analyses of the loss of AC power do not credit operator action or backup diesel generators. The staff also confirmed that the analyses do not credit any action by mitigating plant control systems. The applicant assumed that the MPS performs as designed, with allowances for instrument inaccuracy. TR-122844-P, Revision 0, describes how the applicant's setpoints conform to RG 1.105. Chapter 7 of this SER presents the staff's review of TR-122844-P, Revision 0.

For the limiting RCS pressure and SG cases, the sequence of events is provided in FSAR Table 15.2-13, which illustrates that the loss of AC power causes a TT and loss of power to the PZR heaters, secondary pumps, and CVCS pumps. RCS pressure increases because of the reduction in heat removal. The reactor trips, and SSI and DHRS actuate on the high PZR pressure signal. The RSV is credited to relieve RCS pressure. Secondary pressure increases initially and then declines as stable natural circulation is established in the DHRS.

The applicant identified different initial condition biases as limiting for the RCS pressure, SG pressure and MCHFR cases, as shown in in FSAR Table 15.2-15. The staff reviewed the selected input parameters and initial conditions and notes that the applicant has chosen suitably conservative input as described in DSRS Section 15.2.6, including a 102 percent initial core power level, the maximum time delay to scram with the most reactive rod held out of the core, and the most limiting BOC reactivity feedback.

The event is similar for the MCHFR case in that the reactor trip and SSI and DHRS actuation occur because of the high PZR pressure MPS signal. The applicant provides in Table 15.2-14, "Loss of Non-Emergency Alternating Current Power - Steam Generator Pressure and Minimum Critical Heat Flux Ratio Limiting Case - Sequence of Events," of the FSAR the limiting MCHFR case. The applicant stated that no single failure of an FWIV or MSIV caused more severe consequences for any case because feedwater flow is lost at the start of the transient, and MCHFR occurs before potential valve failure.

The applicant identified different initial condition biases as limiting for the RCS pressure and SG pressure/MCHFR cases. The staff reviewed the selected input parameters and initial conditions and notes that the applicant has chosen suitably conservative input as described in DSRS Section 15.2.6, including a 102 percent initial core power level, the maximum time delay to scram with the most reactive rod held out of the core, and the most limiting BOC reactivity feedback. The staff also audited the applicant's sensitivity studies that investigated the most limiting initial conditions and single failures to confirm that they led to the most limiting results (ML24211A089). The audited material supports the discussions in the FSAR, and the staff finds that the input parameters and initial conditions listed in the FSAR are suitably conservative and result in the most limiting conditions for each of the respective acceptance criteria.

15.2.6.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.2.6 to ensure that they meet the DSRS acceptance criteria. The staff reviewed the sequence of events tables for the loss of AC power event and finds that they are consistent with the event description and assumptions for the protective system actuation and delay times. In addition, the staff reviewed the figures showing the transient progressions and finds that they support the applicant's event description and assertion that the acceptance criteria are met. Furthermore, the figures show that the NPM reaches a stable condition, indicating that the event does not lead to more serious plant conditions.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. For this event, the reported analytical results relative to the MCHFR and RCS pressure figures of merit are the results without the adjusted DHRS modeling, which the staff finds acceptable as MCHFR and RCS pressure are not significantly impacted by this modeling adjustment. The analytical results for secondary pressure figures of merit reported in Table 15.2-16 of the FSAR are the NRELAP5 results obtained after adjusting the DHRS modeling.

FSAR Table 15.2-16, "Loss of Non-Emergency Alternating Current Power - Limiting Analysis

Results," of the FSAR presents the limiting analysis results for this event. The staff finds that the predicted plant response for the most limiting events satisfies the AOO acceptance criteria because the analysis demonstrates that the MCHFR of 2.4 is above the 95/95 limit of 1.43; the maximum RCS pressure of 2,305 psia remains below 110 percent of the RCS design pressure (2,420 psia); and the maximum peak secondary pressure of 1391 psia remains below 110 percent of the secondary system design pressure (2,420 psia). The staff audited (ML24211A089) the applicant's calculation for the Loss of Non-Emergency Alternating Current Power to the Station Auxiliaries event. The staff finds that the applicant performed sensitivity studies on the input parameters of the model following the NRC approved methodology as presented in TR-108601-P-A, Revision 4.

By demonstrating that the AOO acceptance criteria are met for the loss of AC power event, the applicant satisfied the requirements associated with GDC 10, 13, 15, and 26 for this transient.

15.2.6.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.2.6.

15.2.6.6 Conclusion

The staff reviewed the loss of AC power event, including the sequence of events, values of parameters and assumptions used in the analytical model, and predicted consequences of the transient. As documented above, the applicant's analysis shows that the AOO and DSRS acceptance criteria as defined in Table 15.0-2 are met for this event. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of a loss of AC power event meet the relevant requirements set forth in GDC 10, 13, 15, and 26 with respect to this event.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Introduction

The staff reviewed FSAR Section 15.2.7, “Loss of Normal Feedwater Flow,” to ensure that the event was analyzed appropriately and meets the acceptance criteria outlined in Section 15.2.7.3 of this SER. A pump failure, valve malfunction, or LOOP can cause a loss of normal feedwater flow. A loss of normal feedwater flow causes a decrease in the heat removal rate from the RCS resulting in an increase in RCS pressure and temperature.

15.2.7.2 Summary of Application

The information provided by the applicant in FSAR Section 15.2.7 is summarized below.

The applicant stated that a loss of normal feedwater flow could occur from the following scenarios: pump failures, valve malfunctions, or a loss of AC power. A loss of normal feedwater causes a decrease in heat removal in the steam generators resulting in an increase in RCS temperature and pressure which lead to a reactor trip. The applicant stated that the loss of normal feedwater flow event is expected to occur one or more times in the life of the plant, so it is classified as an AOO.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this section of the applicant’s FSAR.

15.2.7.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs.
- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and appropriate controls to maintain these variables and systems within prescribed operating ranges.
- GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.
- GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

DSRS Section 15.2.7 lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

15.2.7.4 Technical Evaluation

The following discusses the staff's technical evaluation of the applicant's analysis of loss of normal feedwater flow.

15.2.7.4.1 Causes

The staff reviewed FSAR Section 15.2.7 to assess the applicant's identification of causes leading to this event. The staff notes, from FSAR Section 15.2.7.1, that a loss of normal feedwater flow could occur as a result of pump failures, valve malfunctions, and loss of AC power. Because of the various causes, the applicant considered and analyzed a range of scenarios to determine the most limiting loss of feedwater (LOFW) events that result in the most severe consequences. For instance, to determine what type of LOFW event resulted in the maximum SG pressure event, the applicant ran a spectrum of cases to assess the sensitivity to how much feedwater flow is lost (perhaps because of a spurious partial valve closure or other LOFW mechanism). The staff notes that a feedwater line rupture can also result in a loss of feedwater flow; however, the staff reviews such an event in Section 15.2.8 of this SER.

Considering this spectrum of cases, the staff notes that the applicant presented various limiting cases in this FSAR section, each dealing with its own acceptance criteria. The limiting cases presented in FSAR Section 15.2.7 are (1) the limiting MCHFR case, (2) and the limiting RCS pressure case and (3) the limiting SG pressure case. Furthermore, the staff notes that the applicant classified this event as an AOO, consistent with the DSRS. The staff finds the applicant's assessment of causes leading to the event acceptable because it considered a range of scenarios that could be experienced in an actual plant due to component failures or malfunctions.

15.2.7.4.1.1 Methodology

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, to analyze the thermal-hydraulic response to the event.

The applicant performed subchannel analyses using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.2.7.4.1.2 Model Assumptions, Input, and Boundary Conditions

The staff reviewed the applicant's modeling assumptions, analysis input, and boundary conditions to assess the adequacy of the transient analysis model. The staff noted that input assumptions varied slightly among the presented limiting LOFW cases because some of these parameters affect different aspects of the transient (e.g., a smaller loss in feedwater flow is more limiting for the peak SG pressure case, whereas a complete loss of feedwater flow is more limiting for the MCHFR case). In any case, the staff reviewed each limiting case to determine if the applicant's proposed parameters were conservatively selected.

For all limiting cases presented as part of the FSAR Section 15.2.7 analysis, the staff confirmed that initial parameters, such as power level, RCS pressure, RCS temperature, RCS flow, PZR level, Doppler reactivity feedback, moderator temperature reactivity feedback, reactor kinetics parameters, and scram characteristics (including assuming a stuck rod) were conservatively selected and applied in the analysis. The staff also confirmed that the applicant considered instrument inaccuracies and that no operator action is credited to mitigate the effects of a loss of normal feedwater flow event.

The staff reviewed the applicant’s single failure assumptions for this event. The staff confirmed through audit (ML24211A089) that no single failures were found to have an adverse impact on the figures of merit for this event.

The staff reviewed the applicant’s assumptions regarding the availability or unavailability of power systems. The staff confirmed that for each limiting case, the applicant’s power availability assumptions were conservatively applied.

The staff confirmed that the applicant’s LOFW analysis-specific assumptions, input, and boundary conditions were selected conservatively.

15.2.7.4.2 Evaluation of Analysis Results

The staff reviewed the results presented in FSAR Section 15.2.7 and Table 15.2-20, “Loss of Feedwater - Limiting Analysis Results,” to determine if they meet the DSRS acceptance criteria. The staff reviewed the transient behavior of several parameters by evaluating plots of the parameters as a function of time. The staff considered the reactor power, reactor and SG pressures, reactor coolant temperature and flow rate, MCHFR, and PZR level and audited additional parameters, such as reactivity.

As part of its review of transient parameters, the staff verified that the sequence of events was reasonable given the automatic actuations of protection systems at their analytical setpoints.

The staff reviewed the applicant’s LOFW case that resulted in a limiting RCS pressure. The staff confirmed that for the worst RCS pressure case, the RCS pressure remained below 110 percent of the design pressure. The staff finds this acceptable because it meets the DSRS acceptance criteria.

The staff reviewed the applicant’s LOFW case that resulted in a limiting SG pressure. The staff confirmed that for the worst SG pressure case, the SG pressure remained below 110 percent of the design pressure. The staff finds this acceptable because it meets the DSRS acceptance criteria.

The staff reviewed the applicant’s LOFW case that resulted in a limiting MCHFR. The staff confirmed that for the worst MCHFR case, the MCHFR remained above the 95/95 DNBR limit.

The staff reviewed the applicant’s analysis in FSAR Section 15.2.7 and confirmed that following a loss of normal feedwater event, stable DHRS cooling can be attained and the reactor can be safely shut down.

FSAR Table 15.2-20 presents the limiting analysis results for this event. The staff finds that the predicted plant response for the most limiting events satisfies the AOO acceptance criteria because the analysis demonstrates that the MCHFR of 2.63 is above the 95/95 limit of 1.43; the maximum RCS pressure of 2,304 psia remains below 110 percent of the RCS design pressure (2,420 psia); and the maximum peak secondary pressure of 1,583 psia remains below 110 percent of the secondary system design pressure (2,420 psia).

15.2.7.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.2.7.

15.2.7.6 Conclusion

The staff concludes that the consequences of a loss of feedwater flow meet the relevant requirements set forth in the GDCs 10, 13, 15, and 26 with respect to this event. As documented above, for this event sequence, the staff determined that the applicant's analysis showed that the AOO and DSRS acceptance criteria are met.

15.2.8 Feedwater System Pipe Breaks Inside and Outside Containment

15.2.8.1 Introduction

The staff reviewed FSAR Section 15.2.8, "Feedwater System Pipe Breaks Inside and Outside Containment," to ensure that the event was analyzed appropriately and meets the acceptance criteria outlined in Section 15.2.8.3 of this SER. A break in feedwater piping inside or outside of containment can cause a decrease in the heat removal rate from the RCS resulting in an increase in RCS temperature and pressure.

15.2.8.2 Summary of Application

The information provided by the applicant in FSAR Section 15.2.8 is summarized below.

An FWLB event is assumed to occur both inside and outside of the CNV. The applicant analyzed a spectrum of FWLB locations and break sizes, with varied core and plant conditions to determine the scenarios with the most severe results.

A FWLB inside the CNV will increase the pressure in the evacuated atmosphere resulting in a loss of containment vacuum and actuating the high containment pressure MPS signal. The high containment pressure MPS signal actuates the RTS, isolates containment, and actuates the SSI, or for smaller breaks, the high pressurizer pressure MPS signal actuates the RTS and SSI.

Feedwater breaks outside of containment will cause a loss of feedwater flow to the steam generators and a heatup and subsequent pressure increase in the RCS resulting in a high pressurizer pressure signal that will actuate the RTS and SSI.

Assuming a loss of AC power will trip the turbine and FW pumps, causing a rapid heatup and pressurization.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this section of the applicant's FSAR.

15.2.8.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 27, as it relates to controlling the rate of reactivity changes to ensure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 28, as it relates to limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other RPV internals to impair significantly the capability to cool the core.

- GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

- GDC 35, as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.

The staff notes that the applicant provided PDC 35 which is functionally identical to GDC 35.

DSRS Section 15.2.8 lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

15.2.8.4 Technical Evaluation

The following sections discuss the staff's technical evaluation of the applicant's FWLB analysis.

15.2.8.4.1 Causes

The staff reviewed FSAR Section 15.2.8 to assess the applicant's identification of causes leading to this event. The staff notes, from FSAR Section 15.2.8.1, that an FWLB in the NuScale plant can occur because of seismic events, thermal stress, or cracking of the feedwater piping. Because of the various causes, the applicant analyzed a range of FWLBs in different locations throughout the system. For instance, a small split crack to a double-ended guillotine rupture of the largest feedwater line was analyzed in locations inside and outside of containment. Because of the variations in event initiation, through a spectrum of analyses, the applicant was able to determine the scenarios producing the most severe results with respect to the DSRS acceptance criteria.

For these reasons, the applicant presented various limiting cases in this FSAR section, each dealing with its own acceptance criteria. The limiting cases presented in FSAR Section 15.2.8 are the limiting MCHFR case, limiting RCS pressure case, limiting SG pressure case, and limiting DHRS function case. Furthermore, the staff notes that the applicant classified this event as an accident consistent with the DSRS. Although the applicant classifies the event as an accident, the FSAR evaluates the event against AOO criteria. The staff finds the applicant's assessment of causes leading to the event acceptable because it considered a spectrum of FWLBs in different locations throughout the system.

15.2.8.4.2 Methodology

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER to analyze the thermal-hydraulic response to the event.

The applicant performed subchannel analyses using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.2.8.4.3 Model Assumptions, Input, and Boundary Conditions

The staff reviewed the applicant's modeling assumptions, analysis input, and boundary conditions to assess the adequacy of the transient analysis model. The staff noted that input assumptions varied slightly among the presented limiting FWLB cases because some of these parameters affect different aspects of the transient. The staff reviewed each limiting case to determine if the applicant's proposed parameters were conservatively chosen.

For all cases presented as part of the FSAR Section 15.2.8 analysis, the staff confirmed that initial parameters, such as power level, fuel temperature, RCS temperature, PZR pressure, PZR level, RCS flow, Doppler reactivity feedback, moderator temperature reactivity feedback, reactor kinetics, and scram characteristics (including the assumption of a stuck rod), were conservatively applied in the analysis. The staff also confirmed that instrument inaccuracies were considered, and that no operator action is credited to mitigate the consequences of an FWLB event.

The staff reviewed the applicant's single-failure assumptions regarding this event. The staff confirmed that the applicant considered and analyzed single failures for each limiting case of the FWLB event.

The staff reviewed the applicant's assumptions regarding break size and location. The staff noted that, as with other model assumptions, the limiting break size and location depended on which figure of merit (e.g., RCS pressure, SG pressure, MCHFR, or DHRS function) was being analyzed. Nevertheless, the staff confirmed through an audit (ML24211A089) that the applicant's assumed limiting break sizes and locations were supported by sensitivity analyses.

The staff reviewed the applicant's assumptions about the availability or unavailability of power systems. The staff confirmed that for each limiting case, the applicant's power assumptions were conservatively determined.

The staff confirmed that the applicant's FWLB analysis-specific assumptions, input, and boundary conditions were selected conservatively.

15.2.8.4.4 Evaluation of Analysis Results

The staff reviewed the results presented in FSAR Section 15.2.8 and Table 15.2-26, "Feedwater Line Break - Limiting Analysis Results," to determine if they meet the DSRS acceptance criteria. The staff reviewed the transient behavior of several parameters by evaluating plots of the parameters as a function of time. The staff considered the reactor power, reactor and SG pressures, core temperatures, RCS and break flow rates, MCHFR, and DHRS heat removal rate. As part of its review of transient parameters, the staff verified that the sequence of events was reasonable given the automatic actuations of protection systems at their analytical setpoints.

The staff reviewed the applicant's FWLB case that resulted in a limiting RCS pressure. The staff confirmed that for the worst RCS pressure case, the RCS pressure remained below 110 percent of the design pressure. The staff finds this acceptable because it meets the DSRS acceptance criteria. Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. For this event, the reported analytical results relative to the MCHFR and RCS pressure figures of merit are the results without the adjusted DHRS modeling, which the staff finds acceptable as MCHFR and RCS pressure are not significantly impacted by this modeling adjustment. The analytical results for secondary pressure figures of merit reported in FSAR Table 15.2-26 are the NRELAP5 results obtained after adjusting the DHRS modeling.

The staff reviewed the applicant's FWLB case that resulted in a limiting SG pressure. The staff confirmed that for the worst SG pressure case, the SG pressure remained below 110 percent of the design pressure. The staff finds this acceptable because this meets the DSRS acceptance criteria.

The staff reviewed the applicant's FWLB case that resulted in a limiting MCHFR. The staff confirmed that for the worst MCHFR case, the MCHFR remained above the 95/95 DNBR limit, as described in the DSRS.

The applicant stated that the radiological analysis of the SLB bounds the radiological consequences for the FWLB. The staff finds this acceptable because the mass release through a feedline break outside containment would be significantly smaller than the mass release through an SLB outside containment. Section 15.0.3 of this SER documents the staff's review of bounding radiological consequence analyses.

The staff confirmed that the DHRS is safety-related. Since the break depressurizes the SG system and its associated DHRS loop, this renders one train of DHRS completely inoperable. In this case, the second train of DHRS automatically actuates and provides adequate heat removal, ensuring coolable core geometry during and after the accident.

FSAR Table 15.2-26 presents the limiting analysis results for this event. The staff finds that the predicted plant response for the most limiting events satisfies the AOO acceptance criteria because the analysis demonstrates that the MCHFR of 2.40 is above the 95/95 limit of 1.43; the maximum RCS pressure of 2,316 psia remains below 110 percent of the RCS design pressure (2,420 psia); and the maximum peak secondary pressure of 1446 psia remains below 110 percent of the secondary system design pressure (2,420 psia).

15.2.8.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.2.8.

15.2.8.6 Conclusion

The staff concludes that the consequences of postulated feedwater line breaks meet the relevant requirements set forth in the GDC 13, 27, 28, 31, and 35 with respect to this event. As documented above, for this event sequence, the staff determined that the applicant's analysis showed that the DSRS acceptance criteria are met.

15.2.9 Inadvertent Operation of Decay Heat Removal System

15.2.9.1 Introduction

An inadvertent operation of the DHRS is an event of anticipated operational occurrence. The is accident can cause a decrease in the heat removal rate from the RCS resulting in an increase in RCS temperature and pressure.

15.2.9.2 Summary of Application

The applicant analyzed a series of limiting inadvertent operation of DHRS events at higher power, which is a heatup event due to the decrease in cooling. Loss of power or an inadvertent control signal to one actuation valve on either DHRS train will open the flow path to the associated DHRS heat exchanger, providing a short circuit flow path for feedwater through the DHRS piping instead of through the steam generator. However, a loss of power also causes a reactor trip and containment isolation, therefore the applicant does not analyze this scenario because reactor trip and containment isolation will terminate the event. A spurious actuation could occur on one or both trains of DHRS. A full actuation of DHRS opens the two actuation valves on each of the two trains and closes the FWIVs and the MSIVs. An inadvertent signal to isolate one or both steam generators by closure of the FWIV and MSIV on the affected train(s) is also evaluated.

The applicant stated that the inadvertent operation of DHRS is expected to occur one or more times in the life of the module, so it is classified as an AOO.

ITAAC: There is no ITAAC item for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this section of the FSAR.

15.2.9.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.

- GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

There is not an SRP section for this decrease in heat removal event; however, DSRS Section 15.2.7 lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

15.2.9.4 Technical Evaluation

The following discusses the staff's technical evaluation of the applicant's analysis of an inadvertent operation of the DHRS.

15.2.9.4.1 Causes

The staff reviewed FSAR Section 15.2.9 to assess the applicant's identification of causes leading to this event. The staff notes, from FSAR Section 15.2.9.1, "Identification of Causes and Event Description," that the limiting cases for an inadvertent operation of the DHRS (IODHRS) occur at full power. The causes analyzed include opening of a single DHRS valve, inadvertent isolation of one SG and actuation of one DHRS train, inadvertent isolation of both SGs and actuation of both DHRS trains, inadvertent isolation of one SG and inadvertent isolation of both SGs.

The applicant presented various limiting cases in the FSAR section, because each case was limiting for one of the several acceptance criteria. The limiting cases presented in FSAR Section 15.2.9, "Sequence of Events and Systems Operation," are the limiting MCHFR case, the limiting RCS pressure case, and the limiting SG pressure case. Furthermore, the staff notes that the applicant classified this event as an AOO since it is expected to occur one or more times during the life of the module. The staff finds the applicant's assessment of causes leading to the event acceptable because it considered a range of scenarios that could be experienced in an actual plant as the result of component failures or malfunctions.

15.2.9.4.2 Methodology

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER to analyze the thermal-hydraulic response to the event.

The applicant performed subchannel analyses using the VIPRE-01 code with the NSP4 CHF correlation to identify the limiting MCHFR. The staff's evaluation of these codes is described in Section 15.0.2 of this SER.

15.2.9.4.3 Model Assumptions, Input, and Boundary Conditions

The staff reviewed the applicant's modeling assumptions, analysis input, and boundary conditions to assess the adequacy of the transient analysis model. The staff reviewed each limiting case to determine if the applicant's proposed parameters were conservatively chosen.

For all cases presented as part of the FSAR Section 15.2.9, analysis, the staff confirmed that initial parameters, such as power level, fuel temperature, RCS temperature, PZR pressure, PZR level, RCS flow, moderator temperature reactivity feedback, reactor kinetics, and scram characteristics (including the assumption of a stuck rod), were conservatively applied in the analysis. The staff also confirmed that the applicant considered instrument inaccuracies and that no operator action is credited to mitigate the consequences of an IODHRS event.

The staff reviewed the applicant's single-failure assumptions for this event. The staff confirmed that the applicant considered single failures for each limiting case of the IODHRS event.

The staff reviewed the applicant's assumptions as to the availability or unavailability of power systems. The staff confirmed that for each limiting case, the applicant's power availability assumptions were conservatively applied.

The staff confirmed that the applicant's IODHRS analysis-specific assumptions, input, and boundary conditions were selected conservatively.

15.2.9.4.4 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.2.9 and Table 15.2-31, "Inadvertent Operation of Decay Heat Removal System - Limiting Analysis Results," to determine if they meet the SAFDL acceptance criteria. The staff reviewed the transient behavior of several parameters by evaluating plots of the parameters as a function of time. The staff considered the reactor power, reactor and SG pressures, reactor coolant temperature and flow rate, MCHFR, and PZR level.

As part of its review of transient parameters, the staff verified that the sequence of events, as presented in Tables 15.2-27 through 15.2-29, were reasonable given the automatic actuations of protection systems at their analytical setpoints.

The staff reviewed the applicant's IODHRS case that resulted in a limiting RCS pressure. The staff confirmed that for the worst RCS pressure case, the RCS pressure remained below 110 percent of the design pressure. The staff finds this acceptable because it meets the DSRS acceptance criteria.

The staff reviewed the applicant's IODHRS case that resulted in a limiting SG pressure. The staff confirmed that for the worst SG pressure case, the SG pressure remained below 110 percent of the design pressure. The staff finds this acceptable because it meets the DSRS acceptance criteria.

The staff reviewed the applicant's IODHRS case that resulted in a limiting MCHFR. The staff confirmed that for the worst MCHFR case, the MCHFR remained above the 95/95 DNBR limit in the DSRS.

The staff reviewed the applicant's analysis presented in FSAR Section 15.2.9 and confirmed that IODHRS events do not result in pressure or temperature transients that exceed the criteria for which the reactor pressure vessel, SG, CNV, or fuel are designed. Therefore, these barriers to the transport of radionuclides to the environment will function as designed.

Table 15.2-31 of the FSAR presents the limiting analysis results for this event. The staff finds that the predicted plant response for the most limiting events satisfies the AOO acceptance criteria because the analysis demonstrates that the MCHFR of 2.40 is above the 95/95 limit of 1.43; the maximum RCS pressure of 2,307 psia remains below 110 percent of the RCS design pressure (2,420 psia); and the maximum peak secondary pressure of 1,438 psia remains below 110 percent of the secondary system design pressure (2,420 psia). The staff audited the calculation packages and finds that values presented in the FSAR are supported by the calculations.

15.2.9.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.2.9.

15.2.9.6 Conclusion

Based on its review of the application, results of the safety analysis, and the acceptance criteria specified in the SRP, the staff concludes that the consequences of inadvertent operation of the DHRS meet the relevant requirements set forth in the GDC 10, 13, 15, and 26 with respect to this event. As documented above, for this event sequence, the staff determined that the applicant's analysis showed that the AOO and DSRS acceptance criteria are met.

15.3 Decrease in Reactor Coolant System Flow Rate

Decrease in RCS flow rate events do not apply to the NPM design because the NPM operates on the principle of natural circulation with no forced cooling.

15.4 Reactivity and Power Distribution Anomalies

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

15.4.1.1 Introduction

The staff reviewed FSAR Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition," to ensure that the event was analyzed appropriately and meets the acceptance criteria outlined in Section 15.4.1.3 of this SER. An uncontrolled CRA withdrawal from a subcritical or low-power startup condition is an AOO that results in a rapid addition of reactivity to the core. This event causes an increase in core power and a decrease in MCHFR.

15.4.1.2 Summary of Application

The applicant provided information in FSAR Section 15.4.1 as summarized below.

An uncontrolled CRA withdrawal from a subcritical or low-power startup condition could be caused by an operator error or a malfunction in the control rod drive system. Withdrawal of the CRA bank causes an unexpected reactivity addition and leads to increases in core power and decreases in MCHFR. The MPS initiates a reactor trip if MPS setpoints are exceeded. The applicant investigated a spectrum of reactivity insertion rates and initial power levels to identify the limiting cases for MCHFR and fuel centerline temperature using NRELAP5 to obtain the NPM time-dependent thermal-hydraulic response and VIPRE-01 to determine the MCHFR and fuel centerline temperature response. The applicant used the statistical subchannel analysis methodology developed in TR-108601-P-A, Revision 4, to determine the limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative MCHFR result. The applicant stated that the limiting cases meet the MCHFR and fuel centerline temperature acceptance criteria. The analyses assume that the control systems and ESFs perform as designed, no operator action is credited.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.4.1.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, which requires that the reactor core and associated coolant, control, and protection systems are designed with appropriate margin so SAFDLs are not exceeded during normal operations, including AOOs.

- 13, which requires availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 20, which requires that the protective systems automatically initiate the operation of the reactivity control system to ensure that SAFDLs are not exceeded as a result of AOOs.

- GDC 25, which requires that the reactor protection system be designed to ensure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

The guidance in SRP Section 15.4.1, “Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition,” lists the following criteria that are acceptable to the staff for demonstrating conformance to the regulatory requirements, as well as review interfaces with other SRP/DSRS sections:

- The thermal margin limits (DNBR (for NuScale, MCHFR)), as specified in SRP Section 4.4, “Thermal and Hydraulic Design,” are met.

- Fuel centerline temperatures as specified in SRP Section 4.2, “Fuel System Design,” do not exceed the melting point.

15.4.1.4 Technical Evaluation

15.4.1.4.1 Evaluation Model

The applicant used the non-LOCA analysis methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to analyze the thermal-hydraulic response to the event. In addition, the applicant used the VIPRE-01 code with the NSP4 CHF correlation to perform the subchannel analysis, which identified the limiting MCHFR and peak fuel centerline temperature. During the staff's audit review, the applicant stated that it revised the analysis for this event. The new analysis uses the maximum fuel centerline temperature rather than linear heat generation rate (ML24346A280) as acceptance criterion. The applicant used the statistical subchannel analysis methodology developed in TR-108601-P-A, Revision 4, to determine the uncertainties associated with the subchannel analyses. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.4.1.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the transient analysis model. The applicant assumed that the entire regulating bank withdraws for this event, which provides the maximum reactivity insertion. MPS signals protecting the NPM for this analysis are high core power (at 25 percent of full power), source range and intermediate range power rate, and high source range count rate. Different initial power levels and reactivity insertion rates result in reactor trips on different MPS signals. Therefore, the applicant analyzed a spectrum of reactivity insertion rates at different initial power levels ranging from 1 watt to 15 percent RTP, which is the upper limit for low-power operation. The applicant analyzed reactivity insertion rates up to 0.0500 dollars per second ($/s), which bounds possible boron dilution scenarios, as well as CRA regulating bank withdrawal at the maximum speed.

The applicant determined that the limiting case for both MCHFR and fuel centerline temperature has an initial power of 38 megawatt and a reactivity insertion rate of 0.00359 $/s. The reactor power rise for the limiting case is terminated by a reactor scram, which is actuated after the high pressurizer pressure MPS limit is reached (the high power rate trip is reached at nearly the same time). The applicant determined the limiting reactivity insertion rate and power level using sensitivity studies.

The staff reviewed other initial parameter values and biases in the FSAR, including initial RCS conditions and reactivity coefficients, to ensure that the applicant selected conservative values for the analysis. The applicant assumed the most positive MTC and least negative DTC, which are conservative because they minimize negative reactivity feedback as moderator and fuel temperatures increase.

The staff confirmed through the audit (ML24211A089) that the subchannel analysis uses limiting axial and radial power shapes in accordance with the subchannel analysis methodology, as stated in FSAR Section 15.4.1. The staff also verified through the audit that appropriate cases from the transient analysis were passed on for subchannel analysis.

According to FSAR Table 15.0-7, the applicant treated the source range count rate trip as an overpower trip that occurs at an analytical limit of 500 kilowatts, which functionally equates count rate to core power.

The applicant did not credit operator action to mitigate the uncontrolled CRA withdrawal from a subcritical or low-power startup condition event. The staff agrees with the applicant's conclusion that no single failure is limiting for this event because a failure of an ex-core detector would have no effect on a symmetric reactivity transient, and a failure of an MSIV or FWIV to close would have no effect on MCHFR since DHRS does not actuate for that event sequence.

The FSAR states that loss of power is not limiting for the event. The staff notes that a loss of normal AC power would trip the feedwater pumps and turbine, but the effect would be negligible because of the low initial power level. Furthermore, the loss of AC power would terminate CRA withdrawal and lead to a less limiting event. Therefore, the staff finds that the applicant's treatment of loss of power for this event is acceptable.

For the reasons stated above, the staff finds that the input parameters, initial conditions, and assumptions associated with the applicant's uncontrolled CRA withdrawal from a subcritical or low power startup condition analysis are suitably conservative and result in the most limiting conditions for the acceptance criteria.

15.4.1.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.4.1 to ensure that they meet the SRP acceptance criteria. The staff reviewed the sequence of events tables that apply to the uncontrolled CRA withdrawal from a subcritical or low-power startup condition event and finds that they are consistent with the event description and assumptions for protective system actuation with delay times. The NRC staff audited the applicant's documentation of RTS delay analytical limits and noted that the applicant models delay in reactor trip from low-power instrumentation **{{**

}}. FSAR Table 14.2-56, "Test # 56 Module Protection System," requires applicants and licensees referencing the US460 SDAA to verify that as-built instrumentation will have response times that are less than or equal to values modeled in the safety analysis. In addition, the staff reviewed the figures showing the transient progression and finds that they support the applicant's event description and assertion that the acceptance criteria are met.

FSAR Table 15.4-3, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition (15.4.1) - Limiting Analysis Results," presents the limiting analysis results for this event. The maximum fuel centerline temperature of 1680 °F is well below the limit of 4791 °F, and the limiting MCHFR of 8.18 remains above the 95/95 limit of 1.43.

The staff finds that the uncontrolled CRA withdrawal from a subcritical or low power startup condition event satisfies the SRP acceptance criteria because the response of the fuel is within the SAFDLs, and thus no fuel damage is anticipated from this event. By demonstrating that the acceptance criteria are met for the most limiting scenario, the applicant satisfied the requirements associated with GDC 10, 13, 20, and 25 for this event.

15.4.1.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.4.1.

15.4.1.6 Conclusion

The staff reviewed the uncontrolled CRA withdrawal from a subcritical or low-power startup condition event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. As documented above, the applicant's analyses show that the SRP acceptance criteria are met for this event. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of an uncontrolled CRA withdrawal from a subcritical or low-power startup condition event meet the relevant requirements set forth in GDC 10, 13, 20, and 25 with respect to this event.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Introduction

The staff reviewed FSAR Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," to ensure that the event was analyzed appropriately and meets the acceptance criteria discussed in Section 15.4.2.3 of this SER. An uncontrolled CRA withdrawal at power is an AOO that causes an unexpected positive reactivity insertion and a corresponding increase in core power. The power increase and resulting RCS temperature and pressure increases lead to a decrease in MCHFR.

15.4.2.2 Summary of Application

The applicant provided information in FSAR Section 15.4.2, summarized below.

The uncontrolled CRA withdrawal at power analysis simulates withdrawal of a regulating CRA bank, which causes an unplanned reactivity addition to the core. Core power increases and because the secondary system lags the primary system response, RCS temperature and pressure increase. Such conditions could challenge SAFDLs. The MPS initiates a reactor trip on high power, high power rate, high PZR pressure, high RCS average temperature, or high RCS hot temperature, depending on the initial conditions and assumptions. The DHRS may be actuated on high RCS hot temperature, high PZR pressure, or high steam pressure to maintain post-trip core cooling.

The applicant analyzed a spectrum of reactivity insertion rates and initial power levels to identify the limiting cases for MCHFR and fuel centerline temperature (as evaluatedLHGR). The applicant used NRELAP5 to obtain the NPM time-dependent thermal-hydraulic response and VIPRE-01 to obtain the MCHFR and LHGR. The applicant stated that the limiting cases meet the SRP acceptance criteria for this event.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.4.2.3 Regulatory Basis

The regulations listed in SER Section 15.4.1.3 are also applicable to FSAR Section 15.4.2. The guidance in SRP Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," lists the same acceptance criteria as SRP Section 15.4.1 for demonstrating conformance with the applicable requirements, as well as review interfaces with other SRP/DSRS sections.

15.4.2.4 Technical Evaluation

15.4.2.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including the NRELAP5 code, to analyze the thermal-hydraulic response to the event. In addition, the applicant used the VIPRE-01 code with the NSP4 CHF correlation and the statistical subchannel methodology (ML24106A160) to perform the subchannel analysis, which identified the limiting MCHFR and LHGR. Section 15.0.2 of this SER describes the staff's evaluation of these codes and methods.

15.4.2.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the transient analysis model. The applicant assumed that the entire regulating bank withdraws for this event, which provides the maximum reactivity insertion. This analysis credits the high core power, high core power rate, high RCS hot temperature, high RCS average temperature, and high PZR pressure MPS reactor trip signals. Different initial conditions and reactivity insertion rates result in reactor trips on different MPS signals. To account for the range of transient progressions and trip scenarios, the applicant analyzed a spectrum of reactivity insertion rates at initial power levels of 15, 50, 75, and 102 percent. The applicant included reactivity insertion rates up to 0.0500 $/s, which bounds possible boron dilution scenarios as well as CRA regulating bank withdrawal at the maximum speed.

The applicant determined that the limiting case for MCHFR has an initial power of 102 percent and a reactivity insertion rate of 0.00224 $/s. This case reaches the high power analytical limit first, and reactor trip is actuated after a 2-second delay. The limiting LHGR case initiates from 75 percent power, with a reactivity insertion rate of 0.00685 $/s. The rapid power increase during the maximum LHGR case causes a high PZR pressure trip.

The staff reviewed other initial parameter values and biases, including initial RCS conditions and reactivity coefficients, to ensure that the applicant selected conservative values for the analysis. The applicant assumed the least negative MTC and DTC, which are conservative because they minimize negative reactivity feedback as moderator and fuel temperatures increase. In addition, the applicant did not model power-dependent insertion limit (PDIL)-imposed restrictions on the total reactivity insertion in order to bound boron dilution scenarios. The staff finds this treatment conservative because it allows for a greater total reactivity insertion than permitted by PDILs for a given power level.

As described in the associated audit documentation, the staff audited the applicant's sensitivity studies that investigated the most limiting initial conditions and reactivity insertion rates to confirm that they led to the most limiting results. The staff also confirmed that the subchannel analysis uses limiting axial and radial power shapes in accordance with the subchannel analysis methodology, as stated in FSAR Section 15.4.2.

The staff confirmed that the applicant treated control systems conservatively in the analysis. If control system operation leads to a less severe plant response, the applicant assumed that the control system is disabled. However, control system operation is allowed if it causes a more severe transient. For example, the applicant allowed PZR spray operation for some MCHFR cases to delay a trip on high PZR pressure. The applicant also evaluated control rod withdrawal at power scenario with letdown turned on. Although the actual operation of the plant does not allow letdown to be set on auto mode, the staff finds analysis of these cases provides additional information on the system performance in response to this event.

The applicant did not credit operator action to mitigate the uncontrolled CRA withdrawal at power event. In addition, the staff agrees with the applicant's assertion that no single failure is limiting for this event because a failure of an ex-core detector would have no effect on a symmetric reactivity transient, and a failure of an MSIV or FWIV to close would occur after the time of limiting MCHFR and maximum power.

The staff examined the treatment of loss of power scenarios through audit and determined that the applicant considered loss of power scenarios to be nonlimiting and did not analyze them, with the exception of EDAS. The NRC staff finds this acceptable because AC power is required to continue CRA bank withdrawal, so loss of normal AC power or loss of normal AC power with loss of EDNS would be equivalent or less limiting than scenarios analyzed. The applicant stated that a loss of EDNS alone would lead to TT and loss of feedwater without immediate insertion of CRAs, and that these consequences would result in an earlier reactor trip and a less limiting event with respect to MCHFR and peak LHGR (ML24346A282). The applicant did not evaluate the loss of EDAS, but EDAS is relied on to remain functional during this event. Section 15.0.0.6.2 of this SER gives the detailed evaluation of EDAS treatment in the Chapter 15 safety analysis.

For the reasons stated above, the staff finds that the input parameters, initial conditions, and assumptions associated with the applicant's uncontrolled CRA withdrawal at power analysis are suitably conservative and result in the most limiting conditions for the acceptance criteria.

15.4.2.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.4.2 to ensure that they meet the SRP acceptance criteria. The staff reviewed the sequence of events tables that apply to the uncontrolled CRA withdrawal at power event and finds that they are consistent with the event description and assumptions regarding protective system actuation and delay times. In addition, the staff reviewed the figures showing the transient progression and finds that they support the applicant's event description and assertion that the acceptance criteria are met. Furthermore, the figures show that the NPM reaches a stable condition, indicating that the events do not lead to more serious plant conditions.

FSAR Table 15.4-7, "Uncontrolled Control Rod Assembly Withdrawal at Power (15.4.2) - Limiting Analysis Results," presents the limiting analysis results for this event. The staff notes that the applicant applies an LHGR limit to the uncontrolled CRA withdrawal at power event to evaluate whether fuel centerline melting occurs or 1 percent cladding strain is exceeded. This is consistent with the methodology described in TR-0915-17564-A, Revision 2, which is approved by the staff and is therefore acceptable. The limiting LHGR of 11.7 kW/ft remains well under the bounding fuel centerline melt limits presented in technical report TR-117605-P, Revision 0, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs", issued December 2022 (ML23304A336 (nonproprietary), ML23304A337 (proprietary)), which is evaluated in Section 4.2 of this SER. The limiting MCHFR of 1.77 remains above the 95/95 limit of 1.43.

The staff finds that the uncontrolled CRA withdrawal at power event satisfies the SRP acceptance criteria because the response of the fuel is within the SAFDLs, and thus no fuel damage is anticipated from this event. By demonstrating that the SRP acceptance criteria are met for the most limiting uncontrolled CRA withdrawal at power event, the applicant satisfied the requirements associated with GDC 10, 13, 20, and 25 for this event.

15.4.2.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.4.2.

15.4.2.6 Conclusion

The staff reviewed the uncontrolled CRA withdrawal at power event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. As documented above, the applicant's analyses show that the SRP acceptance criteria are met for this event. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of an uncontrolled CRA withdrawal at power event meet the relevant requirements set forth in GDC 10, 13, 20, and 25 with respect to this event.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.3.1 Introduction

The staff reviewed FSAR Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)," to ensure that the events were analyzed appropriately and meet the acceptance criteria outlined in Section 15.4.3.3 of this SER. The applicant considered three different control rod misoperation (CRM) events, which are all AOOs, (1) a CRA misalignment (FSAR Section 15.4.3.3), (2) a single CRA withdrawal (FSAR Section 15.4.3.4), and (3) a single or multiple CRA drop (FSAR Section 15.4.3.5). Each of these events has the potential to challenge SAFDLs.

15.4.3.2 Summary of Application

The applicant provided the information in FSAR Section 15.4.3 which is summarized below.

As discussed above, the applicant considered three events for the CRM analysis, each of which could unexpectedly affect core reactivity and power distributions such that SAFDLs could be challenged.

The FSAR discusses two limiting CRA misalignment scenarios. The first scenario assumes a limiting misalignment occurs when the core is operating at steady-state full power with the rods inserted to the PDILs except one rod is left withdrawn. The FSAR states that this scenario is bounded by the single CRA withdrawal. Analysis of static misalignment in this configuration is not presented in the FSAR. The second CRA misalignment event assumed that all CRAs are withdrawn except for one that is misaligned to the 20 percent rated power PDIL with six steps of additional insertion added for rod position uncertainty. The applicant performed steady-state core analyses using SIMULATE5 to identify the limiting CRA misalignment of the second scenario and inform the radial peaking augmentation factor to be input to the subchannel analysis using VIPRE-01. The applicant stated that no transient analysis is needed for this event, as it is a static misalignment in which core power and thermal-hydraulic conditions do not change.

The applicant used NRELAP5 and VIPRE-01 to model the single CRA withdrawal. The applicant credited the high PZR pressure, high power, high power rate, high RCS hot temperature, and high RCS average temperature MPS reactor trip signals for tripping the reactor. The FSAR further states that in CRM events that result in a reactor trip and secondary system isolation, the subsequent actuation of the DHRS is credited with maintaining reactor cooling.

A single CRA withdrawal may occur because of equipment failure or operator error. For this event, the applicant assumed that the regulating bank is inserted to the PDIL plus an additional six steps of insertion for rod position uncertainty, and a single rod withdraws. This adds positive reactivity to the core, increasing power, RCS temperature, and pressure, and causes the power distribution to become asymmetric. The applicant analyzed a spectrum of initial power levels and reactivity insertion rates to identify the limiting cases for MCHFR and LHGR. FSAR Section 15.4.2 analyzes the withdrawal of an entire regulating bank.

A CRA drop can occur because of mechanical or electrical failures and may include a single CRA or an entire group of a regulating or shutdown bank. The applicant analyzed several CRA drop scenarios from different initial power levels to confirm that limiting MCHFR and LHGR are either bounded by steady-state subchannel analysis or by the single CRA withdrawal event.

For each of the events in FSAR Section 15.4.3, the applicant concluded that the limiting cases meet the SRP acceptance criteria.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.4.3.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, which requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin so that SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 13, which requires the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 20, which requires that the protective system automatically initiate the operation of the reactivity control system to ensure that SAFDLs are not exceeded as a result of AOOs.

- GDC 25, which requires that the reactor protection system be designed to ensure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

The guidance in SRP Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)," lists the following acceptance criteria for demonstrating conformance with these requirements, as well as review interfaces with other SRP/DSRS sections:

- The thermal margin limits (DNBR (for NuScale, MCHFR)), as specified in SRP Section 4.4, are met.
- Fuel centerline temperatures, as specified in SRP Section 4.2, do not exceed the melting point.
- Uniform cladding strain as specified in SRP Section 4.2, does not exceed 1 percent.

15.4.3.4 Technical Evaluation

The applicant presents the analyses of a CRA misalignment, single CRA withdrawal, and single or bank CRA drops in FSAR Section 15.4.3. The staff considers this to be a complete scope of CRA misoperation events (excluding the events in FSAR Sections 15.4.1 and 15.4.2) and therefore acceptable.

15.4.3.4.1 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including the NRELAP5 code, to model the thermal-hydraulic response to the single CRA withdrawal and CRA drop events. The applicant also used SIMULATE5 to calculate power distributions as input to the subchannel analyses and the radial peaking augmentation factors for each of the events covered in this review section. The applicant uses SIMULATE5 to calculate the CRA worth and power distributions to confirm that CRA drop is bounded by the single CRA withdrawal event. The applicant used the NRELAP5 and SIMULATE5 results as input to the subchannel analyses using VIPRE-01 and the NSP4 CHF correlation to identify the limiting MCHFR and LHGR. The applicant used the statistical subchannel methodology as defined in TR-108601-P-A, Revision 4, to determine the uncertainties associated with the subchannel analyses using VIPRE-01 code. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.4.3.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the analysis models. The CRM events share some common assumptions. First, the applicant did not credit operator action to mitigate these events. In addition, the analyses assume that plant control systems function as designed, except when they cause less severe consequences for the transient. The staff evaluated the assumptions and finds them to be conservative. The credited MPS reactor trip signals for these events include high power, high power rate, high RCS average temperature, high RCS hot temperature, and high PZR pressure. FSAR Table 7.1-4, "Engineered Safety Feature Actuation System Functions," provides the signals credited for SSI and DHRS actuation. The subsections below discuss event-specific inputs, initial conditions, and assumptions.

15.4.3.4.2.1 Control Rod Assembly Misalignment

FSAR Section 15.4.3 discusses three potential CRA misalignment scenarios. The first occurs at full-power conditions with the rods inserted to the PDIL with one rod left fully withdrawn, which the applicant stated is bounded by a single CRA withdrawal. The staff notes that the single CRA withdrawal analysis simulates essentially the same conditions as CRA misalignment except in a transient simulation. The staff finds this conclusion to be acceptable because this scenario is adequately bounded by single CRA withdrawal event with respect to reactivity insertion and system responses, the reactivity introduced by misalignment will not exceed that introduced by withdrawal of a CRA. The second scenario is that all CRAs are inserted to the PDIL except one is fully inserted. In a letter dated December 11, 2024 ((ML2434A131/ML24346A284 (nonproprietary), ML24346A285 (proprietary)), the applicant stated that this scenario is not credible because core power distribution mapping performed at low power reactor hold points would indicate the PDIL violation due to severe power peaking distortion and operating procedures would prohibit power ascension beyond the hold point. The final scenario, and the one the applicant analyzed, is a case at full power with all CRAs fully withdrawn except one regulating CRA misaligned in at the 20 percent PDIL, plus six steps of rod position uncertainty.

The applicant stated in the FSAR that it considered different power levels, axial offsets, misaligned CRAs, and times in cycle in its SIMULATE5 calculations to find the limiting CRA misalignment. The staff audited the applicant's calculation and confirmed that the applicant identified the limiting case considering a comprehensive range of conditions (ML24211A089). In addition, as part of the audit, the staff verified that the initial RCS parameter values and biases used in the subchannel analysis are conservative with respect to MCHFR. In addition, the staff confirmed the bounding nature of the radial peaking augmentation factor input to the subchannel analysis.

For the reasons stated above, the staff finds that the input parameters and assumptions associated with the applicant's CRA misalignment analysis are suitably conservative and represent the most limiting conditions for the acceptance criteria.

15.4.3.4.2.2 Single Control Rod Assembly Withdrawal

The applicant analyzed a spectrum of reactivity insertion rates from 0.0001 $/s to 0.0110 $/s (ML24215A164**)** to capture the maximum reactivity insertion caused by a single rod withdrawal at initial power levels of 20, 35, 45, 55, 65, 75, 85, 100 and 102 percent of the nominal power. The applicant determined that the case that results in the limiting MCHFR has an initial power level of 85 percent rated power and a reactivity insertion rate of 0.005 $/s. Reactor trip is initiated by the high core average temperature signal, and the high steamline pressure signal subsequently induces SSI and DHRS actuation. The case that results in the limiting LHGR has an initial power level of 45 percent rated power and a reactivity insertion rate of 0.0101 $/s. It results in a reactor trip and SSI and DHRS actuation on the high PZR pressure signal.

The staff reviewed other initial parameter values and biases, including initial RCS conditions and reactivity coefficients to ensure that the applicant selected conservative values for the analysis. The applicant used the least negative DTC and MTC corresponding to BOC conditions, which is conservative since they minimize negative reactivity feedback as fuel and moderator temperatures increase. The applicant analyzed power levels between 20 percent and 102 percent of rated power. The staff audited the analyses for CRA withdrawal event and finds that the applicant demonstrated that the single CRA withdrawal events initiated below 20 percent rated power are bounded by analyzed power levels even though the moderator temperature coefficient has a positive value below 20 percent power (ML24215A166). The staff audited the applicant's sensitivity studies that investigated the most limiting initial conditions and reactivity insertion rates to confirm that they led to the most limiting results and to confirm that those values are implemented in the FSAR analysis. The staff also audited the subchannel analysis calculation for the single CRA withdrawal event and confirmed that the applicant used suitably conservative axial and radial power shapes and the associated peaking factors as input to the subchannel analysis, supporting the statement in the FSAR that limiting radial and axial power shapes were used.

The FSAR states that the limiting single failure for the single CRA withdrawal is a failure of an ex-core detector. The analysis assumes that the remaining detectors see the lowest possible flux resulting from the power asymmetry, which is conservative because it delays MPS actuation on power-related trips. The staff notes that a single failure of an MSIV or FWIV to close would occur after the MCHFR has occurred and is therefore not a limiting single failure.

The applicant stated that a loss of power is not limiting for MCHFR. The staff agrees, noting that a loss of AC power at the beginning of the event would terminate the transient earlier than would otherwise occur for both the MCHFR case and the LHGR case. A loss of power at the time of reactor trip would not affect the drop in core power, which drives the margin to acceptance criteria at that point in the transient. The basis for staff findings that loss of DC power scenarios are non-limiting for the uncontrolled CRA bank withdrawal at power in SER Section 15.4.2.4.2 is also applicable to this event. Therefore, the staff finds that the applicant applied conservative loss of power assumptions.

For the reasons stated above, the staff finds that the input parameters, initial conditions, and assumptions associated with the applicant's single CRA withdrawal analysis are suitably conservative and result in the most limiting conditions for the acceptance criteria.

15.4.3.4.2.3 Control Rod Assembly Drop

The applicant investigated several CRA drop scenarios to determine if any scenarios are not bounded by the single CRA withdrawal event. The applicant examined different initial power levels, times in life, axial offsets, flow rates, initial CRA positions, and dropped CRAs or dropped groups. The evaluation concluded that most CRA drop cases reach the power rate trip limit within 1.5 seconds. In accordance with the non-LOCA methodology discussed in Section 15.0.2, MCHFR and LHGR from these cases are less limiting than steady-state subchannel analysis as maximum local power decreases throughout the transient. For other cases, CRA worth, radial peaking, and quadrant tilt were compared to values used in the single CRA withdrawal analysis. The evaluation concluded that all cases that do not reach the high power rate trip setpoint are bounded by the single CRA withdrawal analysis.

The staff reviewed the initial parameter values and biases, including initial RCS conditions and reactivity coefficients assumed in the rod drop screening analysis. The staff finds that the selected values, particularly selected reactivity coefficients and assumptions regarding initial fuel and coolant temperature changes, are selected to minimize dropped rod worth which conservatively increases the likelihood that a dropped rod case will not initiate power rate trip and pass on to the next screening evaluation.

The staff audited the applicant's sensitivity studies (ML24211A089) that investigated the most limiting rod drop scenarios and initial conditions to confirm that all rod drop cases that do not result in an initial rate trip are bounded by single CRA withdrawal analysis limits.

For the reasons stated above, the staff finds that the input parameters, initial conditions, and assumptions associated with the applicant's CRA drop analysis are suitably conservative and result in the most limiting conditions for the screening criteria.

15.4.3.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.4.3 to ensure that they meet the SRP acceptance criteria. The staff reviewed the sequence of events table for the CRA misoperation transients and finds that it is consistent with the single CRA withdrawal event description and assumptions regarding protective system actuation and delay times. In addition, the staff reviewed the figures showing the transient progressions and finds that they support the applicant's event description and assertion that the acceptance criteria are met. Furthermore, the figures show that the NPM reaches a stable condition, indicating that the events do not lead to more serious plant conditions.

FSAR Table 15.4-11 presents the limiting analysis results for the CRA misoperation events. Similar to FSAR Section 15.4.2, the applicant applied an LHGR limit to the CRA misoperation events to evaluate whether fuel centerline melting and unacceptable cladding strain would occur. The limiting LHGR of 14.0 kilowatts per foot occurs for the single CRA withdrawal and remains under the bounding fuel centerline melt and transient cladding strain limits presented in TR-117605-P, Revision 0, which is evaluated in Section 4.2 of this SER and is incorporated by reference in FSAR Table 1.6-2. The static CRA misalignment produces the limiting MCHFR of 1.79, which remains above the 95/95 limit of 1.43.

The staff finds that the CRA misoperation events satisfy the SRP acceptance criteria because the response of the fuel is within the SAFDLs, and thus no fuel damage is anticipated from this event. By demonstrating that the SRP acceptance criteria are met for the most limiting CRA misoperation events, the applicant satisfied the requirements associated with GDC 10, 13, 20, and 25 for these events.

15.4.3.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.4.3.

15.4.3.6 Conclusion

The staff reviewed the various CRM events, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transients. As documented above, the applicant's analyses show that the SRP acceptance criteria are met for these events. Therefore, the staff concludes that the applicant's analyses are acceptable, and the consequences of a CRM event meet the relevant requirements set forth in GDC 10, 13, 20, and 25 with respect to these events.

15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature

A startup of an inactive loop or recirculation loop at an incorrect temperature is not applicable to the NPM-20 design because the NPM-20 does not have multiple reactor coolant loops.

15.4.5 Flow Controller Malfunction Causing an Increase in Core Flow Rate (Boiling Water Reactor)

A flow controller malfunction causing an increase in core flow rate is not applicable to the NPM-20 design because the NPM-20 design does not have a flow controller that could increase recirculation flow. The NPM-20 operates on the principle of natural circulation with no forced cooling.

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System

15.4.6.1 Introduction

A malfunction in the CVCS or an operator error could result in an inadvertent dilution of boron in the RCS. The inadvertent dilution causes a positive reactivity addition to the core. Section 15.0.5 of this SER documents the staff's review of the potential for boron redistribution following ECCS actuation.

15.4.6.2 Summary of Application

The applicant described the event in FSAR Section 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System".

In the NuScale design, a boric acid blend system allows the operator to adjust the boron concentration in the reactor using coolant makeup water during normal operation. Boron dilution requires operator action, and automatic letdown is not provided while DWS is not isolated. The dilution is also governed by administrative controls with procedures that establish the limits on the rate and duration of dilution. An unintended decrease in boron concentration increases the reactivity of the core and decreases the shutdown margin. An inadvertent decrease in boron concentration in the RCS is expected to occur one or more times during the lifetime of the reactor and is classified as an AOO.

ITAAC: There are no ITAAC for this scenario of AOO accident.

Technical Specifications: The following GTS are applicable to this area of review:

- LCO 3.1.9, "Boron Dilution Control."
- The GTS listed in Section 15.0.0 of this SER.

Technical Reports: There are no technical reports associated with this section of the FSAR.

15.4.6.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS design with appropriate safety margin so that SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so that the pressure boundary is not breached during normal operations, including AOOs.

- GDC 26, as it relates to the control of reactivity changes so that SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate safety margin for accidental boron dilution.

The guidance in SRP Section 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)," lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

15.4.6.4 Technical Evaluation

The CVCS can adjust boron concentration and allows operators to modify boron concentration of the RCS during normal operation. Failures in the CVCS or operator error can result in unborated water being inadvertently introduced into the RCS. This event is considered to occur one or more times during the lifetime of the reactor, and is therefore, classified as an AOO. The limiting CVCS dilution source considered in the safety analysis is the demineralized water system (DWS) supply. Each CVCS makeup pump is assumed to provide 25 gallons per minute (gpm) of demineralized water, which includes 5 gpm to account for uncertainties. Administrative controls prohibit operation with two pumps early in the cycle when critical boron concentrations are high. A makeup flow rate of 50 gpm (two pumps) is considered when critical boron concentration is below the limit specified in FSAR Table 15.4-12, "Bounding Critical Boron Concentrations and Boron Reactivity Coefficients."

The boron dilution analysis, described in FSAR Section 15.4.6, determines the range of possible reactivity insertion rates resulting from an inadvertent boron dilution and evaluates whether boron dilution could lead to a complete loss of TS shutdown margin before detection and isolation of the dilution source.

15.4.6.4.1 Evaluation Model

FSAR Section 15.4.6.3.1, "Evaluation Models," states that two calculation techniques are used to analyze the boron dilution event for the NuScale module. One method assumed unborated water injected into the RCS mixes instantaneously with the effective system coolant volume. The other method evaluated the boron dilution event by using a wave front model to maximize the amount of reactivity as the diluted slug of water sweeps through the core and does not assume any axial mixing. This diluted slug is assumed to move through the riser, SGs, downcomer, and the reactor core. As stated in Section 7.2.16, "Inadvertent Increase in Boron Concentration," of TR-0516-49416-P, Revision 4 described in Section 15.0.2 of this SER, this model does not assume any axial blending to ensure that the reactivity insertion rate is maximized. Both the perfect mixing and wave front models are used to establish a range of reactivity insertion rates for Mode 1 operations to ensure that boron dilution is bounded by CRA withdrawal for evaluation of SAFDLs. Cases performed to evaluate shutdown margin in Mode 1 above hot zero power use the perfect mixing model. For all other modes where limited mixing exists, the wave front model is used. These models were applied in evaluation of the NPM-160 evaluation of the inadvertent increase in boron concentration in the RCS transient. The staff reviewed these two methods for calculations of the boron dilution event for applicability to the NPM-20 design and finds that the methods remain valid because, in comparison with the NPM-160, the changes, (such as increased number of riser holes or power density), in the system design would not change the validity of these models with respect to the progression and consequences of this event. The NRC staff finds that the models are acceptable methods of ensuring that bounding reactivity insertion rates are analyzed in the inadvertent decrease in boron dilution transient for the NPM-20 design.

In addition to these two mixing models, the calculation uses two different methods to assess the reactivity insertion before DWS isolation occurs in Mode 1. The first method assumed that letdown is equal to makeup flow, such that DWS isolation occurs due to reactor trip signals credited in the uncontrolled bank withdrawal transients evaluated in FSAR 15.4.1 and 15.4.2 (i.e. high power, high power rate, high RCS hot temperature, high RCS average temperature, high pressurizer pressure, high SR count rate, or log power rate). The staff audited the underlying calculation and noted that these signals tend to isolate DWS faster in the earlier portion of the cycle, so the total reactivity addition tends to be smaller. The second method assumed that letdown is disabled, and the high pressurizer level trip is credited to isolate DWS. DWS isolation time from this signal is relatively constant throughout the cycle, depending on the dilution flow rate, but reactivity addition is larger at earlier points in the cycle. The staff finds either of these methods acceptable for evaluating a specific point in the cycle because they each consider a subset of available DWS isolation signals.

The second method is applied assuming that the initial boron concentration is 600 ppm, as described in Table 15.4-12. The NRC staff understands that the applicant refers to this as an "EOC" evaluation even though the initial boron concentration is consistent with mid-cycle conditions because applying a mid-cycle boron concentration with this method will bound events occurring at lower initial boron concentrations, i.e. later in the cycle. The first method is also applied at this mid-cycle point and at BOC. The NRC staff finds that applying these methods with time-in-cycle conditions described above provides reasonable assurance that the limiting time-in-cycle for Mode 1 dilution is identified. Additionally, Mode 3 calculations also model disabled letdown flow.

15.4.6.4.2 Input Parameters, Initial Conditions, and Assumptions

For the purposes of the analysis, the CRAs are not assumed to mitigate reactivity changes, in that each of the two regulating bank groups is assumed to be at its respective PDIL so that the rods do not insert automatically as a result of the reactivity addition of an RCS boron dilution. The limiting shutdown margin reactivity credited in the analysis is 503 pcm for Mode 1, 7,527 pcm for Modes 2 and 3, and 11,112 pcm for Modes 4 and 5. NRC staff audited the applicant's calculation and noted that shutdown margin reactivity credited in Mode 1 is dependent on the initial power level.

FSAR Section 15.4.6.2, "Sequence of Events and Systems Operation," states that the loss of AC power is considered in Mode 5 as it results in the loss of the pool cooling and recirculation system. In the absence of forced circulation, NuScale assumed the dilution water is evenly mixed through a reduced fraction of the pool volume. It also states that loss of power scenarios are non-limiting in Mode 1. In Modes 2 and 3, loss of AC power at the time of TT is not considered as the reactor is subcritical and would not produce a grid disturbance following a TT. The staff finds this treatment of loss of power scenarios to be acceptable as loss of AC or DC power would stop CVCS recirculation and injection pumps and/or close demineralized water isolation valves and both terminate the boron dilution process. FSAR Section 15.4.6.2 also states that there are no single failures that could occur that result in a more severe outcome for the limiting cases.

The analysis does not credit operator actions to terminate the event. Instead, the CVCS is designed with automated features that limit the amount and rate of reactivity increase caused by an inadvertent boron dilution event. To mitigate inadvertent dilution events, the CVCS incorporates two redundant safety-related demineralized supply isolation valves. The isolation valves automatically close upon any of the following MPS signals:

- RTS actuation
- high subcritical multiplication
- low RCS flow

GTS 3.1.9 and associated bases limit demineralized flowrate and pump operation based on the RCS boron concentration. FSAR Section 15.4.6.2 stated that operation with two makeup pumps is prohibited when boron concentration is above the limit provided in Table 15.4-12. NuScale evaluated total reactivity insertion based on reactivity insertion rates that bound two pump operation with initial boron concentration at the FSAR Table 15.4-12 limit. The FSAR stated that the CVCS makeup pump flow rate is designed to provide 20 gpm and the analysis assumes 25 gpm to account for uncertainty. The staff finds this acceptable because the TS appropriately include the limits and controls for the CVCS and demineralization system consistent with the assumption in the analysis regarding inadvertent decrease in boron concentration.

FSAR Section 15.4.6.3.2 states that maximum critical boron concentrations are assumed because the rate of change of concentration and associated reactivity is greater for an initial higher concentration. It also states that the Mode 5 calculation uses the minimum initial pool boron concentration to minimize initial boron mass. The staff notes that this also minimizes final boron mass. GTS 3.5.3 limits the minimum pool boron concentration to ensure that the amount of initial shutdown margin assumed in the boron dilution calculations is maintained. The NRC staff finds the use of the minimum initial boron concentration permitted by GTS 3.5.3 with the minimum initial shutdown margin ensured by GTS 3.5.3 acceptable.

The staff reviewed the safety analysis to verify that a boron dilution event has been analyzed for all plant conditions, such as refueling, startup, power operation, hot standby, hot shutdown, and cold shutdown. FSAR Section 15.4.6.3.3 states that Mode 2 analysis is not provided because the results of the Mode 3 scenario are bounding, as Mode 3 has a more negative boron worth and a larger dilution volume than Mode 2. NRC staff audited the calculation and notes that the reason given for a larger dilution volume in Mode 3 is that ECCS valves may be open in this mode. The applicant accounted for this change by increasing the potential dilution volume while keeping the initial RCS volume constant. The staff finds that Mode 3 boron dilution bounds

Mode 2 when these analytical assumptions are made, when initial shutdown margin is the same in each mode (as described in FSAR Section 15.4.6.3.2), and when boron worth is more negative in Mode 3. FSAR Section 15.4.6.3.3 also states that Mode 1 results are presented for hot full power scenarios, but that shutdown margin remaining at DWS isolation bounds partial-power and hot zero power scenarios. The staff audited the boron dilution calculations and confirmed that analysis was performed for partial-power and hot zero power scenarios, and the results confirm the FSAR description.

The staff also verified during the audit of the boron dilution calculations (ML24211A089) that the input parameters are consistent with the NPM-20 design and that the assumptions are conservative for the transient events identified for the inadvertent boron dilution in the reactor coolant.

Mode 1—Hot Full Power

During a boron dilution transient at HFP, reactor power will increase, and RCS temperature and pressure will increase until the reactor trips on high power, high power rate, high PZR pressure, high RCS riser temperature. If letdown flow is disabled, level will increase and the reactor may trip on high pressurizer level. The calculations performed by the applicant for this mode of operation use the perfect mixing model (FSAR Section 15.4.6, Equation 15.4-2). The applicant determined that the reactivity insertion rates in FSAR Table 15.4-13, "Mode 1, Hot Full Power Results, Beginning of Cycle," are bounded by the range of the reactivity insertion rates that are evaluated in the uncontrolled CRA withdrawal analysis presented in FSAR Sections 15.4.1 and 15.4.2. Mathematical models used by the applicant are discussed further in the following sections related to other modes of operation. For hot full power, the applicant showed in FSAR Table 15.4-14, "Mode 1, Hot Zero Power Results, Beginning of Cycle," that after the DWS isolation the remaining shutdown margin is 358 pcm at BOC, 47 pcm at the analyzed transition between BOC and EOC.

Mode 1—Hot Zero Power

During a boron dilution transient at HZP, reactor power will increase; however, RCS temperature, pressure, and level remain relatively constant for rapid boron dilution scenarios when letdown is modeled, as reactor trip occurs quickly on either the high rate or high reactor power setpoint before RCS conditions can degrade. When letdown is disabled, RCS level will not remain constant during the transient. The reactor trip signals that protect the reactor against boron dilution at this power level include high count rate, high power, high startup rate, and high pressurizer level. The staff audited the calculations supporting this event (ML24211A089) and confirmed that the wave front model is used at this power level when letdown is enabled. The staff confirmed that these calculations evaluate the reactivity change from a complete passage of a wavefront through the core, consistent with methodology description in the FSAR and in TR-0516-49416-P, Revision 4. In the audited calculations, DWS isolation on high pressurizer level also preserved shutdown margin when letdown was disabled.

Mode 2 (Hot Shutdown) and Mode 3 (Safe Shutdown)

Mode 3 is modeled with letdown disabled, and the amount of unborated water added by DWS is assessed based on the volume required to increase the pressurizer level from the biased-low initial pressurizer level in mode 3 (i.e., the minimum pressurizer level when reactor power is below 15 percent RTP as indicated in FSAR Table 15.0-6) to the high pressurizer level trip setpoint. NRC staff considers this acceptable because automatic letdown is prohibited in Mode 3 when the DWS is not isolated, so addition of unborated water to the RCS will result in a gradual increase in pressurizer level. NuScale evaluated that Mode 2 is bounded by Mode 3 as discussed above. The calculations performed by the applicant determined that a shutdown margin of 937 pcm remained after the DWS isolation. FSAR Table 15.4-16, “Mode 3 Results,” shows the calculated results.

Mode 4 (Transition) and Mode 5 (Refueling)

During Mode 4, a boron dilution event is precluded because the CVCS is disconnected and isolated from the RCS. The applicant also analyzed the potential to dilute the RCS during Mode 5 refueling operations. FSAR Table 15.4-17, “Mode 5 Results, Limiting Power Unavailable Scenario,” lists internal flooding sources and volumes that have the large potential for dilution. The applicant concluded that a total dilution volume of 967,700 (ML24062A009) gallons would be required to lose shutdown margin; therefore, reactor pool flooding as a result of pipe breaks and potential flooding sources is nonlimiting whereas the total volume of the largest internal flooding source is 600,000 gallons. Hence, boron dilution caused by flooding of the RCS in Mode 5 can be accommodated by the initial reactivity condition of k_{eff} of 0.90 or less.

15.4.6.4.3 Evaluation of Analysis Results

FSAR Tables 15.4-12 through 15.4-17 present the results of the applicant’s analysis for the modes of operation. The results of the analysis demonstrate that for Modes 1 through 4, the MPS ensures that the CVCS dilution source is isolated before shutdown margin is lost without the need for operator action. Additionally, the maximum reactivity insertion rate during HFP is bounded by the range of reactivity insertion rates evaluated for uncontrolled CRA withdrawal reviewed in Section 15.4.2 of this SER, indicating that challenges to SAFDLs during early phases of the transient are bounded by the uncontrolled CRA withdrawal event. Likewise, the reactivity insertion rates from a dilution at HZP are bounded by the analysis performed for uncontrolled CRA withdrawal from a subcritical or low-power startup condition (evaluated in Section 15.4.1 of this SER).

The limiting boron dilution event for the potential loss of shutdown margin occurs in Mode 1 in the middle of the cycle, when initial MTC has increased to such that reactor trip on reactor power and thermal-hydraulic signals is delayed, but initial boron concentration has not yet fallen such that reactor trip on high pressurizer level prevents an appreciable change in boron concentration. The results indicate that the high pressurizer level trip will actuate first such that 47 pcm of shutdown margin remains. Therefore, the results show that automatic isolation of the DWS valves terminates the boron dilution before shutdown margin is lost. The staff determined that the analysis meets the guidance in SRP Section 15.4.6 with respect to subcriticality.

15.4.6.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.4.6.

15.4.6.6 Conclusion

The applicant has demonstrated the MPS will isolate the DWS and terminate a dilution event before the loss of shutdown margin occurs for Modes 1 through 3. For Mode 4 condition, boron dilution is not applicable because the RCS is disconnected from CVCS and remains sealed. Further, there is not a sufficient flooding source to dilute the reactor pool during refueling operations that could result in the loss of shutdown margin. As shutdown margin is maintained and the event is bounded by uncontrolled CRA bank withdrawal transients evaluated in FSAR Sections 15.4.1 and 15.4.2, the reactor coolant boundary pressure remains below 110 percent of the design value, and fuel cladding integrity is maintained as the minimum DNBR remains above the limit. As documented above, the applicant's analyses show that the AOO and SRP acceptance criteria are met. Therefore, the staff concludes that analysis for the decrease in the reactor coolant boron concentration event is acceptable and meets GDC 10, 13, 15, and 26 requirements.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Introduction

An inadvertent loading and operation of a fuel assembly in an improper position is an IE that can result in reduced CHFR, which may challenge SAFDLs. The staff reviewed FSAR, Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," to ensure that this event was analyzed appropriately and does not result in unacceptable consequences.

15.4.7.2 Summary of Application

The applicant discussed the inadvertent loading and operation of a fuel assembly in an improper position event in FSAR Section 15.4.7 for the NPM-20 reactor design. The applicant stated in the FSAR that an inadvertent loading and operation of a fuel assembly in an improper position is an IE that could affect power distribution and power peaking of the reactor core. If undetected, such an event could lead to reduced CHFR and reduced margin to fuel centerline melt.

An inadvertent loading and operation of a fuel assembly in an improper position is not expected to occur during the lifetime of the reactor because of fuel loading controls and procedures. The in-core instrumentation is expected to detect all fuel misloads that result in power shape deviations greater than the detection thresholds, but not all misloads are detectable.

The applicant analyzed a spectrum of fuel misload configurations, including shuffle misloads and 180-degree rotational misloads, using SIMULATE5 to compute core power distributions, identify the limiting undetectable fuel misload, and inform the radial power peaking augmentation factor used as input to the subchannel analysis. The applicant performed the subchannel analysis using VIPRE-01 to obtain MCHFR and fuel centerline temperatures. The applicant concluded that all acceptance criteria in SRP Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," are met and that no fuel damage is expected.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS applicable to this area of review are listed in Section 15.0.0 of this SER.

15.4.7.3 Regulatory Basis

The following NRC regulation contains the relevant requirements for this review:

- GDC 13, as it relates to providing instrumentation to monitor variables over anticipated ranges for normal operations, AOOs, and accident conditions.

The guidance in SRP Section 15.4.7 lists the following acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections:

- Plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel loading errors after fueling operations.

- If the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, offsite consequences should be a small fraction of the 10 CFR Part 100 criteria. For the purpose of a standard design review, the staff applies the 10 CFR 52.137(a)(2)(iv)(A) and (B) criteria. For the purpose of a design certification review, the staff applies the 10 CFR 52.137(a)(2)(iv)(A) and (B) criteria. A small fraction is interpreted to be less than 10 percent of the 10 CFR 52.137(a)(2)(iv)(A) and (B) reference values.

15.4.7.4 Technical Evaluation

The applicant classified the inadvertent loading and operation of a fuel assembly in an improper position as an IE considering existing fuel handling controls and procedures. While operating procedures that apply subsequent to the initial fuel load and startup are deferred to the COL stage, as described in FSAR Chapter 13, “Conduct of Operations,” the staff identified two initial startup tests in FSAR Chapter 14, related to this review section. The acceptance criterion for the Initial Fuel Load Test (Test # 69) states that each fuel assembly and control component is installed in the location specified by the design of the initial reactor core, which provides reasonable assurance that a misload will be prevented for the initial plant startup. The Core Power Distribution Map Test (Test # 85) specifies taking core power maps at 25, 50, 75, and 100 percent power to verify that the power distribution is consistent with design predictions and TS limits.

In addition, FSAR Section 4.3.2.2.8, “Testing,” describes startup physics testing and states, in part, that power distributions must be confirmed for each newly loaded core. This confirmation is achieved by comparing measured and predicted core neutron flux at low, intermediate, and higher power levels. Test # 85 for the first cycle and startup physics testing for subsequent cycles provide reasonable assurance that any fuel-loading errors severe enough to exceed the in-core instrumentation detection threshold will be detected during plant startup.

GTS LCO 3.2.1, which applies during plant operation at and above 20 percent RTP, limits the enthalpy rise hot channel factor ($F_{\Delta H}$) to the value specified in the COLR. The value of $F_{\Delta H}$ is computed continuously based on measurements from the in-core instrumentation with operator notification on unexpected results (in accordance with B 3.2.1), and it must be verified after each refueling and in accordance with the surveillance frequency control program. FSAR, Table 16.1-1, “Surveillance Frequency Control Program Base Frequencies,” provides a base frequency of 31 effective full power days. This LCO and associated surveillance frequency provide reasonable assurance that the actual $F_{\Delta H}$ will remain within the limiting value assumed in the safety analyses.

However, not all fuel-loading errors are detectable. In particular, the in-core instrumentation will detect all fuel assemblies that are 20 percent above or 20 percent below their predicted power.

The NPM-20 design uses self-powered neutron detector (SPND) to measure neutron flux in the core. The applicant assumed that the in-core instrumentation will detect all fuel assemblies that are 20 percent above or 20 percent below their predicted power. However, instead of providing specification for the SPNDs, the FSAR provides a performance requirement to ensure that the SPNDs to be used must be able to detect power distribution with 20 percent of the predicted.

The consequences of a fuel loading error vary based on the reactivity of the misloaded fuel. Interchanging fuel assemblies with large reactivity differences or incorrect orientation (incorrect rotation) would result in higher power peaking and lower MCHFR. The power peaking may also occur in the MOC for fuel assemblies loaded with burnable poison. Therefore, the applicant considered a spectrum of fuel assembly misloads to identify the limiting misload that is undetectable by the in-core instrumentation.

The following paragraphs.provide the staff's review of the applicant's EM, input parameters, and initial conditions, the assumptions for the SPND sensitivity, and analysis results

15.4.7.4.1 Evaluation Model

The applicant performed neutronics analysis with the three-dimensional, steady-state solver, SIMULATE5, to identify the limiting undetectable fuel misload and obtain the associated radial peaking augmentation factor. As discussed in SER Section 4.3, the staff finds the applicant's use of SIMULATE5 for neutronic analysis acceptable.

The applicant performed the subchannel analysis using VIPRE-01 with the NSP4 CHF correlation and a bounding radial peaking augmentation factor as input. As discussed in TR-0915-17564-A, Revision 2, cycle-specific nuclear analyses will confirm that the analyzed bounding radial peaking augmentation factor is not violated. Section 15.0.2 of this SER further discusses the VIPRE-01 code. The staff audited the applicant's calculation (ML24211A089), and finds that the applicant used the methodology presented in TR-108601-P-A, Revision 4, to determine the uncertainties of the results produced by subchannel analyses using the methodology prescribed in TR-108601-P-A, Revision 4.

Because misload of fuel is not a transient event, the staff determined that a transient analysis is not necessary.

15.4.7.4.2 Input Parameters, Initial Conditions, and Assumptions

The applicant considered a spectrum of 231 potential shuffle misloads (i.e., swapping two fuel assemblies), including quarter-core, half-core, and cross-core configurations. The applicant also examined rotational misloads by rotating fuel assemblies by 180 degrees, even though fuel alignment features would preclude such rotation. The staff finds this analyzed spectrum of fuel misloads acceptable because it adequately covers the possible misloading scenarios.

The applicant referred to TR-0915-17564-P-A, Revision 2, for other key inputs and assumptions used in the subchannel analysis and TR-108601-P-A, Revision 4, for determining the uncertainty of the subchannel analyses. The staff has approved both TR-0915-17564-P-A, Revision 2 and TR-108601-P-A, Revision 4 for the NPM-20 design.

The staff confirmed through an audit (ML24211A089), that the applicant used biases consistent with FSAR Table 15.0-6 in the subchannel analysis. On these bases, the staff finds that the applicant used conservative input parameters, initial conditions, and assumptions or the fuel misload analyses and they are acceptable.

15.4.7.4.3 Evaluation of Analysis Results

The staff reviewed the results in FSAR Section 15.4.7 to determine if they meet the SRP acceptance criteria. The applicant determined the limiting undetectable misload to be a cross-core misload. The applicant verified that a radial peaking augmentation factor analysis value of 1.15 bounds the radial peaking augmentation factor resulting from the limiting undetectable misload.

FSAR Table 15.4-18, “Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7) - Limiting Analysis Results,” presents the MCHFR and LHGR assuming a bounding radial peaking augmentation factor of 1.15. The staff audited the calculation for core parameters, design and operation limits and confirmed that this augmentation factor is conservative. The staff also confirmed that the MCHFR remains above the 95/95 analysis limit of the NSP4 CHF correlation, and the LHGR remains under the design limit. Because the SAFDLs are met, and because this event does not involve coolant leakage, no fission product release is postulated. Therefore, radiological evaluation and analysis to determine compliance with 10 CFR 52.137(a)(2)(iv)(A) and (B) are not necessary for the NuScale design.

15.4.7.5 Combined License Information Items

SER Table 15.4.7-1 provides the COL information items related to this review section and their descriptions. COL Items 13.5-3 and 13.5-4 provide for the development, implementation, control, and description of operating procedures. Operating procedures typically include fuel loading and startup physics testing procedures. The staff performed a detailed review of these proposed COL items in SER Chapter 13, “Conduct of Operations.”

Table 15.4.7-1 NuScale COL Information Items Related to 15.4.7

13.5-3	An applicant that references the NuScale Power Plant US460 standard design will describe the process to manage the development, review, and approval of the site-specific procedures that operators use in the main control room and locally in the plant, including normal operating procedures, abnormal operating procedures, and emergency operating procedures. The applicant will describe the classification system for these procedures, and the general format and content of the different classifications.	13.5.2
	An applicant that references the NuScale Power Plant US460 standard design will provide a plan for the development, implementation, and control of operating procedures, including	

13.5-4	preliminary schedules for preparation and target dates for completion. Additionally, the applicant will identify the group within the operating organization responsible for maintaining these procedures.	13.5.2

15.4.7.6 Conclusion

The staff has evaluated the consequences of a spectrum of postulated fuel-loading errors. The staff concludes that some errors are detectable by the available instrumentation (and hence remediable). The in-core instrumentation will be used after each refueling to confirm that power distributions and $F_{\Delta H}$ are consistent with design predictions, and $F_{\Delta H}$ is periodically verified during operation. Furthermore, the applicant has included safe fuel loading procedures and low power physics tests in its initial plant test program as well as COL items that address development of operating procedures that typically cover fuel loading and reload physics testing. Therefore, the applicant has demonstrated that the NPM-20 design meets the requirements of GDC 13 with respect to having adequate provisions to minimize the potential of a misloaded fuel assembly going undetected. The applicant also demonstrated that the NPM-20 design meets the regulatory requirements of 10 CFR 50.100, "Revocation, suspension, modification of licenses, permits, and approvals for cause," because an undetected misload would not lead to release of radioactive material from the reactor. The staff further concludes that the applicant's analysis provides reasonable assurance that no fuel rod failures will result from undetectable fuel-loading errors. For this reason, the applicant does not need to provide radiological evaluation and analysis to demonstrate compliance with 10 CFR 52.137(a)(2)(iv)(A) and (B).

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Introduction

CRA ejection is a postulated accident assuming that a CRA would be ejected out of the reactor by a mechanical failure that causes an instantaneous circumferential rupture of the control rod drive mechanism (CRDM). The CRA ejection may add large positive reactivity to the core in a couple of seconds, which results in a rapid power increase for a short period of time. However, the power rise is limited by the negative reactivity feedback generated by the Doppler effect which increases the resonance neutron absorption reactions. Reactor shutdown is initiated by the MPS upon receipt of a reactor trip (i.e., high core power, high core power rate, or high PZR pressure trip) shortly after the CRA ejection occurs. This event is classified as a postulated accident in FSAR Table 15.0-1.

15.4.8.2 Summary of Application

The applicant analyzed the CRA ejection event using five different initial powers: 0, 20, 50, 75, and 100 percent. Each power level was investigated at BOC, MOC, and end of cycle (EOC). The applicant evaluated the event using methodology described in TR-0716-50350-P, Revision 3. The TR includes several codes, including SIMULATE5 to determine the peaking factors and limiting CRA worth during the CRA ejection event and SIMULATE-3K to determine the transient core average power response. The applicant obtained the nuclear steam supply system response using NRELAP5 and used VIPRE-01 to perform the MCHFR calculation. The VIPRE-

01 computer code was also used to calculate the fuel rod enthalpy and temperature during the REA in order to evaluate margin to fuel failure criteria that are specific to this accident. The statistical subchannel analysis methodology is used to determine the uncertainty of the subchannel analyses as part of the rod ejection methodology defined in TR-0716-50350-P, Revision 3. The NuScale methodology does not permit any fuel failures resulting from exceeding MCHFR or other rod-ejection-specific fuel-failure criteria and therefore no fuel failures were calculated because of a CRA ejection.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The following GTS are applicable to this area:

- LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
    - the GTS listed in Section 15.0.0 of this SER

Technical Reports: There are no technical reports associated with this area of review.

15.4.8.3 Regulatory Basis

The following NRC regulations are the relevant requirements for the safety analyses of the rod ejection event:

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and appropriate controls to maintain these variables and systems within prescribed operating ranges,

- GDC 28, as it relates to the effects of postulated reactivity accidents that result in neither damage to the RCPB greater than limited local yielding nor sufficient damage to significantly impair core cooling capacity.

The following contains guidance for evaluation of a rod ejection accident:

- The guidance in SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

The following document also provides additional criteria or guidance in support of the SRP acceptance criteria to meet the regulatory requirements of GDC 28:

- RG 1.236, "Pressurized- Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," issued June 2020 (ML20055F490).

15.4.8.4 Technical Evaluation

The applicant proposed CRA ejection accident acceptance criteria in TR-0716-50350-P, Revision 3, which the NRC staff has reviewed, subject to the L&Cs in its SER (ML23306A212 (Proprietary), ML23310A166 (Nonproprietary)). The criteria, summarized in FSAR Section 15.4.8.3.3, cover fuel cladding failure, core coolability, and RCS peak pressure.

The applicant has addressed the three L&Cs specified in the SER for the TR. L&C No. 1 in the SER states, "An applicant or licensee referencing this report is required to demonstrate the applicability of the REA methodology to the specific NPM design. The use of this methodology for a specific NPM design requires an applicability review by the NRC staff." Because the methodology presented in TR-0716-50350-P, Revision 3, was developed based on the NPM-20 design, and no NPM-20 design changes have been made over the course of the SDAA review that would challenge the applicability of the methodology, the staff finds that TR L&C No. 1 is met.

L&C No. 2 in the SER for TR-0716-50350-P, Revision 3, states, "The REA methodology is limited to evaluation of REAs for fuel that has not experienced significant depletion with control rods inserted, such as from non-baseload operation." The staff audited the equilibrium cycle calculation package (ML24211A089), the control rod depletion analysis, and the information presented in FSAR Sections 4.2 and 4.3. The staff finds that the NPM-20 is designed to operate in a baseload mode, with CRAs normally withdrawn from the core while at power.

Therefore, the US460 design meets the requirement of L&C 2 as prescribed in the SER for TR-0716-50350-P, Revision 3. However, applicants referencing the NuScale Power Plant US460 standard design for non-baseload operation would be required to justify the applicability of TR-0716-50350-P, Revision 3, or apply a different methodology for the analysis of rod ejection accidents, in addition to fulfilling requirements of COL Item 4.2-1.

L&C No. 3 in the SER states, "The staff's approval is limited to the use of the rod ejection methodology with TR-0616-48793-P-A, Revision 1 (Reference 14), "Nuclear Analysis Codes and Methods Qualification," and TR-108601-P-A, Revision 4 (Reference 13), "Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2, Subchannel Analysis Methodology." The staff finds that the rod ejection calculations used the specified versions of these TRs and are therefore consistent with L&C No. 3.

The staff audited the supporting rod ejection accident analysis documents (ML24211A089). The staff finds that the applicant followed the methodologies presented in TR-0616-48793-P-A, Revision 1, that has been approved by the staff.

15.4.8.4.1 Evaluation Model

The EM is presented in TR-0716-50350-P, Revision 3, referenced in FSAR Chapter 15. The applicant used several codes in the EM for the REA event including SIMULATE5, SIMULATE-3K, NRELAP5, and VIPRE-01:

SIMULATE5

SIMULATE5 is a three-dimensional, steady-state, nodal diffusion, reactor simulator code used to solve the multigroup nodal diffusion equation. SIMULATE5 provides the steady-state nuclear analysis parameters used to initiate SIMULATE-3K.

SIMULATE-3K

SIMULATE-3K is a three-dimensional nodal reactor kinetics code that couples core neutronics with detailed thermal-hydraulic models to model transient neutronic analysis of the REA at various times in core life, power level, CRA positions, and initial core conditions.

NRELAP5

The NuScale NRELAP5 code is based on the Idaho National Laboratory RELAP5-3D code, Version 4.1.3. NRELAP5 addresses unique aspects of the NuScale design and licensing methodology. NuScale's NRELAP5 includes models for characterization of hydrodynamics, heat transfer between structures and fluids, modeling of fuel, reactor kinetics models, and control systems. It is used to calculate the dynamic system response and peak RCS pressure and to provide input to the VIPRE-01 subchannel CHF evaluation.

VIPRE-01

VIPRE-01 was developed based on the COBRA family of codes from Pacific Northwest Laboratory. It is used to evaluate nuclear reactor parameters such as MCHFR and hot channel fuel enthalpy.

The validation and applicability of these codes to the NuScale design is described in FSAR Section 15.0.2, supplemented by TR-0516-49416-P, Revision 4; TR 0516-49422-P, Revision 3; TR-0915-17564-A, Revision 2; TR 0716-50350-P, Revision 3, and TR-108601-P-A, Revision 4 .The staff's evaluation of the validation and applicability for these codes is documented in the respective SERs for the TRs.

15.4.8.4.2 Input Parameters and Initial Conditions

The staff reviewed the input parameters and initial conditions of the various calculation models for rod ejection analyses by reviewing the FSAR, responses to audit questions, and the initial calculations. The final calculations that result in the values in the FSAR were also audited by the staff. The applicant analyzed peak RCS pressure, MCHFR, fuel rod temperature and enthalpy. The analysis covers HZP; power at 20, 50, and 75 percent; and full power. Each power level includes BOC, MOC, and EOC conditions. FSAR Section 15.4.8.3.2 presents the input parameters and initial conditions for the models. The staff compared the input values and initial conditions used against the methodology presented in TR-0716-50350-P, Revision 3. The staff finds that the inputs and initial conditions used to analyze the NuScale response to a rod ejection accident were consistent with the methodology.

The staff audited the rod ejection analysis documents (ML24211A089). The staff finds that the applicant has followed the methodological requirement for various sensitivity studies on fuel heat thermal conductance, size of the time step, and two-phase flow correlations that are used in the subchannel analyses using VIPRE-01. **{{**

}}. The subchannel analyses supporting FSAR Section 15.4.8 use the default values for other input parameters of the VIPRE-01 code established in TR-0716-50350-P, Revision 3. The staff finds the applicant performed the mandatory sensitivity analyses. The staff also determined that the applicant used default values in the subchannel analysis as required by TR-0716-50350-P, Revision 3. On these bases, the staff finds that the subchannel model input parameter values meet the requirements of TR-0716-50350-P, Revision 3.

15.4.8.4.3 Evaluation of Analysis Results

The applicant's analysis shows that maximum core power is reached within a second from initiation of the REA event and is limited by Doppler feedback. Additionally, the applicant's analysis results show some of the conservative inputs such as the conservative scram characteristics per the methodology such as a 2-second delay before dropping the rods into the core. The results are summarized in the following paragraphs.

Peak Reactor Coolant System (RCS) Pressure

The applicant calculated the peak pressure in the RCS to be 2,231 psia based on a bounding assessment. This is below the RPV limit of 2,640 psia. **{{**

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Therefore, the staff considers the reporting of the higher value from the bounding assessment acceptable and finds that the peak pressure criterion is met.

Fuel Cladding Failure

The applicant performed the MCHFR calculations at initial conditions corresponding to HZP and 20, 50, 75, and 100 percent power. Each power level was investigated at BOC, MOC, and EOC conditions. The limiting MCHFR is 3.13, which is above the design limit. Therefore, the applicant's analysis shows that no fuel cladding violates the MCHFR criterion.

The applicant calculated the peak radial average fuel enthalpy after rod ejection at zero power conditions to be 65 calories per gram (cal/g). The hot zero power high temperature cladding failure limit is defined as 100 cal/g in TR-0716-50350-P, Revision 3. Therefore, the limit is met.

In accordance with TR-0716-50350-P, Revision 3, the PCMI failure threshold limit is 33 cal/g. The applicant calculated the maximum change in peak radial average fuel enthalpy as 21 cal/g, which is below the limit defined in TR-0716-50350-P, Revision 3. This limit is applicable to the recrystallization annealed (RXA) fuel cladding used in the NPM-20 reactor.

The applicant calculated the limiting peak fuel temperature to be 1,325 °C (2,417 °F), which is below the fuel melting temperature of 2,644 °C (4,791 °F).

Core Coolability

The applicant discussed the fuel and cladding integrity results in FSAR Section 15.4.8.3.4. The analysis includes various initial power levels and times in cycle, which is consistent with the guidance in SRP Section 15.4.8.

The applicant's analyses result in a limiting peak radial average fuel enthalpy of 65 cal/g, which corresponds to an initial power of 100 percent at BOC conditions. This value is below the limit of 230 cal/g provided in TR-0716-50350-P, Revision 3.

15.4.8.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.4.8.

15.4.8.6 Conclusion

The staff reviewed the control rod ejection accident, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the accident. Through this review, the staff verified that the applicant provided sufficient information and that the review supports the conclusions as discussed in Section 15.4.8.4 of this SER.

The staff concludes that the applicant meets GDC 13 requirements in terms of the rod ejection accident analysis by demonstrating that all credited instrumentation was available, and that actuations of protection systems, automatic and manual, occurred at values of monitored parameters that were within the instrument's prescribed operating ranges. Chapter 7 of this SER provides the staff's review of the instrumentation.

The staff also concludes that the applicant meets GDC 28 requirements for prevention of postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding or result in sufficient damage to impair the core cooling capability significantly. The requirements are met by the use of the approved methodology and demonstrating compliance with the prescribed limits. The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. As the calculations demonstrate peak fuel temperatures below melting conditions, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten uraniuam dioxide presumably did not occur. The pressure surge results in a pressure increase below pressure limits for the maximum control rod worths assumed. The staff believes that the calculations are sufficiently conservative, both in initial assumptions and analytical models to maintain primary system integrity.

15.4.9 Spectrum of Rod Drop Accidents

This event is specific to BWRs and therefore does not apply to the NPM design. The PWR equivalent of a rod drop is rod ejection, which in the staff discusses in SER Section 15.4.8. SER Section 15.4.3 discusses CRMs, including a dropped CRA.

15.5 Increase in Reactor Coolant Inventory

15.5.1 Chemical and Volume Control System Malfunction

15.5.1.1 Introduction

A CVCS malfunction may cause an increase in the RCS inventory and pressure. The relatively cold makeup water combined with a negative MTC can increase core reactivity. For limiting scenarios, the pressurizer pressure or level increase results in a reactor trip. The applicant classified this event as an AOO, which is consistent with the DSRS for the NuScale SMR design.

15.5.1.2 Summary of Application

The applicant identified the malfunction of the two 20 gpm CVCS makeup pumps that maximize makeup flow as the limiting scenario for a CVCS malfunction that increases RCS inventory. The applicant stated that no single failure, including any loss of AC or DC power, would result in a more serious outcome for the increase in RCS inventory events. The thermal-hydraulic analysis of the NPM response to the event was performed using NRELAP5, and the subchannel CHF analysis was performed using VIPRE-01. The applicant's analysis resulted in a peak RCS pressure of 2,288 psia, a peak SGS pressure of 1,285 psia (compared to the acceptance criteria of ≤2420 psia), and an MCHFR of 2.32 (compared to the safety limit and minimum value of 1.43).

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The following GTS are applicable to this area of review:

- LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
- LCO 3.4.10, "Low Temperature Overpressure Protection (LTOP) Valves" and
- the GTS listed in Section 15.0.0 of this SER.

Technical Reports: There are no technical reports associated with this area of review.

15.5.1.3 Regulatory Basis

The following NRC regulations contain relevant requirements for this review:

- 10 CFR 52.137(a)(2), which requires evaluations to show that safety functions will be accomplished. Descriptions shall be sufficient to permit understanding of the system design relationships to the SEs.
- GDC 10, which requires that the reactor core and associated coolant control, and protection systems be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.
- GDC 13, which requires, in part, that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for AOOs, as appropriate, to ensure adequate safety. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.
- GDC 15, which requires that the RCS and its associated auxiliary control and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operations, including AOOs.
- GDC 26, which requires, in part, the reliable control of reactivity changes to ensure that SAFDLs are not exceeded under conditions of normal operation, including AOOs.

DSRS Section 15.5.1, "Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," lists the following AOO acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
- Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see DSRS Section 4.4).
- An AOO should not generate a more serious plant condition without other faults occurring independently.

15.5.1.4 Technical Evaluation

The applicant organized FSAR Section 15.5.1, "Chemical and Volume Control System Malfunction," to reflect the DSRS. This technical evaluation is organized accordingly.

15.5.1.4.1 Identification of Causes and Accident Description

The applicant identified the malfunction of the CVCS makeup pumps resulting in 40-gpm makeup (and zero letdown) as the limiting scenario for the increase in RCS inventory. The applicant stated in the non-LOCA methodology discussed in Section 15.0.2 of this SER, that the pump malfunction could be caused by a spurious PZR water-level signal. The applicant classified this event as an AOO, which is consistent with the DSRS. SER Section 15.4.6 evaluates the case of a CVCS malfunction resulting in a boron dilution event.

15.5.1.4.2 Sequence of Events and Systems Operation

The increase in reactor coolant inventory event terminates with automatic reactor trip on high PZR pressure or high PZR level, CVCS isolation on high PZR level, and DHRS actuation on high PZR or steamline pressure. The applicant credited these automated safety functions (e.g., CVCS isolation, Secondary System Isolation), and did not credit operator action to mitigate this event. The sequence of automatic actions depends on the set of initial conditions. The applicant stated that its analysis assumed the availability of AC and DC (EDNS and EDAS) power because the CVCS cannot function without AC or EDNS power, and the CVCS flow pathways are isolated on a loss of EDAS. The staff agrees that this is a conservative assumption because loss of any of those power supplies would terminate the CVCS flow addition event and be non-limiting. The applicant documented the sequence of events, initial conditions, and results in FSAR Section 15.5.1, Table 15.5-1, "Sequence of Events Chemical Volume and Control System Malfunction—Limiting Reactor Coolant System Pressure Case, Table 15.5-2 "Initial Conditions— Chemical Volume and Control System Malfunction," Table 15.5-3 "Summary of Results— Chemical Volume and Control System Malfunction," and Figures 15.5-1, "Pressurizer Level – Chemical Volume and Control System Malfunction - Limiting Reactor Coolant System Pressure Case" through 15.5-10 "Minimum CHFR – Chemical Volume and Control System Malfunction - Limiting Minimum Critical Heat Flux Ratio Case."

15.5.1.4.3 Evaluation Model

The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including NRELAP5, to analyze the thermal-hydraulic response to the event. In addition, the applicant performed a subchannel analysis using the VIPRE-01 code to identify the limiting MCHFR. Section 15.0.2 of this SER describes the staff's evaluation of these codes.

15.5.1.4.4 Input Parameters, Initial Conditions, and Assumptions

The applicant conducted sensitivity studies to determine the initial conditions that maximize RCS and SG pressure and minimize MCHFR for several of the initial conditions. Those initial conditions included the RCS temperature, PZR level, PZR pressure, CVCS makeup fluid temperature, CVCS makeup flow, and the availability of PZR spray. Other initial conditions, including core thermal power, decay heat, initial pool temperature, coefficients of reactivity, SG tube plugging, FW initial temperature, SG pressure and minimum RCS flowrate were conservatively biased. The applicant varied the initial PZR pressure between the maximum and minimum bias. The limiting SG pressure and RCS pressure cases included initial PZR pressure biased high to 2,070 psia. The limiting MCHFR case came with the PZR pressure biased low to 1,930 psia.

The staff agrees that the applicant's limiting sets of initial conditions are conservative for the following reasons: (1) the applicant provided a conservative basis for each initial condition, and (2) the applicant demonstrated with sensitivity studies, listed in the non-LOCA methodology discussed in Section 15.0.2 of this SER (with exceptions noted below), that reasonable variations of the initial conditions did not significantly alter the peak pressures.

The staff agrees that the applicant's total makeup flow rate of 40 gpm is suitably conservative. The capacity of each of the two CVCS makeup pumps is 20 gpm; the makeup flow rate of 40 gpm assumes each pump operates at full capacity. The applicant did not add additional makeup flow to account for potential uncertainties in the pump flow rate. The staff reviewed this assumption and agrees it is suitably conservative for the following reasons: (1) a small increase in makeup flow to account for uncertainty is not expected to impact peak RCS pressure because RCS pressure is limited by release of RCS inventory to containment by actuation of one of the two RSVs, (2) the maximum peak SG pressure of 1,285 psia remains below 110 percent of the secondary system design pressure of 2,420 psia and has significant margin for small variations in primary makeup flow rate, and (3) minimum MCHFR occurs early in the event and is insensitive to small variations in makeup flow.

15.5.1.4.5 Evaluation of Analysis Results

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. The analytical results for the secondary pressure figures of merit reported in FSAR Table 15.5-3, "Summary of Results - Chemical and Volume Control System Malfunction," are the NRELAP5 results obtained from NRELAP5 without the adjusted DHRS condenser headers modeling, but with a bounding additional 50 psia added to the result to account for the DHRS modeling adjustment. This 50 psia is a value the applicant calculated as a conservative bound beyond the SGS pressure increase **{{** **}}** due to adjusting the DHRS NRELAP5 input from the limiting LOAC non-LOCA methodology event, which the staff agrees is a bounding change to add to the previously calculated SGS pressure values.

FSAR Table 15.5-3 presents the limiting analysis results for this event. The staff finds that the predicted plant response satisfies the AOO acceptance criteria because (1) the analysis demonstrates that the MCHFR of 2.32 is above the 95/95 limit of 1.43; (2) the maximum RCS pressure of 2,288 psia remains below 110 percent of the RCS design pressure (2,420 psia); and (3) the maximum peak secondary pressure of 1,285 psia remains below 110 percent of the secondary system design pressure (2,420 psia). The applicant’s analysis also demonstrated that the NPM reached a stable, safe condition after the CVCS malfunction.

15.5.1.4.6 Radiological Consequences

Based on the results of the analysis and barrier performance, the applicant concluded, and the staff agrees, that the radiological consequences for this AOO are bounded by the results in FSAR section 15.0.3.

15.5.1.5 Combined License Information Items

There are no COL information items associated with FSAR Section 5.5.

15.5.1.6 Conclusion

The staff reviewed the CVCS malfunction event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. The staff concludes that that the DSRS acceptance criteria are met and therefore the consequences of the CVCS malfunction event meet the requirements set forth in GDC 10, 13, 15, and 26, and 10 CFR 52.137(a)(2).

15.6 Decrease in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a Reactor Safety Valve

FSAR Section 15.6.1, “Inadvertent Opening of a Reactor Safety Valve,” states that an inadvertent opening of an RSV has the same thermal-hydraulic effects as, and is bounded by, an inadvertent operation of ECCS. That event is evaluated in FSAR Section 15.6.6, and that evaluation includes the inadvertent opening of an RSV.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Introduction

A break or leak from a line connected to the RCS that penetrates containment can cause a direct release of reactor coolant outside containment. The staff’s review in this section of the SER focuses on the non-radiological aspects of this event (e.g., RCS mass release, break location, fuel integrity) to ensure that conservative and bounding thermal-hydraulic inputs are used in the radiological aspect of this event. Section 15.0.3 of this SER documents the review of the radiological aspect of this event.

15.6.2.2 Summary of Application

The applicant provided an event description in FSAR Section 15.6.2.

Lines that carry primary coolant outside containment are the CVCS lines: injection and discharge lines, PZR spray lines, and RPV high point degasification line. Failure of lines carrying primary coolant outside containment is a non-mechanistic break in any one of these lines. To determine the most severe consequences of the failure of lines carrying primary coolant outside containment, the applicant analyzed a spectrum of break sizes and locations for the CVCS lines after the CVCS isolation valves. Primary coolant is released from the break into the RB until CVCS CIVs close.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this area of review.

15.6.2.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to meeting the SAFDLs during normal operations, including AOOs.
- GDC 55, as it relates to providing isolation valves to primary coolant lines that penetrate primary reactor containment.

The guidance in SRP Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," lists the non-radiological acceptance criteria for demonstrating conformance with these requirements, as well as review interfaces with other SRP/DSRS sections.

15.6.2.4 Technical Evaluation

The following sections contain the staff's technical evaluation of the applicant's analysis of small primary coolant line failure outside containment.

15.6.2.4.1 Causes

In FSAR Section 15.6.2, the applicant stated that the small lines carrying primary coolant outside containment are the CVCS injection and discharge lines, PZR spray line, and the RPV high-point vent line. These lines extend from the RPV through the CNV and include double isolation capability by means of CIVs. The applicant states that the valves isolate on a containment isolation signal. In addition, the valves on the PZR spray line and the RPV high point vent line isolate on low pressurizer pressure. The CVCS injection and discharge lines contain flow restricting venturis. Failure of these four lines is evaluated for both thermal-hydraulic and radiological consequences. A non-mechanistic break in these lines is considered on the CVCS side of the isolation valves. For breaks at the welded connection between the CNV and the inboard isolation valves, NuScale requested an exemption in SDAA

Part 7, Section 18 in ML25057A492. The staff evaluation of that exemption request is contained in Section 15.6.5.3 of this SER

The staff finds the applicant's assessment of causes leading to the event acceptable because it considered a spectrum of CVCS breaks in different locations throughout the system.

15.6.2.4.2 Methodology

In the failure of small lines carrying primary coolant outside containment analyses, the staff notes that the applicant evaluated a spectrum of break sizes and locations. The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER to analyze the thermal-hydraulic response to the event.

The applicant did not perform a CHF calculation for this event. FSAR Section 15.6.2.5 states that the MCHFR response for this event is bounded by the rapid depressurization event in FSAR Section 15.6.6. The staff notes that this is true for breaks on the CVCS side of the CVCS isolation valves. As discussed above, breaks at the welded connection between the CNV and the inboard isolation valves are not considered in this FSAR section and are evaluated as part of an exemption request. Considering only the breaks on the CVCS side of the isolation valves, the staff considers that the behavior of important parameters to MCHFR (e.g., reactor power, RCS flow, core inlet temperature, core exit pressure) for a break in a small line carrying primary coolant is less limiting than for other MCHFR-challenging events, such as cooldown and reactivity insertion events. In addition, fuel temperature decreases upon reactor trip, and the core water level remains well above the top of the active fuel for the failure of small lines carrying primary coolant outside containment when the break is on the CVCS side of the isolation valves. For these reasons, the staff finds that the potential for fuel failure is precluded in these cases.

Section 15.0.3 of this SER documents the staff's evaluation of the radiological effects of failure of small lines carrying primary coolant outside containment for breaks downstream of the isolation valves.

15.6.2.4.3 Model Assumptions, Input, and Boundary Conditions

The applicant's analyses assume that ESFs perform as designed, with allowance for instrument uncertainty, unless otherwise noted. No operator action is credited to mitigate the effects of line breaks outside containment on the CVCS side of the isolation valves. In addition, no external power source is credited.

The applicant evaluated various inputs and assumptions to determine the limiting break scenario with respect to radiological and thermal-hydraulic consequences. The results are presented in graphical form. The maximum RCS pressure scenario identified is a small break in the CVCS discharge line outside containment with a coincident loss of normal AC power. The staff audited the applicant's calculations and noted that the maximum mass and energy release scenario, limiting iodine spiking scenario and maximum mass and energy release scenario post-trip were all evaluated. **{{**

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The staff reviewed the applicant's single-failure assumptions regarding this event. The staff confirmed that the applicant considered single failures but determined that there were none that affect the transient results for the event. The staff audited calculations which indicated that the applicant does consider failure of an isolation valve through consideration of the leakage through the remaining closed redundant isolation valve in its downstream radiological analysis, therefore the staff finds this treatment acceptable.

15.6.2.4.4 Evaluation of Analysis Results

In these analyses, reactor trip and, for the high power cases, DHRS actuation occur, but the ECCS trip setpoints are not reached. Upon isolation of the break, a normal shutdown of the module proceeds using the DHRS, for high power cases. The mass and energy releases to the reactor building are maximized to conservatively maximize the potential radiological consequences. The applicant's results presented in FSAR Section 15.6.2.3.3 for breaks on the CVCS side of the CVCS isolation valves, show that the reactor water level remains well above the top of the active fuel**.** and that the core remains subcritical for all break cases and with all power assumptions. The RCS and fuel temperatures stabilize following the breaks.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 input. The analytical results for the secondary pressure figures of merit reported in Table 15.6-3 of the FSAR are the NRELAP5 results obtained from NRELAP5 without the adjusted DHRS condenser headers modeling, but with a bounding additional 50 psia added to the result to account for the DHRS modeling adjustment. This 50 psia is a value the applicant calculated as a conservative bound beyond the SGS pressure increase **{{ }}** due to adjusting the DHRS NRELAP5 input from the limiting LOAC non-LOCA-methodology event, which the staff agrees is a bounding change to add to the previously calculated SGS pressure values.

The staff audited the applicant's calculations supporting this FSAR section (ML24211A089) and consider them to be reasonable.

The staff finds that the maximum mass and energy releases calculated by the applicant are appropriate for purposes of input into the downstream radiological analysis. The staff also finds that fuel integrity is maintained during this event for breaks on the CVCS side of the CVCS isolation valves because the water level in the reactor vessel remains above the top of active fuel.

15.6.2.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.6.2.

15.6.2.6 Conclusion

The staff's evaluation of the radiological effects of failure of small lines carrying primary coolant outside containment is documented in Section 15.0.3 of this SER, which addresses the regulatory requirements in 10 CFR 52.137(a)(2)(iv)(A) and 10 CFR 52.137(a)(2)(iv)(B).

As documented above, for this event sequence, pending closure of the above noted open items, the staff determined that the applicant's analysis demonstrates that the SRP acceptance criteria are met. Therefore, the staff concludes that the consequences of failure of small lines carrying primary coolant outside containment meet the relevant requirements set forth in GDC 10 and 55, with respect to this event.

15.6.3 Steam Generator Tube Failure (Thermal Hydraulic)

15.6.3.1 Introduction

An SGTR is a postulated accident caused by a rapid propagation of a circumferential crack that leads to a double-ended rupture of the tube. Reactor coolant passes from the primary side of the SG into the secondary side and travels through the main steamlines to the turbine into the environment. A secondary criterion is to prevent overfill of the SG secondary to prevent water from entering the steamlines and potentially preventing closure of the MSIVs.

Radionuclides contained in the primary coolant are discharged through the failed tube until the faulted SG is isolated by automatic closure of the MSIVs.

15.6.3.2 Summary of Application

The applicant analyzed the SGTR event in terms of margin to fuel thermal design limits and maximized radiological consequences. Analyses were performed for many scenarios, including with and without power available, to ensure that the most limiting conditions are considered. The applicant evaluated this event using NRELAP5 to obtain the NPM thermal-hydraulic responses in accordance with the non-LOCA TR, TR-0516-49416-P, Revision 4, described in Section 15.0.2 of this SER. The applicant determined the SGTR coincident with power available to be limiting in terms of MCHFR and radiological consequences. The applicant determined that the MCHFR is above the 95/95 DNBR limit; hence, no fuel failure is predicted to occur. The applicant's FSAR analysis also concludes that, with conservative initial conditions, the SGTR event can be controlled by no operator actions with radiological releases remaining below 10 CFR Part 100 regulatory limits (or within the limits of 10 CFR 50.67 for alternate source term) and that the affected SG liquid level increase does not lead to more severe consequences (i.e., the MSIV is unable to close). In addition, for the NPM-20, the methodology for SGTR mass releases has been revised to use event-specific or bounding values.

It is also important to note that the design of the helical coil SGs, as described in FSAR Section 5.4, is different from conventional PWR design SGs in that the primary coolant is located on the outside (shell side) of the tubes. Thus, the volume of the secondary inventory is considerably smaller than in conventional designs, increasing the potential for SG overfill and other differences in transient responses.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The following GTS are applicable to this area of review:

- LCO 3.4.5, "RCS Operational Leakage"
- LCO 3.4.8, "RCS Specific Activity"
- GTS listed in Section 15.0.0 of this SER

Technical Reports: There are no technical reports associated with this section of the FSAR.

15.6.3.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 14, as it relates to ensuring an extremely low probability of failure of the RCPB

- GDC 19, as it relates to the requirement that a control room be provided from which personnel can operate the nuclear power unit during both normal operating and accident conditions, including a LOCA.

- GDC 34, as it relates to the requirement that a system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded.

DSRS Section 15.0.3 and SRP Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," list the following acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections:

- Affected SG time to isolate and unaffected SG(s) until cold shutdown is established should be identified.

- The potential for fuel failures and the core thermal margins resulting from the postulated accident should be verified.

- It should be verified that the most severe case has been considered with respect to the release of fission products and calculated doses.

- Pressure in the reactor coolant and main steam systems should be maintained below 120 percent of the design values.

15.6.3.4 Technical Evaluation

15.6.3.4.1 Evaluation Model

The event is initiated by the failure of an SG tube that causes a decrease in PZR pressure and level. The applicant used the non-LOCA methodology discussed in Section 15.0.2 of this SER, including NRELAP5 Version 1.7 and Revision 4 of the non-LOCA NRELAP5 plant model, to analyze the thermal-hydraulic response to the event.

The SGTR NRELAP5 model is developed from the NPM base plant base model with an averaged lumped core with point kinetics used to calculate reactivity feedback to the core power from the moderator, fuel, and decay heat. The model simulations are performed using event specific conservatisms. The staff reviewed the conservatisms used and determined them to be adequate.

15.6.3.4.2 Input Parameters, Initial Conditions, and Assumptions

The applicant conducted sensitivity studies to identify the limiting input conditions to identify the most challenging break size and location, SG tube plugging, single failure, feedwater temperature and fuel kinetics with respect to the figures of merit for this event, which are RCS pressure, secondary system pressure and limiting mass released and iodine spiking duration for the SGTR transient. The iodine spiking duration is the time calculated between reactor trip and isolation of the affected SG.

Since NuScale elected to use a bounding release of primary coolant of 23,000 pound mass (lbm) for the dose consequence analysis in FSAR 15.0.3.7.2 cases to target maximum iodine spiking were not necessary, however, the staff confirmed via audit that cases were run to confirm maximum releases were well below the bounding assumption. Thus, the primary concern for the SGTR as presented in FSAR Section 15.6.3 is in determining the maximum primary pressure and maximum secondary pressure. For each analysis performed, the applicant's evaluations of the SGTR event considered a range of initial conditions, biases, and conservatisms. The staff audited those parameters and determined they considered suitably conservative parameters.

The staff reviewed the initial parameter values and biases presented in FSAR Section 15.6.3 for the RCS pressure and secondary pressure figures of merit and noted that the applicant assumed suitably conservative parameters that maximize the consequences of the events including a 102 percent initial core power level, higher initial PZR pressure and level, BOC core parameters, nominal feedwater temperature and no tube plugging. For both cases, the applicant determined it was limiting to assume no single failure of the MSIV to close to maximize pressures.

The staff audited the applicant's SGTR sensitivity studies, which investigated the most limiting initial conditions and loss of power assumptions, to confirm that they led to the most limiting results. The staff finds that the input parameters and initial conditions are suitably conservative and result in the most limiting conditions for the pressure responses and radiological mass releases.

No operator action is assumed; however, the staff noted that the secondary MSIVs that are required to isolate the leakage are non-safety-related valves, as described in FSAR Section 10.3.2.1.2.. This approach is consistent with the existing staff position for crediting non-safety-related components as a backup when assuming a single failure. Specifically, NUREG-0138, Issue 1 allows flexibility in the acceptance of non-safety grade (i.e., non-safety-related) equipment for failures of secondary system piping, in part, because they have a significantly lower potential for release of fission products than a breach of the primary system boundary like the SGTR event. The staff reviewed the basis for crediting the secondary MSIVs for event mitigation, design descriptions and specifications, augmented quality and testing requirements in FSAR Table 3.9-17, FSAR Section 3.9.6, Section 15.0.0.6.6, and associated TS. In addition, in a letter dated October 31, 2023 (ML23304A490), the applicant provided the results of a sensitivity calculation performed to demonstrate that a failure of the secondary MSIVs to close during a SGTR event would not result in exceeding design basis dose release limits. The staff reviewed the calculation and determined the results were consistent with the applicant's statements and showed significant margin to the dose criteria of 10 CFR 52.137(a)(2)(iv).

Based on the information above, including the valve design, testing, surveillance and operability requirements, and the consequences assessed by sensitivity analysis, the staff finds the use of the non-safety-related secondary MSIV for mitigating an SGTR event for the NuScale design acceptable. Section 3.9.6 of this SER includes the staff's detailed review of the augmented quality and testing requirements applied to the secondary MSIVs.

15.6.3.4.3 Results

The tube rupture causes a decrease in PZR pressure and level which results in reactor trip actuation on a low PZR pressure signal or a low PZR level signal. The DHRS is actuated, and closure of the FWIVs and MSIVs follows to isolate the SGs and terminate the loss of reactor coolant to the environment. Core decay heat then drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS associated with the intact SG. Transients that also assume loss of AC power generally result in limiting RCS and SGS pressure conditions.

The applicant performed two primary analyses for the SGTR event. The first analysis uses conservative input parameters that maximize the potential for RCS mass and radiological release. The calculated radiological consequences are compared to those presented in FSAR Section 15.0.3 to confirm the results in FSAR Section 15.0.3 remain bounding. Those analyses were audited by the staff and the specific results are not presented in FSAR Section 15.6.3. The second analysis determines the limiting pressure responses in the RCS and SGS to ensure that peak pressures remain below the design pressures. The applicant presented the results of these limiting pressure response analyses in FSAR Section 15.6.3. These sets of analyses considered three limiting scenarios to identify maximum SGTR: (1) maximum mass release, (2) maximum RCS pressure, and (3) maximum SGS pressure. Additionally, the analyses considered a full double-ended guillotine down to partial tube split breaks ranging from 100 percent to 1 percent of the SG tube area.

The staff reviewed the results of the limiting pressure response analyses presented in FSAR Section 15.6.3 to determine if they meet the SRP acceptance criteria. The staff reviewed the transient behavior of several parameters by evaluating plots of the parameters as a function of time. The staff considered the core power, SG pressures and levels, core temperature and levels, SG tube break flow rate, and DHRS heat removal rates. The SGTR is a slow depressurization that does not result in an increase in power or significant decrease in RCS flow. The applicant stated that fuel integrity is not challenged by an SGTR event and the event is bounded by the inadvertent RVV opening event. Therefore, the applicant did not specifically address fuel integrity in the SGTR analysis results. The staff agreed that the SGTR event is not limiting in regard to MCHFR, and that the CHFR will remain well above the 95/95 DNBR limit based on comparison to more limiting events (e.g., inadvertent opening of an ECCS valve, decrease in feedwater temperature and uncontrolled CRA withdrawal). The staff also found that the fuel temperature decreases upon the reactor trip and that core water level remains well above the top of the active fuel, such that the potential for fuel failure is precluded in all cases.

As part of its review of transient parameters, the staff verified that the sequence of events was reasonable given the automatic actuations of protection systems at their analytical setpoints.

Maximum Mass Release

The magnitude of mass released from an SGTR event depends on the size of the rupture and timing of SG isolation, since SG isolation ends the release of mass from the RPV to other plant areas. Maximizing the mass released to the environment and the duration of the iodine spike (elapsed time from reactor trip to SG isolation) maximizes the radiological consequences.

The analyses the staff audited for this SGTR event **{{**

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The staff audited (ML24211A089) the applicant's calculations and confirmed that the most severe case for mass release of fission products has been considered for comparison to the bounding analysis in FSAR Section 15.0.3.

Maximum Iodine Spiking Time

No sensitivity analysis was provided for the limiting iodine spiking time. The longest time is very conservatively bounded by the assumptions of FSAR 15.0.3.7.2.

Maximum Reactor Coolant System and Steam Generator Pressure

The staff reviewed the applicant's SGTR case that resulted in a limiting RCS pressure and audited details in the applicant's calculations. For the limiting RCS pressure case, a high initial SG pressure, no tube plugging and nominal RCS average temperature were the most limiting. This case assumes a 20 percent split break tube failure at the top of the SG with a coincident loss of normal AC power, resulting in immediate closure of the turbine control valves by 1.0 second. The RCS pressurizes because of loss of SG heat removal, and the MPS actuates a reactor trip based on a high PZR pressure signal at 5 seconds, which also results in containment isolation and DHRS actuation. The peak RCS pressure of 2,296 psia occurs at about 8 seconds. Note the maximum pressure cases for both RCS and SGS will involve lift of the first RSV with the peak RCS pressure occurring near the same time.

Subsequent to completing the majority of the analyses reported in this section, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers NRELAP5 input. For this event, the reported analytical results relative to the MCHFR and RCS pressure figures of merit are the results without the adjusted DHRS modeling, which the staff finds acceptable as MCHFR and RCS pressure are not significantly impacted by this modeling adjustment. The analytical results for secondary pressure figures of merit reported in Table 15.6-7, "Steam Generator Tube Failure – Results," of the FSAR are the NRELAP5 results obtained after adjusting the DHRS modeling.

The staff also reviewed and audited calculation details for the limiting SG pressure case, which assumes a low RCS average temperature, nominal SG pressure, no SG tube plugging, and a 100 percent split break tube failure at the top of the SG, also with coincident loss of normal AC power. The MPS actuates a reactor trip based on a high PZR pressure signal at 3 seconds, which also causes secondary system isolation and DHRS actuation. The SG secondary pressurizes because of the SGTR and the containment isolation. The affected SG reaches a pressure that approaches the RCS pressure and peaks at 2,038 psia at about 1,845 seconds. Afterwards, the RCS and secondary begin a gradual decline as the RCS is cooled by the intact SG and its associated DHRS. The staff confirmed that for the worst RCS pressure and SG pressure cases, the RCS and secondary pressure remained below 120 percent of their design pressures.

15.6.3.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.6.3.

15.6.3.6 Conclusion

Based on the review of the SGTR analyses, including the sequence of events, initial conditions, and single failure assumptions used in the analytical models, and the predicted consequences of the transient, the staff concludes that the applicable requirements of GDCs 13, 14, and 34 have been met. The SGTR accident analysis did not evaluate CHFR margins, but fuel damage is not anticipated due to the large subcooling margin and large collapsed liquid level above the top of the core that are maintained during this event. Section 15.0.3 of this SER provides the staff's assessment of the radiological consequences of a SGTR are provided in.

15.6.4 Main Steamline Failure Outside Containment (BWR)

A main steamline failure outside containment is a BWR-specific event and therefore does not apply to the NPM-20 design.

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

This section describes the evaluation of the applicant's FSAR analyses of the NuScale NPM-20 reactor responses to postulated LOCAs as the result of piping breaks within the RCPB, including LTC up to 72 hours after the event. These analyses are used to determine the LCOs, limiting safety system settings, and design specifications for safety-related components and systems.

15.6.5.1 Loss-of-Coolant Accident

15.6.5.1.1 Introduction

A LOCA is a postulated accident resulting from instantaneous rupture of an RCS pipe within the RCS boundary. A spectrum of break sizes for both double-ended guillotine break and split break types inside the containment are analyzed. For the NPM-20 LOCA events, the limiting MCHFR case for LOCA is the 100 percent CVCS discharge line break. The line is a 2-inch line connected to the CVCS system and the RPV. The methodology used is the LOCA EM described in Section 15.0.2 of this SER**.**

The LOCA event for the NuScale NPM-20 design is unique compared to traditional PWRs because of the small size of the RCS piping and also because RCS inventory lost during a LOCA is preserved within containment and relied on for recirculation back to the core sometime after event initiation, depending on the size of the break. The methodology uses the deterministic 10 CFR Part 50, Appendix K, approach, and the NPM-20 is designed to eliminate or reduce many of the design basis LOCA consequences compared to a typical large PWR, in which the most important LOCA consequences would be PCT resulting from core uncovery, core refilling, core reflooding, fuel cladding swelling and rupture, and fuel metal water reaction. Consequently, NuScale has requested exemptions in SDAA Part 7 from Appendix K, Parts 1.A.4, 1.A.5, 1.B, 1.C.1.b, and 1.C.5.a since phenomena related to these Appendix K criteria are essentially avoided by design of the NPM-20 ECCS. Section 15.0.2 of this SER details the staff's review of the exemption request. The NPM-20 LOCA calculations show significant margins to PCT of 1,204 °C (2,200 °F) as required by 10 CFR 50.46(b)(1) and the other criteria contained in 10 CFR 50.46(b)(2) through (b)(4). The relevant figures of merit are not PCT but (1) collapsed liquid water level above the core, (2) CHFR, and (3) containment pressure and temperature. Therefore, applicability of the LOCA methodology is limited by design as described in SER Section 15.0.2.2 as it does not address post CHF heat transfer phenomena, including cladding oxidation, clad hydrogen production, or clad geometry changes such as swell and rupture, which are not encountered due to the unique design of ECCS recirculation cooling.

The NPM-20 LOCA event addresses the ECCS performance up to the time when stable recirculation flow is established from the containment back to the RPV, pressures and levels in containment and the RPV approach a stable equilibrium condition (i.e., steady flow is recirculating through the RRVs), and core decay heat is removed by boiling in the core with steam exiting through the RVVs and then condensing in the containment. The DHRS is also available to either supplement or provide core cooling and is credited in the LOCA analysis. The DHRS adds additional core cooling capacity during the NPM-20 LOCA that results in faster depressurization and impacts primarily the smaller break size LOCAs. The faster depressurization allows for earlier actuation of ECCS, less loss of RCS inventory, and greater margins to the figures of merit.

15.6.5.1.2 Summary of Application

The applicant provided an event description in FSAR Section 15.6.5 summarized below.

The LOCA event simulates a compromise in the RCPB resulting in RCS inventory loss at a rate that exceeds the capacity of normal makeup flow. The applicant assessed a spectrum of break sizes and locations of the RCS pressure boundary piping, and the event is analyzed for core thermal hydraulic effects and is classified as a postulated accident.

The LOCA break spectrum is separated into two categories: (1) a liquid space break inside containment consisting of the CVCS injection (i.e., charging) line and discharge line and (2) a steam space break inside containment consisting of the high point vent line and PZR spray supply line. The progression of these events is similar, with the steam space breaks depressurizing the RCS faster, resulting in slight differences in timing of the key events because of the composition of the liquid or steam break flow. There are three distinct phases of the LOCA progression:

(1) Phase 0 of the event sequence is defined for pre-scram CHF analysis beginning at break initiation and ending approximately **{{** **}}** after the scram.

(2) Phase 1a is the blowdown of the postulated break in the RCS into the containment to the point that the RPV riser two-phase mixture level drops below the MPS setpoint, which actuates the ECCS (valves start opening). The MPS is actuated early in the event to initiate reactor trip, generally based on high CNV pressure or low PZR pressure, which then isolates containment, and actuates DHRS.

(3) Phase 1b includes a second, more rapid blowdown that begins with the opening of RVV valves resulting in pressure equalization between the RCS and containment allowing the cooled, depressurized RCS inventory to begin filling the containment to the point that RCS pressure drops below the RRVs IAB release pressure and the discharged RCS fluid from the CNV is returned to the RPV downcomer.

(4) Phase 2 includes stable RRV recirculation and transition to long-term cooling.

Generally speaking, ECCS can be actuated by: low and low-low RPV riser level signals; 8 hours after reactor trip; loss of AC power after 24 hours; or loss of DC power. When ECCS is actuated, the two RVVs immediately open, and the RRVs open after the CNV to RPV pressure difference drops below the IAB release threshold. When a loss of EDAS is assumed at event initiation, it causes both RVVs to open immediately. No operator action is credited in this event analysis. The applicant considered break sizes from 100 percent down to 2.2 percent.

The applicant analyzed this event using NRELAP5 to obtain the NPM-20 time dependent thermal hydraulic response for collapsed level above the core and MCHFR. The applicant stated that the input parameters and initial conditions used in the LOCA analysis are selected to provide conservative calculation results in compliance with the requirements in 10 CFR Part 50, Appendix K.

The applicant concluded that criteria 1 through 4 in 10 CFR 50.46(b) are met and that the MCHFR remains greater than the safety limit. The applicant further stated that CNV pressure and temperature remain within design limits, and that the collapsed level remains well above the top of the active fuel.

The transition from the LOCA analysis to the post-LOCA long-term core cooling begins a third phase after natural circulation between the RPV and the containment through the RVVs and RRVs has reached a stable steady state with adequate decay heat cooling. A separate TR addresses the latter phase (up to 72 hours after the event) in; Section 15.0.5 of this SER contains the staff evaluation.

15.6.5.1.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR 50.46, as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the RCPB.

- 10 CFR Part 50, Appendix K, which provides the required and acceptable features of ECCS EMs.

- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- 35, as it relates to demonstrating that the ECCS would provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

- 10 CFR Part 100, as it relates to mitigating the radiological consequences of an accident.

- 10 CFR 52.137(a), as it relates to demonstrating compliance with any technically relevant portions of requirements related to Three Mile Island in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

The staff notes that the applicant provided PDC for GDC 34 and 35 which are functionally identical to GDC 34 and 35 with the exception of the discussion related to electric power, which PDC 34 and 35 eliminate. The SER for Chapter 8 gives a detailed discussion of NuScale's reliance on electric power and the related exemption from GDC 17 and 18, as well as the electrical power provisions of GDC 34 and 35. Neither PDC 34 nor PDC 35 requires the DHRS or ECCS to have electrical power (offsite or onsite) to perform their safety functions for decay heat removal or emergency core cooling. Section 8.1.5 of this SER describes the staff's evaluation of these exemptions under 10 CFR 50.12 from GDC 17 and 18 and the electrical power provisions of GDC 34 and 35.

DSRS Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," lists the following acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP sections:

- The applicant evaluated ECCS performance in accordance with an EM that satisfies the requirements of 10 CFR 50.46. RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," and Section I, "Required and Acceptable Features of the Evaluation Models," of Appendix K to 10 CFR Part 50 provide guidance on acceptable EMs.

- The analyses should be performed in accordance with 10 CFR 50.46, including methods referred to in 10 CFR 50.46(a)(1) or (2). Additionally, the LOCA methodology used in the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b).

- The calculated maximum fuel element cladding temperature does not exceed 1,200° C (2,200 °F).

- The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation, as well as oxidation that occurs during the accident.

- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1 percent of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- Calculated changes in core geometry are such that the core remains amenable to cooling.

- After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value, and decay heat is removed for the extended period required by the long-lived radioactivity.

- An analysis of a spectrum of LOCAs ensures that boric acid precipitation is precluded for all break sizes and locations.

- The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR Part 100 or 10 CFR 50.67. For applications under 10 CFR Part 52, reviewers should use SRP Section 15.0.3.

- The Three Mile Island Action Plan requirements for II.E.2.3, II.K.3.30, and II.K.3.31 in NUREG-0660, Volume 2, "NRC Action Plan Developed as a Result of the TMI-2 Accident," issued May 1980 (ML072470524) and NUREG-0737 "Clarification of TMI Action Plan Requirements" issued November 1980 (ML051400209), have been met.

DSRS Section 15.6.5 lists the following items that are included in the staff's review procedures for demonstrating conformance with these requirements, as well as review interfaces with other SRP/DSRS sections:

- Adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure consistent with DSRS Section 6.3.

- If core uncovery is not expected during the entire period of a LOCA, the staff should ensure that a significant number of fuel rods will not be damaged because of local dryout conditions. This may be demonstrated by showing that the limiting fuel rod heat flux remains below the CHF at a given pressure after depressurization has taken place.

- The parameters and assumptions used for the calculations were conservatively chosen. These choices include taking the initial power level as the licensed core thermal power plus an allowance of 2 percent to account for power measurement uncertainties, using the maximum LHGR, addressing permitted axial power shapes, and conservatively calculating the initial stored energy.

15.6.5.1.4 Technical Evaluation

Subsequent to completing the majority of the analyses reported in this SE, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 non-LOCA input. The staff's estimate of the impact of the modelling change(s) to DHRS for non-LOCA transients is about a **{{ }}** reduction in DHRS system heat transfer. The staff understands that these modifications to DHRS modelling were not also made to NRELAP5 models built according to the LOCA TR EM, but considers this acceptable given that DHRS typically operates for a very limited period of time in LOCA and IORV transients before ECCS actuates and dominates the NPM-20 depressurization and cooldown. As LOCA break size gets smaller, DHRS runs for longer times before ECCS actuates, however, the RPV CLL relative to the top of active fuel LOCA figure of merit is insensitive to the DHRS operating time prior to ECCS actuation because design basis breaks retain the leaked coolant in the containment and the minimum CLL relative to the top of active fuel occurs after—and is caused by—ECCS actuation. Furthermore, the SER for the LOCA TR includes L&C #13, which applies a **{{ }}** NRELAP5 fouling factor penalty to both sides of the DHRS tubes for peak containment pressure and temperature calculations.

The staff notes that for LOCA and IORV transients, while DHRS heat transfer is always credited, it is most needed to support containment response due to ECCS actuation arising due to very small breaks that leak into containment. Staff considers that the uncertainty associated with DHRS heat transfer performance, as stipulated by L&C #13 in the SER for the LOCA TR, encompasses the known adjustment to DHRS condenser headers, such as those described in the following SER section.

15.6.5.1.4.1 Evaluation Model

The applicant used the LOCA methodology discussed in Section 15.0.2 of this SER, including NRELAP5 Version 1.7 and revision 5 of the NRELAP5 NPM-20 plant model, to analyze the thermal-hydraulic response to the event. Note that NuScale reset the NRELAP5 input model revision numbers to 1 for the NPM-20 design that is part of the US460 SDAA. Section 15.0.2 of this SER also provides a summary of the staff's evaluation of the code and methodology, as well as the applicant's request for exemption from certain requirements in 10 CFR Part 50, Appendix K. Therefore, the staff's evaluation is based on the NRELAP5 Version 1.7 analysis and the corresponding information in FSAR Section 15.6.5.

The NRELAP5 input model for LOCA was developed from the applicant's base NPM-20 plant model, which was developed generically for both LOCA and non-LOCA transient analyses. The base modeling contained a set nodalization to model thermal-hydraulic fluid volumes and connecting heat structures for (1) the reactor vessel primary loop, including the lower plenum, core, riser (including one group of lower riser holes and four groups of upper riser holes at discrete elevation levels along the riser wall), PZR, SG primary side, RPV downcomer, with CVCS piping for RCS injection, discharge, and PZR spray lines, (2) reactor vessel secondary systems, including the helical coil SG secondary, steam lines and feedwater lines, (3) condensate and feedwater pumps, (4) the containment, (5) the reactor pool with DHRS included, and (6) ECCS valves and connections. The base model considers averaged reactor kinetics.

The LOCA NRELAP5 input model primarily differs from the general NRELAP5 basemodel by: **{{**
}}. The NuScale-specific CHF correlation NSPN-1 used in the LOCA EM was developed using a subchannel-like correlation in a lumped channel code to predict the minimum CHF ratio. NuScale's LOCA methodology uses different CHF options and core modeling criteria based on whether minimum CHF ratio or minimum level above TAF is being computed in the analysis. The NSPN-1 correlation is used **{{**
}} if the minimum CHF ratio is being computed, and **{{**
}} if the minimum CLL is being computed. **{{** **}}** developed based on benchmarks to the VIPRE subchannel code. The NRC staff reviewed this change within the LOCA TR review and found the **{{** **}}** and incorporation of the NSPN-1 CHF correlation acceptable for this application. The NPM-20 model also credits the DHRS system in the LOCA analysis. The staff reviewed the NPM-20 DHRS model and found that it added additional core cooling capacity and resulted in faster depressurization primarily in the smaller break size LOCAs.

The applicant requested an exemption in SDAA Part 7 from portions of Appendix K to 10 CFR Part 50. Section 15.0.2 of this SER evaluates this exemption request.

15.6.5.1.4.2 Input Parameters, Initial Conditions, and Assumptions

The staff reviewed the applicant's input parameters, initial conditions, and assumptions to assess the adequacy of the analysis model. This included checking input parameters and initial conditions used in the LOCA analysis to ensure that selections provide conservative assumptions for initial stored energy in the RCS, MCHFR, and minimum CLL in the core. The staff's review included considerations of the MPS signals, setpoints, and ranges (see these specified in FSAR Table 7.1-4, "Engineered Safety Feature Actuation System Functions". Of relevance to design basis LOCAs, the MPS ECCS actuation signals are 1) low RPV riser level, 2) low-low RPV riser level, and 3) "8 hour" reactor trip timer.

The key modeling biases for LOCAs are maximum initial core power of 102 percent of the rated power, maximum RCS average temperature to maximize RCS energy, elevated PZR pressure of 2,070 psia to maximize RCS flow out of the break, lowered PZR level of 52 percent to minimize available RCS inventory, and a reactor pool temperature of 60 °C (140 °F) to minimize heat transfer rate to the pool. The staff noted that the applicant selected the RCS average temperature 285 °C (545 °F) consistent with FSAR Table 15.0-6 and that the RCS flow was set to the minimum value in FSAR Table 15.0-6. The NRC staff finds this modeling acceptable because it maximizes the amount of stored energy within the RCS and maximizes the residency time of coolant in the core region, which is conservative for LOCA analyses.

With respect to ECCS actuation signals, NuScale accounted for built-in uncertainty in their thermal-hydraulics parameter monitoring sensors for ECCS actuation, which, for the NPM-20, are intended to signal MPS to actuate ECCS based on the equivalent level (equivalent height, i.e. the *collapsed* level) of liquid coolant above the core. There are two elevations above the core with thermal dispersion switch sets—at the top of the riser wall and about 6' down the riser inner wall—instrumenting the low and low-low riser level marks, respectively. For practical reasons, the thermal dispersion switches that generate the low and low-low riser signals correspond to vertical segments on the inner riser wall as opposed to discrete positions. This means that, for example, the low riser level mark is actually a 1' tall low riser level span in which the thermal dispersion switch(es) (there are four, at this elevation) are required to generate signals when the void fraction in this span is greater than or equal to 90 percent. The fact that the low riser level mark is a vertical span dictates that the low riser level mark signal is uncertain: 90 percent void fraction somewhere between 540" and 552" of CLL above the point of reference, which is the bottom of the reactor pool. In the same manner, the low-low riser level mark signal on very high void fraction in the primary coolant is to be generated by the four thermal dispersion switches in a vertical span between 460" and 472" above the same point of reference (bottom of reactor pool).

NuScale performed sensitivity analysis on the NPM-20's ECCS response due to the uncertainty in primary coolant riser level for the low riser level signal and concluded that the uncertainty in riser level is essentially an ECCS actuation time uncertainty and that the uncertain but constrained time of ECCS actuation on low riser level always results in acceptable ECCS performance. Specifically, NuScale performed the sensitivity by **{{**

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By audit of supporting calculations and corroboration from its own confirmatory analysis, the staff finds both that this simplified sensitivity study method is sufficient and that the results of the sensitivity study show acceptable results for the figures of merit of minimum CLL and MCHFR. The applicant and the staff both note that the ECCS actuation timing will be different for liquid space and vapor space breaks due to the vapor space breaks (e.g. high point vent line) leading to depressurization-induced moderate phase change swelling of the primary coolant generating sustained void fractions in the riser that are below 90 percent. In the most extreme challenges to the ECCS actuation signal viability, a vapor space break can evade triggering the low riser level signal due to interference from the T-5 interlock (T_{cold}<440F). In LOCA events where the low riser signal is evaded, the NRELAP5 simulation results still show that the NPM is adequately protected (i.e. figures of merit of no core uncover and no CHF are true) in one of two ways. In the first way, ECCS eventually actuates on the low-low riser level signal. In the second way, which arises due to small break area percentages on the HPV line, the NPM RPV depressurizes and cools down without ECCS actuation by way of DHRS operation and energy loss through the break into containment. In these unique NPM-20 HPV line small break LOCAs where both ECCS riser level actuation signals (low and low-low) do not occur, ECCS automatically actuates 8 hours after reactor trip. For the sole purpose of understanding the would-be event progression in its calculations, the applicant disabled the 8 hour ECCS actuation timer and showed that the NPM LOCA figures of merit were still acceptable more than 11 hours post reactor trip without ECCS actuation.

The modeling of the ECCS actuation signals on the riser levels described above satisfies LOCA TR L&C #11 “ECCS RPV Riser Level Instrument Setpoint Modeling.”

Since the NPM relies on natural circulation for reactor coolant flow and does not include external RCS piping, there are no large-diameter pipe breaks to consider. Consequently, the applicant postulated a LOCA spectrum of breaks at various locations in comparatively small piping within the RCS pressure boundary. The breaks analyzed focused on the CVCS injection and discharge, high-point vent, and PZR spray lines inside containment.

The staff reviewed the break locations considered by the applicant. As part of the LOCA break location and size evaluation, the staff identified that FSAR Chapter 15 does not include an analysis, nor includes a surrogate or other similar location yielding representative (or bounding) results, for the hypothetical loss of coolant at the reactor vessel-to-ECCS valve flange (i.e., vessel to piping system boundary, but upstream of the venturi), or in the CVCS lines outside the containment between the CNV and CIVs. Applying the LOCA EM to a non-mechanistic break at any of these locations will potentially produce greater consequences than what is currently analyzed within the design-basis. NuScale requested an exemption from 10 CFR 50.46 and GDC 35 in ML25057A492. Section 15.6.5.3 of this SER evaluates this exemption.

Additionally, the staff confirmed through audited documents that the inner diameter of the CRDM nozzle remains bounded by the flow area of the inadvertent opening of a reactor vent valve venturi analyzed in FSAR Section 15.6.6. The staff concludes that the applicant considered the break spectrum considered in accordance with 10 CFR Part 50, Appendix K, paragraph I.C.1.

The staff additionally reviewed initial parameter values and biases including initial RCS conditions and reactivity coefficients, to ensure that the applicant selected conservative values for the analysis. The staff agreed that the parameters are sufficiently conservative for LOCA initial conditions.

The applicant did not credit operator action to mitigate a LOCA event. The applicant also considered the single failures of an RVV or RRV to open. Since the maximum rate of depressurization during ECCS activation yields the minimum core collapsed levels, the assumption of no single failure of ECCS valves to open should generally provide the limiting results. However, the staff notes that the applicant’s definition, in TR-0516-49422-P, Revision 3, of “collapsed level” is a volume based accounting of the liquid in the core, riser, and bypass regions and is less limiting than the conventional method that uses an axial node length summation of the liquid fractions. The volume based approach more heavily weights liquid that is not physically in the core. Also, the staff notes that the conventional method was used in the NRELAP5 RPV level calculations to benchmark the NuScale Integral System Test Facility (NIST) test data. Therefore, staff considers the NuScale collapse level prediction as a volume-averaged riser level rather than a minimum core collapsed level. Additionally, staff believes this method may mask depressed levels in the hot channel however, the margins for minimum CLL for NPM-20 design is sufficiently large that the difference in the limiting level results is no longer significant. In addition, since MCHFR remains a figure of merit for the event, in addition to CLL, if the hot channel level is depressed, that would also be reflected in the MCHFR figure of merit.

The loss of normal AC power and EDAS was determined to conservatively worsen the RCS thermal conditions at and after LOCA event initiation. When the EDAS is lost, control rods insert and both RVVs begin to open—without any delay or any time spent with DHRS cooling—causing the maximum depressurization rate for a LOCA, with the level swelling in the riser, resulting in the maximum break mass and enthalpy flow rates and maximum inventory loss because the RPV stored energy is at its maximum value.

The staff audited (ML24211A089) the applicant's LOCA break spectrum calculations and reviewed the results summarized in FSAR Table 15.6-12, "Loss-of-Coolant Analysis - Discharge Line Break Spectrum - Minimum Critical Heat Flux Ratio Sensitivity to Loss of Power," that lists the most limiting initial conditions, single failures, and loss of power assumptions to confirm that they led to the most limiting LOCA MCHFR results. LOCA breaks outside the CNV that are not isolated by the CIVs are not evaluated in this break spectrum calculation, but are addressed in FSAR Section 15.6.5.6.2 and evaluated in Section 15.6.5.3 of this SER. The audited material generally supports the discussions in the FSAR, and the staff confirmed that the input parameters and initial conditions listed in the FSAR are suitably conservative and result in the most limiting for each of the respective acceptance criteria.

15.6.5.1.4.3 Results

The NPM break spectrum is separated into two categories: (1) liquid space breaks (CVCS injection line or discharge line) and (2) steam space breaks (high-point vent line and PZR spray supply). A steam space break initiates a blowdown of the RCS inventory into the CNV from the top of the RPV. A liquid space break causes blowdown of the RCS inventory into the CNV from the RPV downcomer or riser. The event progresses much faster if the break is from a liquid space.

The applicant determined the 100 percent CVCS discharge line break at 102 percent RTP to be the limiting LOCA for MCHFR. The CVCS discharge line connects from the bottom of the upper downcomer region and CVCS system through the CNV and into the CVCS system. The limiting case assumed a loss of AC and EDAS power and the immediate ramp opening of both RVVs. RRVs are opened later after the pressure difference between RPV and CNT is below the IAB release pressure. Core voiding resulting from the combined break opening and RVV opening causes short term flow decrease when the core heat flux is still high, which leads to a sudden reduction in MCHFR. The MCHFR of 1.35 is reached within 1 second. The staff reviewed the accident progression and agrees that this 100 percent CVCS discharge line break case maximizes the blowdown rate into the containment, which leads to the minimum CHFR of any postulated LOCA in the NPM-20. Within the first few seconds after the break, flow stagnation, core voiding, and high fuel stored energy cause the minimum departure from nucleate boiling (MCHFR) to occur. If electrical power is not assumed to be lost with the break, then the RVVs do not begin to open at the same time that the postulated break occurs and the MCHFR would be higher. Additionally, limiting CHF scenarios may occur for very small breaks when these are taken to be initiating events where flow stagnation combines with a very brief increase in RCS pressure that is caused by MPS tripping the reactor off and halting feedwater flow to the steam generators—constituting a simultaneous loss of heat sink with ongoing thermal decay power in the shutting down core. Based on the staff's evaluation, small break LOCAs did not provide the bounding MCHFR value.

After blowdown, the core thermal energy is steadily discharged to the reactor pool by conduction through the containment wall as vaporized primary coolant condenses in the containment. A fraction of total heat transfer out of the RPV also concurrently occurs through the DHRS thermal circuit. At this point, the gradual cooldown and depressurization continue, and the LOCA event transitions to the post-LOCA LTC phase.

The minimum CLL above the top of the active fuel was determined to be the 100 percent CVCS injection line break at 102 percent RTP for the LOCA spectrum of breaks. The CVCS injection line connects the lower riser region and CVCS through the CNV and RPV into the CVCS system. The limiting case assumed a single failure of one ECCS division to open (i.e., one RVV and one RRV) without loss of power. This causes only one RVV to ramp open at ECCS actuation and only one RRV to open later after the IAB pressure difference is reached. The reactor is tripped on CNV pressure early resulting in containment isolation, isolation of feedwater, main steam, and actuation of DHRS. The blowdown is extended and RRV opening is delayed since the ECCS capacity is limited to one RVV, causing a larger inventory loss to the CNV. Additionally, ECCS recirculation is limited to one RRV and consequently more of the RCS inventory will reside in the CNV. Even with this single failure assumption, the minimum CLL above the top of the core was more than 8 feet, which is significant. The staff reviewed the accident progression and agrees that the 100 percent CVCS injection line break case maximizes the blowdown period and rate resulting in the limiting minimum CLL for LOCA.

For postulated LOCAs the minimum predicted CHFR is approximately 1.35, which is above the LOCA CHFR safety limit of 1.20. Since the MCHFR remains above the safety limit, the applicant concluded that the acceptance criteria for LOCA are met for the maximum PCT, total percentage of fuel cladding oxidation, amount of hydrogen generation, and maintenance of coolable geometry of the reactor core. The staff reviewed the plotted results and the sequence of events and finds that they are consistent with the event description and progression of ECCS responses and support the applicant’s assertion that acceptance criteria are met. The staff review of the LOCA CHF limit, which is documented in the LOCA EM TR and described in Section 15.0.2 of this SER, concluded that the correlations used are adequate to reasonably predict CHF. Further, staff noted since the CLL above TAF is significant, any potential momentary heat up would be quickly quenched by the weight of this high water column. The staff finds that there is adequate margin to CHF and the minimum CLL.

The post-LOCA LTC assessment described in FSAR Section 15.6.5 evaluates the ECCS LTC capability of the NPM after a successful initial short term response to the DBEs discussed in Section 15.6.5.1 of this SER up to a period of 72 hours after the events. The assessment does not credit normal AC power, the non-safety-related DC power system, or any operator action. The methodology used is based on the XPC EM reviewed by the staff in Section 15.0.2 of this SER, and which is incorporated by reference in FSAR Table 1.6-2, supplemented by ML24346A132. This TR addresses LTC phenomena related to both LOCA and non-LOCA events. The staff evaluates the SDAA calculations performed with the XPC EM in SER Section 15.0.5 and addresses LTC analytical results for events initiated from the LOCA (SER Section 15.6.5.1) event or IORV event (SER Section 15.6.6), as well as non-LOCA events.

15.6.5.1.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.6.5.

15.6.5.1.6 Conclusion

The staff reviewed and audited the applicant's break spectrum calculations that investigated the most limiting initial conditions, single failures, and loss of power assumptions to confirm that they led to the most limiting LOCA results.

Additionally, the staff performed a preliminary confirmatory analysis using the NRC's TRACE thermal-hydraulic code for the 100 percent CVCS injection line break case, which demonstrated that the applicant's methodology produces reasonably conservative results. The staff found that the overall NRELAP prediction, with embedded conservatisms required by 10 CFR Part 50, Appendix K, is more conservative than TRACE in predicting the peak containment pressure and core collapsed level.

Based on the LOCA analysis results reviewed, and pending closure of the above noted open items, the staff determined that the requirements in 10 CFR 50.46, 52.137(a); GDCs 13, 35, and 27; 10 CFR Part 50, Appendix K; and 10 CFR Part 100 regarding the LOCA event have been met, except for the exemptions noted in Sections 15.0.2.4.1 and 15.6.5.3 of this SER. The staff found that the loss-of-coolant analysis resulting from a spectrum of postulated piping breaks within the RPV pressure boundary is acceptable and demonstrates compliance with 10 CFR 50.46 criteria with a large margin due to the unique design of the ECCS where RCS coolant is cooled and recirculated from CNV and downcomer back to the core.

15.6.5.2 Long-Term Cooling after a Loss-of-Coolant Accident

The NRC staff's evaluation of LTC after a LOCA is contained in Section 15.0.5 of this SER.

15.6.5.3 Beyond Design Basis Event Breaks

15.6.5.3.1 Introduction

The staff raised two issues during its review of NuScale's LOCA break spectrum analysis, specifically regarding two locations and whether LOCAs need to be considered DBAs. NuScale and the staff discussed these issues through the audit and referred to the issues as High Impact Technical Issues (HITIs) 2 and 10. For simplicity, this section of the SER may also refer to these two areas as "subject locations." The applicant treated the locations identified below as beyond-design-basis (BDB) events. This section contains the staff's evaluation of this treatment and corresponding exemption request under 10 CFR 50.12 to the regulatory requirements of 10 CFR 50.46(a)(1)(i) and GDC 35.

<u>ECCS Flanged Connection</u>

HITI 2 relates to the ECCS valves that are bolted directly to a flange on the reactor vessel. A flow-restricting venturi is inserted in the main valve body to limit blowdown flow when the valves open, slowing the depressurization rate and mitigating the consequences. The accident analysis in FSAR Section 15.6.6 only evaluates inadvertent opening of the ECCS valves, not the failure of the ECCS valve flange, which would be a larger flow path with different consequences. SDAA Part 7, Section 18 contains NuScale's exemption request for these connections. Section 15.6.5.3.4.3 of this SER contains the staff's evaluation of this exemption request.

<u>CVCS Piping System Between the CNV and CIVs</u>

HITI 10 relates to the four CVCS lines that penetrate the containment and connect to the RCS. Both CIVs on each line are located outside containment. NuScale only evaluates breaks outboard of the two CIVs in FSAR Section 15.6.2—i.e., those that could be isolated by the valves—not within the piping system between the CNV and the valve assembly. SDAA Part 7, Section 18 contains NuScale's exemption request for these connections. 15.6.5.3.4.3 of this SER contains the staff's evaluation of this exemption.

15.6.5.3.2 Summary of Application

FSAR Section 15.6.5.6, "Beyond-Design-Basis Event Breaks," provides the description and results of the breaks considered as BDB breaks. The applicant stated that the LOCA EM TR is used to analyze these BDB breaks, with some allowable alternative assumptions listed in FSAR Table 15.6-18, "Acceptance Criteria and Allowable Alternative Assumptions for Evaluation of Beyond-Design-Basis Breaks." The criteria used relative to core cooling, containment response and dose are: 1) MCHFR >1.15 AND liquid level above the top of active fuel, OR PCT <2,200ºF, 2) containment pressure <1,200 psia AND containment temperature <600ºF, and 3) EAB dose <25 rem TEDE AND LPZ <25 rem TEDE.

For the ECCS flange breaks, the FSAR states that the core cooling criteria of MCHFR and liquid level are met without the need to evaluate PCT. Similarly, the containment pressure and temperature criteria are met, as well as the EAB and LPZ dose criteria (no fuel failures are predicted).

For the CVCS breaks, the FSAR states that the core cooling criterion for PCT is met. For injection line breaks, the PCT does not increase above its steady state value. For high point vent (HPV) breaks, the PCT does not increase above its steady state value for the first 24 hours without active mitigation. With mitigation via inventory addition PCT does not increase above its steady-state initial value beyond 24 hours. Similarly, the containment pressure and temperature criteria are met, and the applicant also noted that since these breaks are outside containment, they do not directly impact containment pressure and temperature. The EAB and LPZ dose criteria are also met, and the applicant noted no fuel failures are predicted.

15.6.5.3.3 Regulatory Basis

The regulatory basis for granting an exemption to applicable regulatory requirements is provided in 10 CFR 50.12. The following regulatory requirements do not provide allowance to treat their relevant postulated accidents as beyond-design-basis; however, the staff utilized the requirements because they contain technically relevant content and guided the staff in its evaluation for the treatment of LOCAs from specific locations in the NuScale US460 design and determine appropriate analytical acceptance criteria.

- 10 CFR 50.46, as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the RCPB.

- 10 CFR Part 50, Appendix K, which provides the required and acceptable features of ECCS EMs.

- GDC 35, as it relates to demonstrating that the ECCS would provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

- GDC 38, as it relates to demonstrating containment pressure and temperature following any loss-of-coolant accident is rapidly reduced and maintained acceptably low.

- 10 CFR 52.137(a)(2)(iv), as it relates to the ability of plant systems to mitigate the radiological consequences of accidents and the evaluation and analysis of the offsite radiological consequences with fission product release.

15.6.5.3.4 Technical Evaluation

15.6.5.3.4.1 Evaluation Model

In FSAR Section 15.6.5.6.1, “Connections between Reactor Pressure Vessel and Emergency Core Cooling System Valves,” and Section 15.6.5.6.2, “Connections between Containment Vessel and Chemical and Volume Control System Containment Isolation Valves,” the applicant specifies that the LOCA EM, with modified acceptance criteria, is used to evaluate losses of coolant from the ECCS flanged connections and CVCS lines between the CNV and CIVs, respectively. FSAR Table 15.6-18 prescribes the acceptance used for the analyses of these LOCA scenarios. The staff finds that the acceptance criteria identified in FSAR Table 15.6-18 aligns with fuel cooling, containment, and radiological dose acceptance criteria for the design-basis of the facility and is therefore acceptable. The staff finds the applicant’s evaluation method, including the associated base model, acceptable for use in these scenarios because it is based on the LOCA EM which has been developed, validated, and reviewed for the NPM-20 design-basis LOCA analyses. Section 15.0.2 of this SER contains the staff’s evaluation of the LOCA EM code and associated base model. Section 15.6.5.3.4.3 of this SER contains the staff’s determination regarding the acceptability of treating these LOCA scenarios as BDB events. The staff notes that the use of PCT is an appropriate figure of merit for demonstrating acceptable core cooling; however, this evaluation method has not validated NRELAP5 to calculate PCT results. The applicant has not requested approval of, nor is the staff approving, the post-CHF models in NRELAP5. Nevertheless, for the purposes of a risk-informed approach supporting the staff’s evaluation of the requested regulatory exemption, unvalidated PCT results can provide insights into the magnitude of consequences, and due to the large margins to the acceptance criteria, the staff finds this approach reasonable for this beyond design basis event. As discussed in Section 15.6.5.3.4.3, the results for these LOCA scenarios indicate significant margin to PCT acceptance criteria. Accordingly, evaluation of the post-CHF model was not warranted at this time based on the current design and analysis.

15.6.5.3.4.2 Input Parameters, Initial Conditions, and Assumptions

In FSAR Table 15.6-18, the applicant details the alternative assumptions used for some of the analytical inputs for these BDB LOCA scenarios that depart from what is specified in the LOCA EM TR. In general, the staff considers that these allowed alternative assumptions align with nominal, best estimate, conditions. The staff performed an audit of the analyses for these LOCA scenarios (ML24211A089) and observed that some parameters such as the assumed values for decay heat and scram worth may be moderately non-conservative. **{{**

}}. Based on

independent staff confirmatory analysis, the staff concludes that use of a more appropriate value could impact the timing of the sequences but will have overall a minimal effect on the physics. The staff also confirmed that the potential for cliff edge effects will be appropriately mitigated. For limiting scenarios, the staff observes that areas of non-conservatism, uncertainty, and cliff edge effects are balanced by manual operator action to provide active mitigation via inventory addition. FSAR Section 15.6.5.6.2 identifies the CVCS, and containment flooding and drain system (CFDS) as inventory addition systems. The parameters for the CVCS makeup pumps are provided in FSAR Table 9.3.4-1, “Chemical and Volume Control System/Module Heatup System Major Equipment with Design Data and Parameters”. The staff assumption for the capability of the CFDS pump is based on a regulatory audit of thermal-hydraulic calculations supporting the Level 1 PRA. Specifically, the calculation assumed the CFDS makeup flowrate is **{{ }}**. Specific to the CVCS line scenarios, the 8-hour ECCS actuation timer was not credited and the staff acknowledges that this provides additional conservatism to the analysis results for cases where ECCS actuation due to low riser level is not achieved within the first 8 hours of the transient. Additionally, the analysis considered loss of electrical power assumptions consistent with the design-basis LOCA EM, which that staff finds appropriate and acceptable for these scenarios. Considering the alternative assumptions as a whole, the staff finds these inputs to be suitable for analysis of these BDB LOCA scenarios.

15.6.5.3.4.3 Results

Core Cooling and Containment Consequences of a Failure at the ECCS Valve Flanges

The staff reviewed the applicant’s results submitted in letter dated March 20, 2025, (ML25079A197 (proprietary), ML25079A196 (non-proprietary)) and provided in reference 15.6-5 in FSAR Section 15.6.7. The limiting results for core cooling and containment response occurred for the **{{ }}.** The applicant calculated the lowest MCHFR as **{{ }}**, which is well above the limit of 1.15. Further, the applicant calculated that the liquid level would not decrease below **{{ }}** above the top of active fuel. Peak containment pressure was **{{ }}** and peak containment temperature was **{{ }}**. The staff agrees with the applicant’s conclusions that core cooling and containment integrity are maintained for this ECCS valve flange break case. The staff also audited the applicant’s calculations relative to core cooling and containment pressure and temperature figures of merit, and confirmed the methodology used, including inputs and assumptions, are consistent with those stated in FSAR Section 15.6.5.6.1.

Core Cooling and Containment Consequences of a Failure of CVCS line occurring between the outside of the CNV and the CIVs

The staff reviewed the applicant’s results submitted in letter dated March 20, 2025, (ML25079A197 (proprietary), ML25079A196 (non-proprietary)) and provided in reference 15.6-5 in FSAR Section 15.6.7. The limiting results for collapsed liquid level occurred for the 100 percent HPV line break case with the ECCS riser level actuation at the low end of the range and is presented as **{{ }}** of liquid level above the top of active fuel. However, based on a staff audit (ML24211A089) this result is calculated **{{**

}}, the staff observed the NPM-20 active fuel axial extent is immersed in a two-phase water mixture which, according to NRELAP5 calculations, is always of sufficient mass—and replenished by continuous gravity driven feed from the downcomer—such that the fuel cladding temperature does not increase beyond the initial, pre-transient full power operation value. NRELAP5 simulations show that the core/cladding rejects heat by vaporizing the coolant that is passively flowing into the core from the ECCS operation.

FSAR Section 15.6.5.6.2 states that if active mitigation is considered beginning no earlier than 30 minutes, PCT does not increase above its steady-state initial value regardless of failure location or size. During the regulatory audit, the applicant clarified that this statement **{{ }}** was not based on the calculation of record that uses the evaluation method described in Section 15.6.5.3.4.1 of this SER. The staff finds this acceptable because the purpose of this statement in the FSAR is to describe and document the active mitigation capabilities of the design and is not relied on to provide the documented results for the limiting scenario for PCT.

For containment pressure and temperature, the limiting case is **{{ }}**. The applicant calculated peak containment pressure to be **{{ }}** and peak containment temperature to be **{{ }}**, both of which are well within the acceptance criteria.

The staff audited (ML24211A089) the applicant's calculations relative to core cooling and containment pressure and temperature figures of merit, and confirmed the methodology used, including inputs and assumptions, are consistent with those stated in FSAR Section 15.6.5.6.2. In addition, the staff performed independent confirmatory analysis. The approach in the staff's confirmatory analysis followed more closely with the applicant's current LOCA evaluation methodology, considering various pool temperatures, decay heat standards and multipliers, and ECCS actuation setpoints/delays to better understand parametric sensitivities, whereas the applicant used normal or best-estimate parameters. Accordingly, the staff's use of the LOCA EM's range of values for sensitive parameters led to more severe core uncoveries, including cases with NRELAP5-calculated fuel heat up. Nonetheless, the staff believes the NuScale US460 design has reasonable design capabilities yielding a mitigating strategy to account for uncertainties in the assumptions than can result in potential cliff edge effects; specifically, the consequences of these CVCS line breaks can be mitigated by taking manual operator action to actuate ECCS earlier in the transient and add coolant to the containment from the containment flood and drain system or to the RPV if there is an intact CVCS line that is capable of injecting into the RCS. Considering the alternative assumptions as a whole, the staff finds these inputs to be suitable for analysis of these BDB LOCA scenarios.

Radiological Consequences of a Failure at the ECCS Valve Flanges

As discussed above, a failure of the ECCS valve flanges does not result in core damage, and coolant inventory loss is contained inside of the containment. Therefore, no significant release of radioactive material would occur. In terms of the radiological consequences, other accidents discussed in FSAR Chapter 15 bound this accident regarding doses to the public, doses to the control room, and doses in and around the reactor containment. Therefore, the staff finds it to be acceptable.

Radiological Consequences of a Failure of CVCS line occurring between the outside of the CNV and the CIVs

The radiological consequence analysis of CVCS line break outside the CNV but inside of the CIVs is described in FSAR Section 15.6.5.6.2 and FSAR Table 15.6-18. The postulated event results in a non-isolatable loss of reactor coolant which bypasses reactor containment. As discussed in FSAR Section 15.6.5.6.2, the NuScale SDA design maintains adequate core cooling following this event and does not result in fuel damage or a gap release of the radioactive material between the fuel pellet and the cladding of the fuel rod. As a result, the radiological consequences are limited to a release of the primary coolant through the break, with an assumed coincident iodine spike that raises the equilibrium appearance rate by a factor of 500 for 8 hours.

Staff audited NuScale's calculation to determine the dose from the coolant release as a result of the worst case CVCS line break between the outside of containment and inside the isolation valves. The accident results in nearly **{{ }}** of coolant release, **{{ }}** compared to the small line break accident discussed in FSAR Section 15.0.3.7.1. Most of the release occurs within the first two hours of the accident. Even when considering 72-hour power loss with control room habitability system activation, NuScale calculated a dose of less than **{{ }}** in the control room. The EAB and LPZ dose were both calculated to be less than **{{ }}**. This is significantly less than the dose criteria for the main control room and the dose criteria for the EAB and LPZ. When comparing these results to the results of the small line break discussed in FSAR Table 15.0-10, the staff finds these results to be consistent with staff analysis given the quantity of coolant released and the timing of the release. Since the event results in a non-isolatable release from containment, the most significant factor in ensuring that dose consequences following this event are consistent with the values specified above and that there are not potentially significant offsite radiological dose consequences and potentially significant radiological dose consequences to control room operators and other plant personnel, is that core damage and a gap release from the core are both prevented during this event. Based on this and the results of the calculations audited by the staff, the staff finds there is reasonable assurance that the radiological consequences in the MCR and at the EAB and LPZ are within applicable dose criteria and acceptable.

<u>Exemption from 10 CFR 50.46(a)(1)(i) and GDC 35</u>

The staff reviewed the applicant's request for exemption from certain requirements of 10 CFR 50.46(a)(1)(i) and GDC 35, as described in SDAA Part 7, Section 18 (ML25057A492), related to the exclusion of ECCS cooling performance analysis for losses of coolant in two areas of the NPM: the ECCS bolted flange connections, and CVCS piping systems between the CNV and associated CIVs. The specific requirements in 10 CFR 50.46(a)(1)(i) and GDC 35, from which the applicant requested exemption are the following:

- 10 CFR 50.46(a)(1)(i) in relation to evaluation of postulated loss-of-coolant accidents
- 10 CFR, Appendix A, GDC 35 in relation to selection of LOCA break spectrum

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 52. As 10 CFR 52.7 further states, the Commission's consideration will be governed by 10 CFR 50.12, "Specific exemptions," which states that an exemption may be granted when: (1) the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security; and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request.

The US460 design contains design features with the purpose of reducing the overall risk of LOCAs and LOCA-like events. These attributes include elements such as a significant volume of water in the RCS compared to thermal power; elimination of penetrations near or below the core; shortened or eliminated pipe length; the ability to use passive core cooling capabilities for LOCAs inside containment; an inventory retention strategy via the containment isolation system; slower accident progression and minimal operator actions; the capability to provide inventory addition via diverse high and low pressure systems; and other enhanced design features.

The staff performed its evaluation of the requested exemption based on the information provided in the US460 SDAA and the staff confirmed the applicant's results through regulatory audits and independent confirmatory analyses. The staff's safety determination considers the overall safety significance of LOCAs for the US460 design at the subject locations and is based on a set of integrated risk-informed principles, such as, likelihood, consequences, safety margin, defense in depth, and performance monitoring.

Authorized by Law

The NRC staff has determined that granting of the applicant's proposed exemption will not result in a violation of the Atomic Energy Act (AEA) of 1954, as amended, or the Commission's regulations because, as stated above, 10 CFR Part 52, allows the NRC to grant exemptions. The staff also determined that granting the applicant's proposed exemptions will not result in a violation of the AEA. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is authorized by law.

No Undue Risk to Public Health and Safety

The staff review of the exemption request to determine if the exemption would present an undue risk to the public health and safety (10 CFR 50.12(a)(1)) is described below. In its exemption request, the applicant stated that it will not impact the consequences of any design-basis event and will not create a new accident precursor. The applicant also performed alternate analysis of the excluded LOCA locations and demonstrated that mitigation capabilities exist if a loss of coolant at one of the excluded locations does occur, and the dose consequences would remain within regulatory limits. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that, based on the holistic approach to reduce the LOCA risk though enhanced prevention balanced with mitigation capability, the exemption poses no undue risk to the public health and safety.

Consistent with Common Defense and Security

The proposed exemption does not affect design, function, or operation of any structures or plant equipment that is necessary to maintain a secure plant status. In addition, the proposed exemption has no impact on plant security or safeguards procedures. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security is not impacted by this exemption.

Special Circumstances

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of 10 CFR 50.46, which includes prescriptive criteria that support implementation of GDC 35, is to ensure the ECCS is appropriately sized such that the core will remain sufficiently cooled following the arbitrary or hypothetical loss of coolant from the reactor coolant pressure boundary. 10 CFR 50.46(a)(1)(i) pertains to the calculational framework and spectrum of LOCAs to be analyzed which intends to ensure that the most severe LOCAs are calculated, and that uncertainty is accounted for such that there is a high probability abundant core cooling will be provided. During promulgation of 10 CFR 50.46, the Commission ensured a strong balance between prevention of accidents and mitigation capabilities using implicit probabilistic considerations. The Commission acknowledged that traditional reactor pressure vessels are sufficiently robust such that the ECCS need not be sized for catastrophic failure or rupture of the vessel. Likewise, it is understood that only those locations that result in the most severe consequences need to be analyzed and not every location or size must be explicitly analyzed. Codification to define LOCAs as “breaks in pipes” within 10 CFR 50.46 was based on the longstanding practice of traditional large light-water reactor designs to size the ECCS due to failures in RCS piping. This yields a simplified bounding approach that also addresses other non-limiting component failures and break sizes in the reactor coolant pressure boundary. For example, designing an ECCS to an arbitrary loss of coolant from the largest pipe in the RCS inherently covered uncertainties and unknown failure mechanisms for components or connections (e.g., steam generator manways, valve bodies, reactor vessel head) eliminating the need to explicitly analyze them. This robust approach to LOCA evaluations ensured mitigation capabilities exist for a sufficiently broad spectrum of failures within the reactor coolant pressure boundary. In combination with means of prevention via application of highest quality standards, defense-in-depth is maintained, and reasonable assurance of adequate protection of public health and safety is provided.

To maintain consistency with this historical interpretation and implementation of 10 CFR 50.46, including the definition of a LOCA contained in 10 CFR 50.46, and the previously approved NuScale US600 design, the staff focused its LOCA evaluation review of the NuScale US460 design on only those limiting locations for components and connections within the reactor coolant pressure boundary that are pipes or function as pipes (i.e., those locations where the primary function is to transfer fluid). The staff emphasizes that the NuScale US460 design contains numerous bolted penetrations as part of the reactor coolant pressure boundary (e.g., in-core instrumentation penetrations, maintenance access ports, etc.) which are substantially larger than the ECCS bolted flanged connections and have not been designed with enhanced design and programmatic controls or analyzed as LOCA locations. Based on the historical interpretation and implementation described above, only those components that transfer fluid and function as pipes have been considered within the scope of locations subject to LOCA considerations for the US460 design.

The locations described in Section 15.6.5.3.1 of this SER are two areas for which application of the design-basis LOCA EM would calculate more severe consequences than the analyzed design-basis break spectrum evaluated in Section 15.6.5.1 of this SER. Accordingly, NuScale developed an approach to balance prevention (likelihood of failure) and mitigation (consequences of failure). Specifically, the approach utilizes mechanistic considerations and design enhancements to provide justification that a loss of coolant from the subject locations remains highly unlikely. In addition, consequence analyses were performed using a similar methodology to the LOCA EM with limited modification to certain assumptions, **{{ }},** in order to calculate best-estimate results for a scenario that assumes all safety equipment functions as designed. Section 15.6.5.3.4.1 and Section 15.6.5.3.4.2 of this SER contains the staff's evaluation of the methodology and assumptions used in the consequence analysis.

Related to the bolted ECCS valve flanges, the likelihood of failure is sufficiently low given provisions for design, inspection, and monitoring of the flange locations. NuScale has provided for augmented inspections, lower design stress limits compared to code-required stress and cumulative usage factors, high-strength bolting, and ultrasonic examination at service intervals. The design also contains containment leakage detection, controlled by technical specifications, and automated protection system actuation based on containment pressure increases. These provisions aid in detecting bolting failures before they would proceed to gross rupture. Sections 3.13.4.2, 3.13.4.4, 4.5.1.4.1 of this SER contain the staff's detailed evaluation of these design provisions. Section 15.6.5.3.4.3 of this SER provides the staff's evaluation of the results of the applicant's thermal-hydraulic analysis and concludes that the figures of merit for MCHFR, collapsed liquid level, containment response, and dose consequences are within the acceptance criteria limits.

Similarly, the likelihood of failure of the CVCS piping system between the CNV and CIVs is limited by design and inspection of the piping and weld locations. Specifically, NuScale imposed conservative stress limits compared to code-required stress and cumulative usage factors, prescribed inservice inspection, and chemistry controls and monitoring to prevent or detect degradation. SER Section 3.6.2.4.1.1 contains the staff's detailed of these design provisions. Section 15.6.5.3.4.3 of this SER provides the staff's evaluation of the applicant's thermal-hydraulic analysis and staff confirmatory analysis which concludes that the figures of merit for PCT, containment response, and dose consequences are within the acceptance criteria limits. The staff notes that **{{ }}**. Therefore, the staff finds that, given other factors considered in authorizing an exemption, this deviation from traditional practice is acceptable.

The staff finds that the analytical approach for treating these narrowly scoped locations as not within the design-basis to be acceptable based on the information provided in FSAR 15.6.5.6.1 and the elements that compensate for parameter and response uncertainty, non-conservatisms, and cliff edge effects; these include, but are not limited to, the expected availability of active mitigation via operator action and inventory addition, not crediting the 8-hour ECCS actuation timer, and consideration of all electrical power scenarios.

While the SDAA Part 7, Exemption 18 focuses on weld failure rates for the CVCS lines, the components themselves within the piping system are also subject to consideration under 10 CFR 50.46. The staff does not approve any numerical failure frequency as the basis for treating these limited LOCA locations as not within the design-basis. Instead, the staff bases its findings on the unique features of the US460, including those that are specifically intended to reduce the overall module risk of LOCAs, in combination with the qualitative design and programmatic enhancements of the subject locations that are complemented with the demonstration of mitigation capability to ensure low consequences. As described above, the staff finds the thermal-hydraulic analyses performed by the applicant as an acceptable surrogate for component failures not explicitly described, such as the CVCS nozzle safe ends and CIV test fixtures. In addition, the staff's exemption findings are applicable to gross ruptures or failures in the subject locations, and notes that small breaks or leaks in these locations as part of normal operation, including AOOs, must still be treated within the design-basis.

This exemption request is focused on the requirements contained in 10 CFR 50.46(a)(1)(i) related to the specifications for which postulated LOCAs must be calculated in accordance with an acceptable EM demonstrating adequate ECCS cooling performance. Because LOCAs at the subject locations are excluded from the design basis under 10 CFR 50.46, a LOCA at these locations would also be excluded from the design basis for the purposes of all other design requirements that consider the consequences of LOCAs. Examples of these requirements include 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," and GDCs 19, 38, 41, 44, and 50. Likewise, this exemption request does not exempt a future applicant or licensee from other regulatory requirements applicable to these licensing basis events or the ECCS. For example, the components of the subject locations and the accident analyses of beyond design basis losses of coolant from them, including the associated methods, are evaluated in the FSAR and shall be subject to the change control process of 10 CFR 50.59, or equivalent.

Based on the above the staff finds that the underlying purpose of the rule is met by the applicant through an approach that ensures gross ruptures and failures resulting in a LOCA at the subject locations remains highly unlikely through design enhancements while maintaining a LOCA analyses that demonstrates mitigation capability to cool the core, protect containment integrity, and minimize dose consequences.

The applicant stated in SDAA Part 7, Exemption 18, that special circumstances described in 10 CFR 50.12(a)(2)(iii), related to undue hardship, and 10 CFR 50.12(a)(2)(vi,) related to any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption, are present. However, where the staff finds that other special circumstances are present in accordance with another provision of 10 CFR 50.12(a)(2), a staff finding on whether special circumstances are present in accordance with 10 CFR 50.12(a)(2)(iii) or 10 CFR 50.12(a)(2)(vi) are not necessary for the exemption to be granted. Because the staff finds that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), the staff makes no finding regarding the presence of special circumstances described in 10 CFR 50.12(a)(2)(iii) or 10 CFR 50.12(a)(2)(vi).

15.6.5.3.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.6.5.

15.6.5.3.6 Conclusion

Based on the information documented in the NuScale US460 SDAA, and for the reasons given above, as set forth in 10 CFR 50.12(a), the staff concludes that the proposed exemption requested in SDAA Part 7, Section 18 regarding requirements stated in 10 CFR 50.46(a)(1)(i) and GDC 35 related to the selection of the LOCA break spectrum to ensure the most severe LOCAs are analyzed using an acceptable evaluation model are authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, the special circumstances in 10 CFR 50.12(a)(2)(ii) are present, in that the design-basis treatment of a loss of coolant accident at the subject locations in the particular circumstances is not necessary to achieve the underlying purpose of these rules. Therefore, the staff concludes that an exemption to the requirements of 10 CFR 50.46(a)(1)(i) and GDC 35 for consideration of gross rupture and failure from the subject locations is justified and can be authorized. In addition, the staff finds the applicant’s consequence analysis documented in FSAR 15.6.5.6 for losses of coolant from the bolted ECCS valve connection and CVCS lines between the CNV and CIV to be acceptable given their treatment as beyond-design-basis events.

15.6.6 Inadvertent Operation of the Emergency Core Cooling System

15.6.6.1 Introduction

A spurious signal, hardware malfunction, or operator error can cause an ECCS valve to inadvertently open, resulting in a loss of reactor coolant from the RPV and an RPV depressurization. This event is classified as an AOO.

15.6.6.2 Summary of Application

The applicant provided an event description in FSAR Section 15.6.6, “Inadvertent Operation of Emergency Core Cooling System.”

An inadvertent operation of ECCS causes a reactor vessel depressurization and decrease of reactor vessel coolant inventory that could be caused by a spurious electrical signal, hardware malfunction, or operator error. The inadvertent opening of more than one RRV valve is not considered as discussed in Section 15.6.6.4.1 of this SER due to the design of the IAB feature (Section 15.0.0.5 of this SER discusses single failures). Because the RVVs do not have the IAB feature, inadvertent opening of more than one RVV is considered. The failure of an ECCS valve to a partially open position was evaluated and determined not to be a credible initiating event.

15.6.6.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS being designed with appropriate margin so SAFDLs are not exceeded during normal operations, including AOOs.
- GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to ensure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

- GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so the RCPB is not breached during normal operations, including AOOs.

- GDC 20, which requires that the protective system automatically initiate the operation of the reactivity control system to ensure that SAFDLs are not exceeded as a result of AOOs.

- GDC 26, as it relates to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

- GDC 29, which requires that protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

- GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal water reactor must be limited to negligible amounts.

The staff notes that the applicant provided the PDC for GDC 35. PDC 35 proposed by NuScale is functionally identical to GDC 35.

FSAR Table 15.6-16, "Inadvertent Operation of the Emergency Core Cooling System - Results - Limiting Minimum Critical Heat Flux Ratio Case" and FSAR Table 15.0-2 list the acceptance criteria for demonstrating conformance with these requirements.

15.6.6.4 Technical Evaluation

The following discusses the staff's technical evaluation of the applicant's analysis of inadvertent operation of ECCS.

Subsequent to completing the majority of the analyses reported in this SE, NuScale adjusted its DHRS NRELAP5 modeling of the condenser headers in the NRELAP5 non-LOCA input. The impact of the modelling change(s) to DHRS for non-LOCA transients is about a **{{ }}** reduction in DHRS system heat transfer. The staff understands that these modifications to DHRS modelling were not also made to NRELAP5 models built according to the LOCA TR EM, but considers this acceptable given that DHRS typically operates for a very limited period of time in LOCA and IORV transients before ECCS actuates and dominates the NPM-20 depressurization and cooldown. As LOCA break size gets smaller, DHRS runs for longer times before ECCS actuates, however, the RPV CLL relative to the top of active fuel LOCA figure of merit is insensitive to the DHRS operating time prior to ECCS actuation because design basis breaks retain the leaked coolant in the containment and the minimum CLL relative to the top of active fuel occurs after—and is caused by—ECCS actuation. Furthermore, the SER for the LOCA TR includes L&C #13, which applies a **{{ }}** NRELAP5 fouling factor penalty to both sides of the DHRS tubes for peak containment pressure and temperature calculations.

The staff notes that for LOCA and IORV transients, while DHRS heat transfer is always credited, it is most needed to support containment response due to ECCS actuation arising due to very small breaks that leak into containment. Staff considers that the uncertainty associated with DHRS heat transfer performance as stipulated by LOCA SE L&C #13 encompasses the known adjustment to DHRS condenser headers, such as those described in this SER section, below.

15.6.6.4.1 Causes

The staff reviewed FSAR Section 15.6.6 to assess the applicant's identification of causes leading to this event. The staff notes that the applicant analyzed the inadvertent opening of one or two RVVs and the inadvertent opening of one RRV. The staff understands only the opening of one RRV was analyzed because of the IAB feature of the RRV valves. Per FSAR Section 6.3.2.2, "Equipment and Component Descriptions," the IAB feature blocks the RRV opening until the RCS pressure difference between RPV and containment reaches a specific release pressure difference of approximately 450 psid. The IAB is not being considered as subject to single failure (see Section 15.0.0.5 of this SER). Unlike the RRVs, the RVVs do not have the IAB feature and a loss of DC power (EDAS) or an inadvertent ECCS actuation signal will cause both DC-operated trip solenoid valves to open, resulting in the opening of the RVV main valves. Therefore, inadvertent opening of one or two RVVs needs to be considered. The opening of one RSV, with and without EDAS DC power is also evaluated. The RVVs and RRVs include flow restricting venturis between the RPV and the inlet to each valve. The purpose of the venturis is to slow the depressurization rate by limiting blowdown flow during inadvertent valve opening events

In FSAR Section 15.6.6.1, "Identification of Causes and Accident Description," the applicant stated that it does not expect the spurious opening of a single ECCS valve to occur during the lifetime of a module; however, the applicant categorized this event conservatively as an AOO.

15.6.6.4.2 Evaluation Model

FSAR Section 15.6.6.3.1 states that the inadvertent operation of an ECCS valve is evaluated in accordance with the LOCA EM reviewed by the staff and described in Section 15.0.2 of this SER. The IORV events are classified as AOOs. The acceptance criteria and event progression are slightly different from those of LOCAs. For these AOOs, the staff highlights the CHF phenomena modeling since MCHFR is the most important figure of merit for the IORV event. The staff notes the applicant included **{{**

}}, the applicant also developed a new CHF correlation, NSPN-1, to improve the CHF performance in LOCA and IORV conditions; previously, in earlier versions of its LOCA EM, NuScale **{{** **}}**. The staff discusses these CHF correlations in Section 4.11 of the SE on Revision 4 of the LOCA EM TR.

Section 5 of the LOCA EM TR covers the methodology for the EM for this event. The staff's evaluation of the methodology relative to these requirements and the calculational framework established in RG 1.203 is documented in the SER for the LOCA EM TR referenced in Section 15.0.2 of this SER.

15.6.6.4.3 Model Assumptions, Input, and Boundary Conditions

FSAR Section 15.6.6.3.2 states that input parameters and initial conditions were selected which produce conservative results and minimize MCHFR. SER Table 15.6.6-1 provides these input parameters. The staff's evaluation of the input parameters is also provided in Table 15.6.6-1 of this SER and in the subsequent paragraphs. The applicant selected several input parameters based on the results of sensitivity analyses. The staff reviewed the detailed model results for the bounding case submitted (ML23011A012) and then used the NRELAP5 input files to conduct its own sensitivity studies. During this review, the staff observed that the outcome of sensitivity analyses associated with the inadvertent operation of the ECCS event was consistent with the statements made in FSAR Section 15.6.6.3.2. Additional parameter selection is based on the methodology presented in the LOCA EM TR described in Section 15.0.2 of this SER.

Table 15.6.6-1: Initial Conditions and Input Parameters of Note for the Inadvertent Operation of ECCS Event

Model Parameter	Applicant's Assumption	Purpose
Initial power level	Biased high to 102 percent of nominal power	Maximize core power to minimize MCHFR
Initial RCS average temperature	Biased high to 545 degrees F	Maximize initial RCS energy
RCS flow	Biased low	Minimize MCHFR in the core
PZR pressure	Biased low	**{{ }}**
PZR level	nominal (60 percent)	(nominal value)
Reactivity feedback coefficients	Minimums	Restrain the rate of core thermal power decrease
Kinetics parameters	Beginning of cycle + additional biasing	Amplify the production of delayed neutrons, biasing the core thermal power high
Scram characteristics	Maximum time delay, bounding scram worth with most reactive rod stuck, bounding control rod drop rate	Minimize negative reactivity insertion rate
Axial power distribution	Bounding, bottom peaked shape	Maximize the highest axial peaking factor, with the peak at the worst axial

Model Parameter	Applicant's Assumption	Purpose
		location possible for MCHFR
Radial power distribution	{{ }}	{{ }}
Fuel Gap Conductance	Minimum value, {{ }}	Minimum value maximizes initial stored energy in fuel
Limiting ECCS valve	1 RRV with loss of AC and EDAS at event initiation (which also opens both RVVs)	Found to be the worst case after sensitivity studies
Hot assembly inlet flow	{{ }}	{{ }}

Additional considerations include loss of electrical power and single failure. FSAR Section 15.6.6.1 states there is little difference in MCHFR between the limiting case of opening one RRV with loss of all electrical power and opening two RVVs without loss of all electrical power (FSAR 15.6.6.3.2 specifies the loss of power scenarios). The staff agrees the difference in the scenarios is small.

As discussed in FSAR Sections 15.6.5.3.2 and 15.6.6.3.2, the applicant incorporated an ECCS actuation signal into the design coming from the localized low and localized low-low RPV riser two-phase mixture void fraction. The ECCS actuation on low and low-low RPV riser levels are modeled by specific void fraction values of corresponding NRELAP5 riser nodes. Per the sensitivity results discussed in FSAR Section 15.6.5.3.3, the applicant evaluated the impact of the actuation signals and concluded they had a negligible effect on IORV figures of merit. The staff agrees with the applicant's ECCS actuation modeling and timing uncertainty assessment that IORV figures of merit are not challenged in the context of the specified low riser level ECCS actuation signal setpoint range (540" to 552" riser level in Table 15.0-7). This sensitivity study addressed the staff's concern regarding the instrumentation error and uncertainty and fulfilled the L&C #11 of LOCA EM TR. NuScale provided its modeling approach in FSAR Section 15.6.5.3.2, and Section 15.6.5.1.4.2 of this SER provides the staff evaluation of that modeling approach.

The staff reviewed the applicant's single-failure assumptions for this event. The IAB valve is a first-of-a-kind, safety-significant, active component in the NuScale ECCS. Section 6.3 of this SER describes the ECCS system and the IAB, and Section 3.9.6 of this SER contains the component descriptions. During its review, the staff noted that the applicant did not apply the SFC to the IAB of RRV valves; specifically, the valves' function to close. Section 15.0.0.5 of this SER contains a discussion of the Commission decision in SRM-SECY-19-0036. FSAR Section 15.6.6.3.2 states that the single-failure evaluation considered one RVV failing to open, one RRV failing to open, and failure of one ECCS division causing one RVV and one RRV failing to open. The staff agrees with the applicant's assertion that the assumed single failures have no adverse impact on the limiting MCHFR case because these single failures reduce flow rate out of the RPV which increases MCHFR.

15.6.6.4.4 Evaluation of Analysis Results

The staff reviewed the results presented in FSAR Section 15.6.6 to determine if they meet the DSRS acceptance criteria. The staff reviewed the transient behavior of several parameters by evaluating plots of the parameters as a function of time. The staff considered the reactor power, RPV and containment pressure, flow rates (including ECCS valve flow rates and RCS flow rates), CLL above top of active fuel, RCS temperature, and fuel and clad temperatures. FSAR Figure 15.6-46 shows that the CHFR (minimum transient value 1.41) remains above the safety limit of 1.2 during the inadvertent operation of ECCS event (one RRV open, with loss of normal AC and EDAS at event initiation causing two RVVs to open).

Furthermore, FSAR Table 15.6-16 shows that the RCS pressure is maintained below 110 percent of design pressure. Further, the staff performed a confirmatory analysis of a similar event (spurious opening of two RVVs) using TRACE and NRELAP5 and found that both TRACE and NRELAP predict similar phenomena and major plant parameter trends for the inadvertent ECCS valve opening event; spurious opening of two RVVs was an interim limiting IORV event during the review and the limiting MCHFR for such an event is almost equal to the spurious opening of an RRV with loss of AC and DC, which became the final MCHFR-limited event for IORV AOO. The staff audited the applicant's inadvertent operation of ECCS calculations that investigated the most limiting initial conditions, single failures, and loss of power assumptions to confirm that they led to the most limiting results.

Based on the event description, and results in FSAR Section 15.6.6, the staff finds the MCHFR fuel safety limit remains above the acceptance criterion and that RCS pressure is maintained below 110 percent of design pressure because the inadvertent operation of the ECCS event is a depressurization event for the primary side of the NPM. NuScale performed a sensitivity study on the impact of ECCS actuation signal due to riser level signals uncertainties on NPM LOCA response and due to the similarity between NPM-20 LOCA and NPM-20 IORV, this sensitivity study for LOCA applies to IORV cases.

FSAR section 15.6.6.5 states that the event escalation acceptance criteria are satisfied because the NPM reaches a safe stable state and continues to be cooled with natural circulation through the ECCS valves. Based on the staff's evaluation of the potential for event escalation in Section 15.0.0.2 of this SER, the staff finds these criteria satisfied.

15.6.6.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.6.6.

15.6.6.6 Conclusion

The staff reviewed the inadvertent operation of the ECCS event, including the sequence of events, values of parameters and assumptions used in the analytical models, and predicted consequences of the transient. The staff concludes that the applicant's analysis of this event is acceptable and meets the requirements of GDC 10, 13, 15, 20, 26, 29 and 35 with respect to this event because it satisfies the acceptance criteria in the DSRS.

15.7 Radioactive Release from a Subsystem or Component

This section of the FSAR addresses events that could result in a radioactive release from a component or system other than the RCS. The sources of such releases are waste processing systems and fuel handling systems. The NuScale US460 standard design is similar to current generation PWRs with respect to waste processing systems and fuel handling systems. The design is unique in that the entire NPM is moved for refueling. FSAR Section 15.7 points to other parts of the FSAR that contain the evaluation of these events.

15.7.1 Gaseous Waste Management System Leak or Failure

SER Section 11.3contains the staff's evaluation of a gaseous waste management system leak or failure.

15.7.2 Liquid Waste Management System Leak or Failure

SER Section 11.2contains the staff's evaluation of a liquid waste management system leak or failure.

15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

SER Section 11.2 contains the staff's evaluation of a postulated radioactive release resulting from liquid-containing tank failures.

15.7.4 Fuel Handling Accidents

A fuel handling accident may occur during movement of fuel. The failure of one entire assembly is assumed to occur when an assembly is dropped in the pool above the spent fuel racks or weir wall, an assembly is dropped in the reactor core during refueling, or an assembly impacts a spent fuel cask during loading. Activity is assumed to be instantaneously released into the pool water from all irradiated fuel rods in the assembly. Specific isotopes are assumed to remain in the pool water or enter the RB atmosphere instantaneously, either fully or partially, and are assumed to be directly released into the environment over a two-hour period.

Fuel handling accidents are classified as PAs as indicated in FSAR Table 15.0-1. SER Section 15.0.3 provides the staff's evaluation of the radiological consequence analysis for the fuel handling accident.

15.7.5 Spent Fuel Cask and NuScale Power Module Drop Accidents

FSAR Section 15.7.5, "Spent Fuel Cask and NuScale Power Module Drop Accidents," states that the use of the reactor building crane (RBC) to move the spent fuel cask and NPMs in the

RB refueling area precludes the need to perform load drop evaluation. As a result, the applicant has not performed a DBA analysis to assess the radiological consequences of a spent fuel cask drop accident or NPM drop accident. The applicant stated that RBC system design conforms to ASME standards so that a credible failure of a single component will not result in the loss of capability to stop and hold a critical load. SRP Section 15.7.5, "Spent Fuel Cask Drop Accidents," states that accident analysis for a spent fuel cask drop is not required if the spent fuel cask handling design and procedures prevent the cask from falling or tipping onto spent fuel. The staff agrees that, based on crane design single failure provisions, the dropping of a spent fuel cask is not considered to be a credible design basis event. Section 9.1.5 of this SER presents the staff's evaluation of the RBC system design and capabilities. Chapter 19 of this SER provides additional information on module drop events.

15.8 Anticipated Transients without Scram

15.8.1 Introduction

An anticipated transient without scram (ATWS) is characterized as a failure of the MPS to initiate a reactor trip in response to an AOO. The probability of an AOO, in coincidence with a failure to scram, is much lower than the probability of any other event analyzed in this chapter. Therefore, an ATWS event is classified as a BDBE. The regulatory requirements associated with the mitigation of the consequences of ATWS appear 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants" ((the "ATWS rule").

The underlying purpose of the specific design features required by the ATWS rule is to lesson the risk associated with ATWS events by reducing the likelihood of failure of the reactor protection system to shut down the reactor (scram) following anticipated transients and to mitigate the consequences of ATWS events. For evolutionary plants where the ATWS rule does not explicitly require a diverse scram system, SRP Section 15.8, "Anticipated Transients without Scram," notes that an applicant may provide either of two options to reduce the risks associated with ATWS. The first option is to provide a diverse scram system, which would reduce the probability of a failure to scram. The Statement of Considerations for the ATWS rule in SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," issued July 1983, suggests that the safety goal of the specific design features in 10 CFR 50.62 is to reduce the expected core damage frequency (CDF) associated with ATWS to about 1×10^{-5} per year. Therefore, a diverse scram system or other design feature should reduce the ATWS CDF to a level close to 1×10^{-5} per year to reduce the risks of ATWS to an acceptable level to satisfy this option. The second option is to demonstrate that SRP Section 15.8 ATWS safety criteria are met when evaluating the consequences of an ATWS occurrence.

15.8.2 Summary of Application

In FSAR Section 15.8, "Anticipated Transients Without Scram," the applicant stated that, in the NuScale design, the ATWS contribution to CDF is significantly below the safety goal of 1x10-5 per year, as demonstrated in the PRA described in FSAR Section 19.1. This low contribution is based on the reliability of the reactor trip function of the MPS. The MPS, which is described in FSAR Sections 7.1 and 7.2, includes a robust reactor protection system with internal diversity, which avoids common cause failures and reduces the probability of a failure to scram. The MPS uses the highly integrated protection system (HIPS) platform. TR-1015-18653-P-A, Revision 2, “Design of Highly Integrated Protection System Platform Topical Report,” issued on September 13, 2017 (ML17256A894) describes integration of fundamental instrumentation and controls design principles into the HIPS design. The HIPS platform encompasses the principles of independence, redundancy, predictability and repeatability, and diversity and defense in depth. The applicant further stated that the redundancy and diversity of the MPS design ensures that an ATWS occurrence is a very low probability event for the NuScale Power Plant, which meets the intent of the first criterion of SRP Section 15.8 for evolutionary plants. The applicant also stated that the NuScale design supports an exemption from the portion of 10 CFR 50.62(c)(1) requiring diverse TT capabilities because the NuScale design does not rely on a TT to reduce the risk associated with ATWS events.

Additionally, the applicant stated that the NuScale design does not include an auxiliary FW system, and therefore, the portion of 10 CFR 50.62(c)(1) that requires diverse capability to initiate an auxiliary FW system is not applicable to the NuScale design. FSAR Section 19.2 describes the analysis of this beyond design basis ATWS event.

SDA Part 7, Section 3 documents the applicant’s request for exemption from the TT requirement of 10 CFR 50.62(c)(1), which states the following:

> Each pressurized-water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

The applicant provided an additional description of ATWS and the MPS mitigation systems in FSAR Chapter 19 and FSAR Chapter 7 respectively.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this section of the applicant’s SDAA.

Topical Reports: TR-1015-18653-P-A, Revision 2, is associated with this section of the SDAA.

15.8.3 Regulatory Basis

SRP Section 15.8 acceptance criteria for ATWS are based on meeting the relevant requirements of the following Commission regulations:

- 10 CFR 50.62 (the ATWS rule), as it relates to the acceptable reduction of risk from ATWS events via (1) inclusion of prescribed design features and (2) demonstration of their adequacy.

- 10 CFR 50.46, as it relates to maximum allowable PCTs, maximum cladding oxidation, and coolable geometry.

- GDC 12, as it relates to whether the design of the reactor core and associated coolant, control, and protection systems ensures that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed.

- GDC 14, as it relates to ensuring an extremely low probability of failure of the RCPB.

- GDC 16, as it relates to ensuring that containment design conditions important to safety are not exceeded because of PAs.

- GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reaction must be limited to negligible amounts.

- GDC 38, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment.

- GDC 50, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

The staff notes that the applicant provided the PDC for GDC 35, which is functionally identical to GDC 35.

The guidance in SRP Section 15.8 details the acceptance criteria for demonstrating conformance with these requirements, as well as review interfaces with other SRP/DSRS sections.

15.8.4 Technical Evaluation

The staff reviewed FSAR Section 15.8, to ensure that the regulatory and technical acceptance criteria described in the SRP for this section are satisfied. For evolutionary plants, the Statement of Considerations for the rule in SECY-83-293 allows the option either to provide a diverse scram system satisfying the design and quality assurance requirements specified in SRP Section 7.2, "Reactor Trip System," or to demonstrate that the consequences of an ATWS event are within acceptable values. The applicant stated that the redundancy and diversity of the MPS design ensures that an ATWS occurrence is a very low probability event which meets the underlying purpose of the first criterion of SRP 15.8 for evolutionary plants. The MPS includes a robust reactor protection system with internal diversity, which avoids common cause failures and reduces the probability of a failure to scram.

Chapter 7 of this SER documents the staff's review of the applicant's request for exemption from the TT requirements of 10 CFR 50.62(c)(1) The key component of the MPS is the HIPS, which provides the reactor trip, diversity, and defense-in-depth functions necessary to meet the intent of the ATWS rule. The staff approval of the HIPS design is documented in the SE for the HIPS TR.

During preapplication interactions with NuScale for the previous NPM-160 design, the staff documented its view that the 10 CFR 50.34(f)(2)(xii) and 10 CFR 50.62(c)(1) requirements relating to conventional PWR auxiliary FW system automatic initiation may not apply to the NuScale SMR design and that an exemption request may not be needed. NuScale's view was documented as Gap 3, "Auxiliary Feedwater System Actuation and Flow Indication," of NP-RP-0612-023, Revision 1, the "Gap Analysis Summary Report," issued July 2014 (ML14212A832). The staff documented the basis for its statements in a February 2016 letter to the applicant (ML15272A208). The staff's rationale for its statements was that the NuScale DHRS would perform the decay heat removal functions that would normally be expected of auxiliary feedwater systems found at typical light-water PWRs. Since the NPM-20 DHRS design performs the same functions as the prior NPM-160 design, the staff's previous determination remains applicable. The staff reviewed the information in FSAR Section 5.4.3, "Decay Heat Removal System", which states that the design supports an exemption from 10 CFR 50.62(c)(1). Section 5.4.3 of this SER documents that review.

Analyses of ATWS event sequences performed by the applicant and documented in FSAR Chapter 19, show if the RSVs fail to open and DHRS is unavailable, RPV pressure continues increasing until ECCS is actuated on the beyond design basis high-high RCS pressure signal included in FSAR Table 7.1-4, "Engineered Safety Feature Actuation System Functions". Opening of at least one RVV and one RRV prevents core damage. The staff independently confirmed this conclusion. Because the decay heat removal function can be performed passively, the staff finds that a conventional PWR auxiliary FW system automatic initiation does not need to be applied to the NuScale small modular reactor design to prevent core damage. Therefore, the 10 CFR 50.62(c)(1) requirement associated with automatic initiation of auxiliary feedwater does not apply to the NuScale design.

The staff reviewed the information submitted in FSAR Section 15.8, to ensure compliance with the requirement in 10 CFR 50.62(c)(6), that information sufficient to demonstrate the adequacy of items in 10 CFR 50.62(c)(1)–(5), was submitted in accordance with 10 CFR 50.4, "Written communications." The staff finds that the information submitted in support of the request for exemption from the TT requirements of 10 CFR 50.62(c)(1) is sufficient to demonstrate the adequacy of items in 10 CFR 50.62(c)(1) based on the staff's SER for FSAR Chapter 7. The staff confirmed that the items in 10 CFR 50.62(c)(2)–(5) do not apply to the NuScale design.

Chapter 19 of this SER discussed the staff evaluation of the ATWS BDBE and its associated CDF.

15.8.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.8.

15.8.6 Conclusion

As described in Section 7.1.4.4.5.1 of this SER, the staff concludes that the NuScale design meets the exemption criteria in 10 CFR 50.12 for the applicable portion of the ATWS rule. The other portions of the ATWS rule are not applicable, for the reasons described above. As discussed in further detail in SER Section 7.1.5.4.5.1, the NuScale design provides acceptable reduction of risk from ATWS events by (1) inclusion of prescribed design features that provide redundancy and diversity through the MPS design and (2) demonstration of their adequacy by ensuring that an ATWS occurrence is a very low probability event. The staff finds that special circumstances in accordance with 10 CFR 50.12(a)(2)(ii) are present because the NPM design meets the underlying purpose of the ATWS rule as specified in the SRM and the discussion contained in the Statement of Considerations for the final rule.

15.9 Stability

15.9.1 Introduction

Thermal-hydraulic and coupled thermal-hydraulic/neutronic instabilities can occur in reactors using natural circulation to drive primary flow. Certain conditions, such as low flow and high power, could result in two-phase flow (boiling, subcooled boiling, or flashing), which causes flow and pressure oscillations. Such oscillations are typically denoted as density wave oscillations. If these oscillations are growing, in primary pressure and flow, they are called density wave instabilities. For natural circulation systems, oscillations or instabilities could occur because of buoyancy induced density difference, including those from transition to two-phase flow, and are also coupled with neutron kinetics (core power) and secondary side changes. One approach is to determine the range of parameters and conditions in which the system remains stable and exclude operation of the reactor outside this range. Such an exclusion region limits operation to conditions under which instabilities will not develop. However, an unmitigated instability could result in a new steady state condition or initiation of the applicant's MPS.

15.9.2 Summary of Application

With the natural circulation design feature of the NPM, core flow is driven by density variations in place of pumps that produce forced circulation. Thus, variations in power level may result in changes in flow. The response of the NPM to perturbations and the behavior to flow instability are evaluated. The evaluation considers reactivity coefficients that span the range associated with beginning to end of cycle. The applicant demonstrates that the NPM is protected from unstable flow oscillations provided that operation is limited by an exclusion zone that precludes boiling in the riser area above the core.

ITAAC: There are no ITAAC for this area of review.

Technical Specifications: The GTS listed in Section 15.0.0 of this SER are applicable to this area of review.

Technical Reports: There are no technical reports associated with this section of the applicant's FSAR.

15.9.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 10, as it relates to the RCS being designed with appropriate margin to assure that SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 12, which requires that power oscillations which can result in conditions exceeding SAFDLs, be either not possible or reliably and readily detected and suppressed.

- GDC 13, as it relates to instrumentation provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions, and to maintain these variables and systems within prescribed operating ranges.

- GDC 20, which requires the reactor protection system to initiate automatic action to assure that SAFDLs are not exceeded as a result of AOOs. Conditions that result in unstable power oscillations are AOOs.

- GDC 29, which requires that protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

DSRS Section 15.9.A lists the acceptance criteria for demonstrating conformance to these requirements, as well as review interfaces with other SRP/DSRS sections.

15.9.4 Technical Evaluation

The staff reviewed FSAR Section 15.9 to assess the applicant's approach to preventing or mitigating instabilities and power oscillations that could pose unfavorable flow or thermal conditions and result in SAFDLs being exceeded.

15.9.4.1 Exclusion Region Protection

In FSAR Section 15.9.1, "Consideration of Thermal-Hydraulic Stability," the applicant stated that use of the exclusion region stability protection solution is an acceptable approach for preventing the occurrence of instabilities in the NPM. The staff reviewed the exclusion region that the applicant applies to their long term stability-solution. The staff noted the analytical boundaries that separate the acceptable NPM operating region from the exclusion region are plotted as RCS hot temperature versus PZR pressure in FSAR Figure 4.4-2, "Analytical Design Operating Limits".

In FSAR Section 4.4.3.2, "Analytical Design Operating Restrictions," the applicant states that the low pressure analytical limit is 1,200 psia for RCS hot-leg temperatures less than 260 °C (500 °F) and 1,850 psia for RCS hotleg temperatures greater than 260 °C (500 °F). The staff notes that the subcooling margins are slightly less than 2.8 °C (5 °F) (approximately 2.7 °C (4.8 °F and 4.9 °F, respectively)) for the analytical limits at and near the exclusion boundary point represented by (326 °C (620 °F), 1,850 psia) in FSAR Figure 4.4-2. There is additional margin to the exclusion boundary point represented by (260 °C (500 °F), 1,200 psia) in FSAR Figure 4.4-2. While the exclusion region is not strictly defined by a subcooling margin of 2.8 °C (5 °F) (or greater), the staff finds it acceptable because it provides an exclusion region consistent with the referenced stability methodology.

15.9.4.2 Evaluation Model

The applicant used the stability methodology discussed in Section 15.0.2 of this SER, including use of the PIM thermal hydraulic computer code to simulate the dynamics of the flow in the NPM coolant loop with attention to optimal resolution of its stability. The applicant's evaluation methodology for the stability analysis is described TR-0516-49417-P-A, Revision 1. The applicant determined that the referenced TR is applicable to the US460 design using the NPM-20. The staff reviewed the applicant's justification (ML24346A303) that confirmed this applicability and agrees that the design and operational changes did not introduce new physics not captured in the existing models, as developed in the above referenced TR.

15.9.4.3 Input Parameters and Initial Conditions

FSAR Table 15.9-1, "Initial Conditions (5 Percent of Rated Power Cases)," presents the input parameters and initial conditions for the limiting stability case, and FSAR Sections 15.9.3.1.3, 15.9.3.2.3, and 15.9.3.5.3, discuss the input parameters and initial conditions for the transient scenarios analyzed. The applicant stated that both BOC and EOC conditions were analyzed; however, BOC results are presented for the perturbed steady state analyses, while EOC results are typically presented for the transient analyses in FSAR Section 15.9, since these results were more limiting. NuScale further stated that input conditions are nominal input parameter values for three core power levels of 12.5 MWt (5 percent RTP), 250 megawatts thermal (MWt) (100 percent RTP) and 50 MWt (20 percent RTP). The staff audited (ML24211A089) selected stability calculations and confirmed that the applicant applied reasonable values to some key parameters and conditions. The staff also reviewed the key conditions described in Sections 15.9.4.3.1 through 15.9.4.3.4 of this SER and the allowable ranges for these parameters based on similar considerations for other events analyzed in FSAR Chapter 15.

15.9.4.3.1 Core Inlet and Core Average Temperature

The core inlet and core average temperatures can be affected by changes in secondary side operation or other operational considerations such as SG tube fouling. The stability characteristics (and analysis results) are not sensitive to the primary side temperature so long as the primary side temperature is maintained within the exclusion region boundary of 2.8 °C (5 °F) subcooling. This is because of a cancellation of errors in the steady state operation whereby a higher temperature in the core corresponds to higher temperature throughout the RCS, leading to essentially the same stability characteristics. In the transient depressurization analysis, the initial temperature has a negligible effect on the analysis because the RPV is depressurized until the pressure reaches the MPS trip setpoint. Therefore, the analysis does not need to consider variation in the RCS temperature in the same manner as other events in Chapter 15.

15.9.4.3.2 Pressurizer Pressure

PZR pressure may vary by 35 psia. However, it is not necessary to consider such variation in the stability analysis. For decay ratio (DR) calculations, the PZR pressure (much like the RCS temperature) affects both the hot and cold legs equally and can therefore be expected to have minimal effect on the stability characteristics of the power module. The initial pressure, much like RCS temperature, affects the subcooling margin. However, the depressurization event is analyzed in such a manner that the RPV is depressurized until an MPS trip occurs which subsequently protects the subcooling margin. Therefore, the initial subcooling margin does not affect the analysis.

15.9.4.3.3 Feedwater Temperature and Feedwater Flow

The feedwater temperature and feedwater flow can affect the initial RCS temperature; therefore, the impact of these parameters on the steady-state DR is the same as for the RCS temperature. During the transient analysis, the FWS parameters are not assumed to change, and thus the effect of these parameters on the transient analysis is the same as the impact given by the associated change in the RCS temperature. In other words, for the same reason that variations in RCS temperature are insignificant, the variations in feedwater temperature and flow are insignificant to the stability analysis.

15.9.4.3.4 Reactor Coolant System Flow and Reactor Power Level

The RCS flow and reactor power level are interrelated. The uncertainty in RCS flow rate at a given power level is considered part of the stability analysis methodology and factors into the DR acceptance criterion. Therefore, the impacts of the initial conditions here are related because a change in the power level causes a change in the core flow rate. To this end, the stability analysis considers a variation of power and flow conditions ranging from 5 percent to 100 percent power conditions. Since the reactor tends to become more stable at higher power levels, the low power level of 5 percent analyzed is sufficient to address any staff concerns regarding the range of initial power levels. The applicant stated that the least stable flow stability condition occurs at the 12.5 MWt (5 percent) power level with a BOC reactivity condition. The staff notes that performing the analysis down to the 5 percent power level is conservative because the stability analysis methodology only requires consideration of the DR for power levels where thermal limits may be challenged, which occurs for core powers in excess of 5 percent.

Additionally, the staff notes that reactor physics parameters are selected based on the stability EM described in TR-0516-49417-P-A, Revision 1.

15.9.4.4 Evaluation of Analyses

The staff reviewed the analyses presented in FSAR Section 15.9 to determine if they meet the DSRS acceptance criteria. The staff notes that the applicant performed stability analyses over a spectrum of events that include perturbation of steady state and transient operations where the initiating events are variations of selected AOOs. The staff further notes that the applicant considered transient events from six AOO classification types and considered two other events: startup and cooldown.

15.9.4.4.1 Perturbed Steady-State Operation

The staff reviewed NPM stability under conditions of steady-state operation in FSAR Section 15.9.2. The applicant considered a variety of power levels and flow conditions in the presence of a small perturbation in operating conditions to demonstrate that the DR remains below the acceptance criterion (0.8) for all conditions. The staff notes that the most limiting case the applicant analyzed is for low power (5 percent RTP) with a BOC reactivity condition where the DR is 0.58. The results of the calculations show that for steady state operation conditions, there is no challenge to fuel limits and the NPM is stable.

15.9.4.4.2 Increase in Heat Removal by the Secondary System

The staff reviewed NPM stability AOOs that increase heat removal by the secondary system in FSAR Section 15.9.3.1. The applicant analyzed an increase in feedwater flow by 30 percent that results in an increase in heat removal that would likely result in an automatic trip of the reactor. The staff agrees that the analysis is conservative since it ignores (does not simulate) the trip to bound less severe feedwater flow increase events. The staff confirms that the applicant determined the limiting conditions with respect to this class of AOOs for stability analysis. The staff notes that a feedwater flow increase yielding a power increase sufficient to produce a reactor trip bounds less severe flow increases that would not necessarily result in a reactor trip. The applicant performed calculations at rated power initial conditions and at 20 percent of rated power (50 MWt), which show that the NPM remains stable during a postulated increase in heat removal by the secondary system.

15.9.4.4.3 Decrease in Heat Removal by the Secondary System

The staff reviewed NPM stability AOOs that decrease heat removal by the secondary system in FSAR Section 15.9.3.2. For stability analyses, conditions that produce the largest reduction in secondary side heat removal are not limiting because they would likely lead to a prompt, automatic reactor trip due to an increase in PZR pressure. The most adverse event from a stability perspective would be an AOO that maximizes the potential for the riser to void while avoiding the high PZR pressure trip.

To establish a conservative, bounding analysis method, TR-0516-49417-P-A, Revision 1, assumes a 50 percent feedwater flow reduction. This reduction is large enough to cause a reactor trip due to high PZR pressure, which is not credited in the TR-0516-49417-P-A, Revision 1, analysis. Therefore, TR-0516-49417-P-A, Revision 1, defined a methodology that, if followed, conservatively bounds feedwater flow reduction transients.

However, the analysis in FSAR Section 15.9.3.2, departs from the methodology in TR-0516-49417-P-A, Revision 1, since the feedwater flow is reduced by 30 percent compared to the 50 percent reduction prescribed by TR-0516-49417-P-A, Revision 1. According to the applicant's analyses, and its description in FSAR Section 15.9.3.2.2, the 30 percent feedwater flow reduction was chosen to determine the acceptability of a partial loss of feedwater and conservatively bounds any smaller changes to feedwater flow because a reactor trip is not simulated. The applicant presented the transient responses for both 100 and 20 percent rated power at both BOC and EOC conditions in FSAR Figures 15.9-5 and 15.9-6. The results show that the NPM is stable during a postulated decrease in heat removal by the secondary system.

To demonstrate long term stability, the applicant presented results in FSAR Figures 15.9-8 and 15.9-11 from analyses where the high hot-leg temperature trip is ignored. These results show core flow and power oscillations that rapidly decay in time over intervals of 2,000 to 3,000 seconds, which are greater than 10 times the period of the reactor and sufficiently long to characterize long term stability. Since the applicant departed from the methodology and did not use the prescribed 50 percent feedwater flow reduction, the staff requested additional justification and agrees with the analysis provided behind the selection of a 30 percent feedwater flow reduction (ML25065A161). Therefore, the FSAR Section 15.9.4.4.3 analysis demonstrates that the long-term stability solution meets the requirement of GDC 12 since the riser voiding is mitigated following a decrease in secondary heat removal which prevents instabilities that could challenge SAFDLs.

The applicant's analyses for feedwater flow reduction demonstrate that the RCS heats up and that a reactor trip protecting the riser subcooling margin is eventually initiated. Based on these analyses, the staff finds that this event progression is generally applicable and occurs regardless of the magnitude of the feedwater flow reduction (ML25065A161). The staff also finds that the long-term stability solution is effective in preventing the reactor from reaching an unstable condition by initiating a reactor trip before such an instability would occur and therefore meets the requirement of GDC 12.

The analysis presented in FSAR Section 15.9.3.2, is a departure from, and non-conservative with respect to, the stability analysis methodology presented in TR-0516-49417-P-A, Revision 1. Therefore, the staff finds that the FSAR Section 15.9.3.2 safety conclusions shall not be construed as approval of the departure or as tacit acceptance of a change in the stability evaluation methodology presented in TR-0516-49417-P-A, Revision 1. Nevertheless, for the reasons indicated above, the staff notes that for the NuScale NPM, the long-term stability solution is effective in preventing instability during AOOs that cause a decrease in secondary side heat removal.

15.9.4.4.4 Decrease in Reactor Coolant System Flow Rate

The staff reviewed NPM stability AOOs that decrease RCS flow rate in FSAR Section 15.9.3.3. The applicant stated that it does not consider a decrease in the RCS flow rate to be a credible event for stability analyses because there is no source for changing the primary system flow without other influences, and because there are no primary system pumps in the NPM to directly influence primary system flow. However, the staff notes that during a postulated AOO, it is conceivable that a sequence involving inadvertent operation of components related to the CVCS that could lead to reduced or increased primary system flow (such as CVCS pump overspeed or pump trip). The staff notes that since the CVCS is essentially external to the primary flow circuit, such AOOs could impact the RCS without other effects. The staff disagrees with the applicant’s assertion that a CVCS malfunction leading to a reduction in RCS flow rate is not a credible event. However, the staff finds it a credible but nonlimiting event and agrees with the applicant that this class of events is bounded by events resulting in a decrease in secondary-side heat removal. Therefore, a separate analysis is not required for this class of events.

15.9.4.4.5 Increase in Reactor Coolant Inventory

The staff reviewed NPM stability AOOs that increase reactor coolant inventory in FSAR Section 15.9.3.4. The applicant stated that the effects of increasing RCS inventory are not important in the stability assessment because subcooling margin in the riser increases with increasing RCS pressure, and that overall stability behavior is not sensitive to pressure changes for a single-phase system. However, a decrease and subsequent loss in riser subcooling margin could result in unstable behavior. Therefore, the applicant dispositioned pressurization events as unimportant to stability analyses, since these events would increase the subcooling margin. The staff finds the applicant’s disposition acceptable as events that increase the RCS inventory and simultaneously increase pressure do not result in unstable RCS behavior because the subcooling margin in the riser increases with increasing pressure.

15.9.4.4.6 Reactivity and Power Distribution Anomalies

The staff reviewed NPM stability AOOs with respect to reactivity and power distribution anomalies in FSAR Section 15.9.3.5. The applicant stated that boron concentration changes via the CVCS are slow and that these events would likely be bounded by other analyses. The staff agrees that a CVCS malfunction resulting in boration or dilution would likely be a slowly evolving transient and would be bounded, or at least similar to, the events that increase or decrease heat removal from the primary system. In terms of control rod withdrawal, the staff agrees that protective trips are designed to protect thermal margins for control rod withdrawal events. The staff notes that the applicant analyzed a hypothetical reactivity increase of 0.90 dollars for both 100 percent power and 20 percent power. The staff finds this approach to be reasonable given these considerations. Both BOC and EOC conditions were considered, with the EOC case being the more limiting event. FSAR Figures 15.9-7 and 15.9-8 show transient responses for the limiting case of 20 percent power at EOC. The applicant noted, and the staff agrees, that the NPM is maintains a substantial stability margin during a postulated addition of reactivity event.

15.9.4.4.7 Decrease in Reactor Coolant Inventory

The staff reviewed NPM stability AOOs that decrease reactor coolant inventory in FSAR Section 15.9.3.6. The staff notes that a decrease in inventory without a commensurate decrease in pressure would simply result in a reactor trip based on low PZR level, and therefore, those types of events need not be considered. The applicant discussed events that result in reduced pressure and concluded that events that do not reduce pressure sufficiently to result in riser flashing will not result in instability. The staff agrees that a low PZR pressure trip would occur before loss of subcooling margin, protecting the NPM against instabilities.

Additionally, the staff reviewed AOOs that could result in riser voiding. In FSAR Section 15.9.3.6, the applicant simulated a depressurization transient with two rapid depressurizations assumed. The first is from the initial pressure to just above the low PZR pressure trip setpoint over a period of 5 seconds so that no system operations occur in response to the event. The results from this event indicate that the NPM is stable for depressurizations that do not result in a reactor trip. The second depressurization is from the initial pressure to well below the low pressurizer pressure trip setpoint over a period of 10 seconds so that the trip occurs, which is consistent with the opening of a single RVV. These results indicate show that the MPS is effective at shutting down the NPM before instability occurs. FSAR Tables 15.9-11 and 15.9-12, and FSAR Figures 15.9-9 and 15.9-10 present results for both events. The applicant stated, and the staff agrees, that the NPM is stable during a postulated decrease in reactor coolant inventory.

15.9.4.4.8 Demonstration of Module Protection Systems to Preclude Instability

The staff reviewed the capability of the MPS to preclude instability in FSAR Section 15.9.4. The applicant stated that the NPM minimum loop transit time is greater than 60 seconds at rated power, while the time to scram is less than 11 seconds. The applicant concluded, and the staff agrees, that the MPS will enforce the exclusion region and shut down the reactor before violating thermal limits because the scram time is significantly less than the loop transit time. The applicant demonstrated that in the event of the secondary side becoming unstable, the MPS will ensure thermal limits are met.

15.9.4.4.9 Secondary Side Oscillations

The staff reviewed the NPM stability in the context of continuously present, imposed oscillations on the secondary side with and without the MPS response in FSAR Section 15.9.5. The applicant concluded, and the staff agrees, that the spectrum of analyses presented in FSAR Sections 15.1 and 15.4 bound the effects of an oscillation-induced initiating event. The MCHFR for the limiting oscillation case in FSAR Section 15.9.5 is not more limiting than cooldown events analyzed in Section 15.1 nor the rod withdrawal events analyzed in FSAR Section 15.4. Further, the evaluation shows that the MPS effectively mitigates oscillations that have sufficient effect on the primary side conditions prior to the conditions challenging margin to specified acceptable fuel design limits.

15.9.4.5 Barrier Performance

The applicant concluded, and the staff agrees, that the pressure in the reactor coolant and main steam systems is maintained below 110 percent of the design values for this event, and the minimum DNBR remains above the 95/95 limit. Based on these conclusions, the staff finds that there is no challenge to any of the fission product barriers for this AOO.

15.9.4.5.1 Radiological Consequences

Based on the results of the analysis and barrier performance, the applicant concluded, and the staff agrees, that there are no radiological consequences associated with events that could result in thermal-hydraulic instability.

15.9.5 Combined License Information Items

There are no COL information items associated with FSAR Section 15.9.

15.9.6 Conclusion

The staff concludes that the consequences of postulated instabilities and power oscillations meet the relevant requirements set forth in the GDC 10, 12, 13, 20, and 29. As documented above, the applicant's analysis and exclusion region showed that the DSRS acceptance criteria are met.

15.10 Core Damage Event

The applicant evaluated the CDE, with an associated CDST consisting of a set of key parameters derived from a spectrum of surrogate accident scenarios. The applicant's analysis of the CDE as compared to the acceptance criteria of 10 CFR 52.137(a)(2)(iv) provides reasonable assurance that, even in the extremely unlikely event of a CDE, the NuScale Power Plant US460 standard design features and site characteristics provide adequate protection of the public. Section 15.0.3 of this SER provides the staff's evaluation of the radiological consequence analysis for the CDE.