# 20 MITIGATION OF BEYOND-DESIGN-BASIS EVENTS

This chapter of the final safety evaluation report (FSER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 20, "Mitigation of Beyond-Design-Basis Events," of the NuScale Power, LLC (hereinafter referred to as the applicant), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report." The staff's regulatory findings documented in this report are based on Revision 5 of the DCA, dated July 29, 2020 (Agencywide Document Access and Management System (ADAMS), Accession No. ML20225A071). The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the DCA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this safety evaluation 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 DCA and not converted.

During review of the application, the NRC amended its regulations (Title 10 of the *Code of Federal Regulations* (10 CFR) 50.155, "Mitigation of Beyond-Design-Basis Events") to establish regulatory requirements for nuclear power reactor applicants and licensees to mitigate beyond-design-basis events. This rule, in part, makes NRC Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012, and Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012, generically applicable. In addition, the rule relocates requirements for loss of large areas (LOLA) of the plant due to fire or explosion from 10 CFR 50.54, "Conditions of Licenses," to 10 CFR 50.155.

## 20.1 Mitigating Strategies for Beyond-Design-Basis Events

### 20.1.1 Introduction

The NuScale design incorporates several innovative design features that provide enhanced capabilities for mitigating an extended loss of electrical power compared to currently operating nuclear reactor plants. These features include the use of passive safety systems capable of maintaining core cooling, containment, and spent fuel cooling functions and a large reactor pool, which serves as the ultimate heat sink (UHS) for the facility. These features are intended to enable the NuScale design to mitigate beyond-design-basis external events (BDBEEs) using only installed plant equipment for an extended duration (greater than or equal to the first 72 hours following the event) without the need for alternating current (ac) power. Although the regulation governing mitigation of beyond-design-basis events (10 CFR 50.155) does not apply to applicants for design certification, NuScale is voluntarily seeking the NRC's approval of its proposal in the DCA to use installed design features for mitigation of BDBEEs.

The staff applied the following approach to complete the evaluation of issues associated with the functions of NuScale design features related to certain provisions of 10 CFR 50.155, consistent with SECY-19-0066, "Staff Review of NuScale Power's Mitigation Strategy, for Beyond-Design-Basis External Events," dated June 26, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19148A443). Specifically, SECY-19-0066 states the following:

- The staff will verify that the design capacities and capabilities of the permanently installed structures, systems, and components (SSCs) in the NuScale design, as described in the FSAR, are capable of providing adequate core cooling, containment, spent fuel pool (SFP) cooling, and SFP level instrumentation consistent with the requirements of 10 CFR 50.155(b)(1), (c), and (e) for 72 hours following a BDBEE.

- Consistent with the review approach applied for operating reactors and the previous APR1400 DCA review, the staff does not plan to review the NuScale design feature capacity and capability beyond 72 hours following a BDBEE in its review of the DCA. However, if the staff determines that there are no credible transient phenomena (e.g., return to power) that could challenge core cooling, containment, or SFP cooling beyond 72 hours following a BDBEE, then no additional review or approval of these capabilities would be required at the combined license (COL) stage. If credible transient phenomena could challenge core cooling, containment, or SFP cooling, then the COL applicant would be required to provide mitigation strategies to address these phenomena. Under 10 CFR 50.155(b)(1) and 10 CFR 52.80(d), a COL applicant referencing the NuScale design will be required to describe mitigation strategies to maintain or restore core cooling, containment, and SFP cooling for an indefinite period, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies. The level of detail needed in this area would be commensurate with the time available to provide additional capability (i.e., capabilities that are needed a longer time after the event can be described in less detail than those that are needed at an earlier time). For example, the COL applicant will need to identify the source of the site-dependent makeup water and a plan to add that water to the reactor pool.

- In 10 CFR 50.155(e), the NRC requires SFP level instrumentation and requires power to maintain instrumentation function until offsite resource availability is reasonably assured. The staff is not planning to review the SFP level instrumentation capability beyond 72 hours following a BDBEE in its review of the DCA. The COL applicant referencing the NuScale design will be required to address SFP level instrumentation in accordance with 10 CFR 52.80(d). Regulatory Guide (RG) 1.227, “Wide-Range Spent Fuel Pool Level Instrumentation,” issued June 2019 (ADAMS Accession No. ML19058A013), provides acceptable guidance for satisfying the requirements of 10 CFR 50.155(e).

- The staff plans to document, in its review of the DCA, that instrumentation, excluding SFP level instrumentation, is not relied upon for the mitigation of beyond-design-basis events for core cooling and containment functions for the initial 72 hours. In addition, the staff plans to document that instrumentation is available and provides additional assurance that systems have responded as designed.

### 20.1.2 Summary of Application

**DCA Part 2, Tier 1:** None.

**DCA Part 2, Tier 2:** DCA Part 2, Tier 2, Section 20.1, “Mitigating Strategies for Beyond-Design-Basis Events,” discusses mitigating strategies and reliable SFP instrumentation.

The applicant stated that, following a loss of all ac power concurrent with a loss of normal access to the normal heat sink, automatic responses of safety-related equipment establish and maintain the key safety functions of core cooling, containment, and SFP cooling by placing the reactor modules into a safe, stable, shutdown state with passive core and containment cooling.

Following the initial, automatic response of safety-related equipment—which requires no operator action and no electrical power (ac or direct current (dc))—the reactor modules and the SFP rely only on the large inventory of the reactor, refueling, and SFPs, which comprise the UHS, to maintain uninterrupted and long-term heat removal.

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

**Technical Specifications:** There are no generic technical specifications for this area of review.

**Technical Report (TR):** TR-0816-50797, "Mitigation Strategies for Loss of All AC Power Event," Revision 3, issued incorporated by reference in DCA Part 2, Tier 2, Section 20.1, Revision 3.

### 20.1.3 Regulatory Basis

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

- 10 CFR 50.155, which requires measures for the mitigation of BDBEEs, specifically regarding the appropriate provisions within 10 CFR 50.155(b)(1), (c), and (e)

The guidance for the staff's review consists of the following documents:

- RG 1.226, "Flexible Mitigation Strategies for Beyond-Design-Basis Events," issued June 2019 (ADAMS Accession No. ML19058A012)

- RG 1.227

- SECY-19-0066

### 20.1.4 Technical Evaluation

#### *20.1.4.1 Protection of Equipment*

In this section of the staff's report, the staff reviews the applicant's design information provided to satisfy the requirements described in 10 CFR 50.155(c), which contains two parts: 10 CFR 50.155(c)(1) and 10 CFR 50.155(c)(2). The report begins with a discussion on 10 CFR 50.155(c)(2) and then follows with a discussion on 10 CFR 50.155(c)(1).

In 10 CFR 50.155(c)(2), the NRC requires reasonable protection of the equipment relied on for the mitigation strategies and guidelines required by 10 CFR 50.155(b)(1) from the effects of natural phenomena and sets the hazard level for which "reasonable protection" of the equipment must be provided. The hazard level is the level determined for the design basis for the facility for protection of safety-related SSCs from the effects of natural phenomena.

In DCA Part 2, Tier 2, Section 20.1, the applicant described the response of the NuScale design to the assumed damage state of a BDBEE in 10 CFR 50.155(b)(1). The damage state described in 10 CFR 50.155(b)(1) is a loss-of-all-ac-power condition concurrent with a loss of normal access to the normal heat sink (passive reactor designs). In DCA Part 2, Tier 2, Section 20.1.3, the applicant stated that, following a loss of all ac power concurrent with a loss of normal access to the normal heat sink, automatic responses of safety-related equipment establish and maintain the key safety functions of core cooling, containment, and SFP cooling by placing the reactor modules into a safe, stable, shutdown state with passive core and containment cooling.

The staff evaluated whether the installed equipment and UHS (large reactor, refueling, and SFPs) that are credited to be available following BDBEEs are reasonably protected. The staff reviewed the structures that provide protection for installed mitigation equipment and cooling inventories with regard to whether they are designed to be reasonably protected (i.e., robust), consistent with RG 1.226, which endorses the guidelines in Nuclear Energy Institute (NEI) 12-06, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” Revision 4, issued December 2016.

In TR-0816-50797, Sections 3.1.2 and 3.1.5, the applicant stated that installed plant equipment is the only equipment necessary to satisfy the language and intent of 10 CFR 50.155, and that the installed plant equipment credited to meet this regulation is designed to be robust with respect to design-basis external events. Robust design is defined in NEI 12-06, Revision 4, Appendix A, “Glossary of Terms,” as the design of an SSC that either meets the current plant design basis for the applicable external hazard(s) or the current NRC design guidance for the applicable hazard (e.g., RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1, issued March 2007) or has been shown by analysis or test to meet or exceed the current design basis.

The reactor building (RXB) and control building (CRB) structures are credited to provide protection for installed equipment and cooling water inventories. The UHS is credited to maintain uninterrupted and long-term heat removal. The UHS pool walls and pool liner are designed to seismic Category I requirements and are completely contained within the seismic Category I RXB. The staff reviewed and documented its safety findings in Section 3.8 of the SER for the seismic Category I RXB and CRB structures for the applicable design-basis external hazards. Based on the safety review documented in Section 3.8 of this SER, the staff concludes that the seismic Category I RXB and CRB structures are adequately designed against the design-basis external hazards and are robust, since the structures meet the current plant design basis for the applicable external hazard(s). Therefore, consistent with staff guidance (RG 1.226), the staff finds that reasonable protection to the installed plant equipment housed in the RXB and CRB is provided, and the requirements of 10 CFR 50.155(c)(2) are met.

In 10 CFR 50.155(c)(1), the NRC requires that equipment relied on for mitigation strategies have sufficient capacity and capability to perform the functions required by 10 CFR 50.155(b)(1). The NuScale mitigation strategies utilize installed active valves but do not utilize installed pumps or dynamic restraints.

NuScale DCA Part 2, Tier 2, Section 20.1.3, identified the decay heat removal system (DHRS), emergency core cooling system (ECCS) valves, and containment isolation valves (CIVs) as safety-related equipment credited for maintaining core cooling and containment capabilities in response to a BDBEE that results in a loss-of-all-ac-power event. TR-0816-50797 more specifically details the functions of the valves credited in the systems above as follows:

(1) CIVs, which isolate the containment vessel (CNV), deenergize and close.

(2) The DHRS actuation valves deenergize and open, and the main steam isolation valves and feedwater isolation valves deenergize and close to place the DHRS passive condensers in service.

(3) The ECCS valves deenergize and open when plant conditions require or when the 24-hour timer expires.

Note—The applicant used the terminology “deenergize and open/close” in TR-0816-50797, but DCA Part 2, Tier 2, Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints,” and the American Society of Mechanical Engineers Operations and Maintenance Code use “fail-safe testing.” The staff understands these two phrases to refer to the same technical aspect of the valves discussed above.

The applicant stated in an October 31, 2017, letter (ADAMS Accession No. ML17304B482), that DCA Part 2, Tier 2, Section 20.1, credits safety-related valves in the CNTS (CIVs), DHRS, and ECCS for mitigation strategies, and that DCA Part 2, Tier 2, Table 3.2-1, “Classification of Structures, Systems, and Components,” provides information on classification, seismic category, and quality group classification for the system valves. The applicant also stated that none of the valves have performance requirements that exceed their safety-related design and performance criteria. The staff verified that all safety-related valves credited for the mitigation strategies are listed in DCA Part 2, Tier 1, Table 2.8-1, “Module Specific Mechanical and Electrical/I&C Equipment,” and that ITAAC 1, 3, and 6 in DCA Part 2, Tier 1, Table 2.8-2, “Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria, “specify seismic, environmental , and functional qualification for these valves. The staff also verified that these valves are included in the inservice testing program described in DCA Part 2, Tier 2, Section 3.9.6. Based on the above, the staff determined that the applicant identified all of the safety-related installed valves that are credited in the mitigation strategies. Further, the staff finds that the applicant described provisions to assure seismic, environmental, and functional capability of the safety-related installed valves to perform their intended functions as part of the mitigation strategies to ensure core cooling, containment , and SFP cooling capabilities during a loss-of-all-ac-power event at a NuScale nuclear power plant. Accordingly, the DCA satisfies the requirements of 10 CFR 50.155(c)(1) as it pertains to the capability of these valves to perform the functions required for the mitigation strategies.

The applicant also stated that valves that are not safety related are not relied upon to maintain key safety functions following a loss of all ac power. The staff finds the applicant’s response acceptable because the design relies only on safety-related valves to maintain the key safety functions during a loss of all ac power.

*Mechanical Equipment Conclusion*

The staff evaluated NuScale DCA Tier 2, Section 20.1, regarding equipment protection mitigation strategies for BDBEEs. For the reasons described above, the staff finds that the applicant’s approach is consistent with the guidance in RG 1.226. All permanent installed, safety-related equipment that is credited and utilized in the mitigation strategies for BDBEE and the UHS are housed inside the RXB and CRB. The RXB and CRB are safety-related seismic Category I structures and are designed for design-basis external hazards. The staff’s review and safety evaluation of the seismic Category I RXB and CRB structures for the design-basis external hazards are provided in Section 3.8 of the SER, which concludes that the design of these structures for design-basis external hazards is acceptable.

Based on the above findings, the staff concludes that the structures credited to provide protection for installed mitigation equipment and cooling inventories are designed to be robust in accordance with the guidelines in NEI 12-06. On this basis, the installed equipment and the UHS housed in the RXB and CRB credited in the mitigation strategies will be protected from the applicable external hazards, consistent with the provisions of 10 CFR 50.155(c)(2), and will be available following a BDBEE.

The staff also concludes that the provisions in the DCA for the seismic, environmental, and functional capability of active mechanical equipment to perform its intended function as part of the mitigation strategies to ensure core cooling, containment function, and SFP cooling capabilities during a loss of all ac power resulting from a BDBEE are acceptable and thus satisfy the requirements of 10 CFR 50.155(c)(1) as it pertains to the capability of these valves to perform the functions required for the mitigation strategies.

*20.1.4.2 Assessment of Electrical Power*

As stated in SECY-19-0066, the NuScale design incorporates several innovative design features that provide enhanced capabilities for mitigating an extended loss of electrical power compared to currently operating nuclear reactor plants. These features include the use of passive safety systems capable of maintaining core cooling, containment, and spent fuel cooling functions and a large reactor pool, which serves as the UHS for the facility. These features are intended to enable the NuScale design to mitigate a BDBEE using only installed plant equipment for an extended duration (greater than or equal to 72 hours) without the need for ac power.

In DCA Part 2, Tier 2, Chapter 20, the applicant stated the following:

> No operator action is required to establish or maintain the required safety functions for at least 50 days following the onset of a loss of all AC power. Therefore, no instrumentation is necessary to support operator actions. Although not necessary because of the fail-safe and passive design, monitoring instrumentation (safety display and indication system, SDIS) is maintained in the main control room for at least 72 hours to provide additional assurance that systems have responded as designed.

In its review, the staff confirmed that instrumentation other than SFP level instrumentation is not relied upon for the mitigation of beyond-design-basis events for core cooling and containment functions for the initial 72 hours following a BDBEE. However, the staff reviewed the capability and capacity of the NuScale electrical power system design to supply electrical equipment (e.g., instrumentation, lighting, ECCS solenoid valves) during a BDBEE, providing additional assurance that systems have responded as designed.

The staff's review of the electric power systems available for a BDBEE included NuScale DCA Part 2, Tier 2, Chapter 20, Revision 3, Section 20.1, and TR-0816-50797. The focus of the staff's review is on the capability and capacity of the NuScale electric power system design to support monitoring of plant conditions for the first 72 hours following a BDBEE.

*Background*

RG 1.226, Revision 0, states that Section 3 of NEI 12-06, Revision 4, provides performance attributes, general criteria, and baseline assumptions for use in the development and implementation of the strategies and guidelines under 10 CFR 50.155(b)(1). RG 1.226, Revision 0, also states that NEI 12-06, Revision 4, further provides that licensees should use these criteria and assumptions for analyses used to establish a baseline coping capability. Furthermore, RG 1.226, Revision 0, explains that assumptions described in NEI 12-06, Revision 4, Section 3.2.1.3, "Initial Conditions," includes an extended loss of ac power consisting of a loss-of-offsite power affecting all units at a plant site and the specification that all “design-basis installed sources of emergency on-site ac power and station blackout (SBO) alternate ac power sources are assumed to be not available and not imminently recoverable.” The staff’s review of the baseline coping capability assumptions described in TR-0816-50797 corresponding to the assumptions in NEI 12-06, Revision 4, Section 3.2.1.3, is discussed below.

*Electrical Power System Evaluation*

Section 3.2.1.3, of NEI 12-06, Revision 4, lists initial plant conditions and assumptions that should be utilized in developing FLEX mitigating strategies. The applicant provided the NEI 12-06, Revision 4, Section 3.2.1.3, list of initial event conditions and assumptions in TR-0816-50797 and stated that NEI 12-06 provides a generic list of event initial conditions and assumptions to apply while determining the baseline coping capability.

In TR-0816-50797, the applicant stated, in part, that plant equipment that is designed to be robust, with respect to design-basis external events, is assumed to be fully available, and plant equipment that is not robust is assumed to be unavailable. In TR-0816-50797, the applicant defined robust as the design of SSCs either meets the current plant design basis for the applicable external hazards, or the current NRC design guidance for the applicable hazard (e.g., RG 1.76, Revision 1), or the design has been shown by analysis or test to meet or exceed the current design basis.

TR-0816-50797, initial event assumption Number 1, assumes that the initial condition is a loss-of-offsite power at all units at a plant site resulting from an external event that affects the offsite power system either throughout the grid or at the plant with no prospect for recovery of offsite power for an extended period. In TR-0816-50797, the applicant stated, in part, that the EHVS (13.8 kilovolt and switchyard system), EMVS (medium voltage ac electrical distribution system), and ELVS (low voltage ac electrical distribution system) are assumed to be unavailable per initial event assumption number 1.

TR-0816-50797 initial event assumption Numbers 2, 5, and 6, shown below, relate to the electric power systems:

- Number 2 states that station batteries and associated dc buses remain available.
- Number 5 states that plant equipment that is contained in structures with designs that are robust for the applicable hazard is available.
- Number 6 states that installed electrical distribution systems, including inverters and battery chargers, remain available, provided they are protected consistent with current station design.

In accordance with initial event assumption Numbers 2, 5, and 6, the applicant concluded that the “highly reliable” dc power system (EDSS) batteries and the associated distribution system will survive the BDBEE and remain fully available. TR-0816-50797 states that the EDSS is the source of 125-volt dc power to plant loads such as the module protection system (MPS), the plant protection system, the neutron monitoring system, the SDIS, and the main control room (MCR) emergency lighting. TR-0816-50797 notes that the EDSS design consists of two separate and independent portions, both of which are expected to be available in a BDBEE; those portions are known as EDSS-common (EDSS-C) and EDSS-module specific (EDSS-MS).

TR-0816-50797 notes that the EDSS-MS and EDSS-C SSCs are qualified to seismic Category I standards and located within seismic Category I areas of the RXB and CRB.

The initial conditions and assumptions in NEI 12-06, Revision 4, and RG 1.226, Revision 0, assume that station batteries, which are usually safety related, would remain available following a BDBEE since they are considered robust. While the EDSS batteries are not "safety related," DCA Part 2, Tier 2, Chapter 8, "Electric Power," asserts that the EDSS SSCs will be qualified to seismic Category I standards and installed in Class I structures that are considered robust. Therefore, the staff finds that the initial assumptions described in TR-0816-50797 following a BDBEE are consistent with the NEI 12-06, Revision 4, guidance, which is endorsed by RG 1.226.

In TR-0816-50797, the applicant stated that the UHS is the only plant system heat sink credited for coping with a loss of all ac power. The UHS is a large pool of water consisting of the combined water volume of the reactor pool, the refueling pool, and the SFP and is housed in the RXB. In 10 CFR 50.155(e), the NRC requires, in part, that each licensee provide reliable means to remotely monitor the wide-range water level for each SFP at its site until 5 years have elapsed since all of the fuel within that SFP was last used in a reactor vessel for power generation. NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,"" Section 3.6, which is endorsed by RG 1.227, Revision 0, states that the normal electrical power supply for each channel shall be provided by different sources such that the loss of one of the channels' primary power supply will not result in a loss of the power supply function to both channels of SFP level instrumentation. All channels of SFP level instrumentation shall provide the capability of connecting the channel to a source of power (e.g., portable generators or replaceable batteries) independent of the normal plant ac and dc power systems.

In DCA Part 2, Tier 2, Section 20.1, the applicant explained that power for the four UHS level instruments during a BDBEE scenario is provided from the EDSS-C. The redundant SFP level instruments are powered from separate electrical buses. Additionally, a replaceable battery that is isolated from faults on the normal power supply provides an alternate source of power independent from the plant ac and dc power systems. DCA Part 2, Tier 2, Section 9.2.5.6.2, "Level Instrumentation," states that the electrical distribution system allows for the connection of alternate power to power the SFP level instruments.

In DCA Part 2, Tier 2, Section 8.3.2, "Direct Current Power Systems," the applicant noted that the 72-hour battery duty cycle for the EDSS-C provides a minimum of 72 hours of power for equipment supporting postaccident monitoring (PAM). In response to a BDBEE, TR-0816-50797 notes that the EDSS-C provides power to the instrumentation and controls equipment to track PAM variables. The PAM variables are provided in TR-0816-50797 and include SFP water level, which is monitored by the UHS instruments following a BDBEE. According to DCA Part 2, Tier 2, Section 8.3.2.2.2, "Onsite Direct Current Power System Conformance with Regulatory Framework," the EDSS batteries will be sized per the Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 485-1997, "IEEE Recommended Practice for Sizing of Lead-Acid Batteries for Stationary Applications," as endorsed by RG 1.212, "Sizing of Large Lead-Acid Storage Batteries," issued November 2008. The staff reviewed the ratings for major dc equipment (i.e., loads) provided in DCA Part 2, Tier 2, Tables 8.3-4, "Highly Reliable Direct Current Power System - Common Nominal Loads," and 8.3-5, "Highly Reliable Direct Current Power System - Module Specific Nominal Loads," and "highly reliable" dc power system major component data (nominal values) in Table 8.3-3, "Highly Reliable Direct Current Power System Major Component Data Nominal Values," of DCA Part 2, Tier 2, Section 8.3.2, for the EDSS-C and EDSS-MS batteries. Based on its review of this information, the staff finds that EDSS-C and EDSS-MS batteries should have adequate capacity to supply the expected loads for 72 hours following a BDBEE, given the stated ampere-hour capacity of the EDSS-C and EDSS-MS batteries in DCA Part 2, Tier 2, Chapter 8. In addition, per the staff's disposition of Condition 4.2 in Section 1.4.3.2.2 of this SER, the qualification testing plan for the EDSS batteries is to include environmental and seismic qualification and a technical functional requirement for valve-regulated lead-acid batteries to show they can perform as intended. The technical functional requirement for these batteries will demonstrate that they have sufficient capacity and capability to perform their intended function. DCA Part 2, Tier 2, Section 8.3.2.1.1, "Highly Reliable Direct Current Power System," further states that qualification provisions are applied to the EDSS.

The staff notes that the SFP level instruments can be powered by the EDSS or a replaceable battery that is independent of the NuScale electrical distribution system. As the staff concluded above, the EDSS system is capable of powering the SFP level instruments for 72 hours following a BDBEE. The applicant stated in DCA Part 2, Tier 2, Section 20.1.4.2, "Description," that "[p]ower to the redundant level instruments is from separate bus sources such that the loss of one supply will not result in a loss of power supply function to both divisions of UHS level instrumentation." While the staff's review is limited to the first 72 hours following a BDBEE, the staff notes that the replaceable battery is intended to supply at least 14 days of SFP level monitoring. The replaceable batteries will allow for easy replacement to power UHS monitoring level instruments indefinitely. The replaceable batteries are independent from the normal power supply. Accordingly, the staff has determined that the electrical power supply for the SFP level instrumentation follows the guidance provided by RG 1.227, Revision 0, and therefore, satisfies 10 CFR 50.155(e) as it relates to power supplies for the first 72 hours following a BDBEE.

In DCA Part 2, Tier 2, Section 20.1.4, "Spent Fuel Pool Instrumentation," the applicant stated that the UHS level instruments and their power supplies, EDSS-C, are physically and electrically separated and independent. In DCA Part 2, Tier 2, Section 20.1.4.2, the applicant stated that "power to the redundant level instruments is from separate bus sources such that the loss of one supply will not result in a loss of power supply function to both divisions of UHS level instrumentation." The staff's evaluation of electrical separation, which considers the EDSS batteries' capability to supply power to the UHS level instruments, is provided in Section 8.3 of this report.

EDSS-C also provides power for emergency lighting. The evaluation of emergency lighting is discussed in Section 9.5.3 of this report.

*Conclusion for Assessment of Electrical Power*

The staff has reviewed the capability and capacity of the NuScale electrical power system design to supply power to electrical equipment (e.g., instrumentation, lighting, ECCS solenoid valves) for the first 72 hours following a BDBEE. As described above, the staff confirmed that instrumentation other than SFP level instrumentation is not relied upon for the mitigation of beyond-design-basis events for core cooling and containment functions for the initial 72 hours. In addition, the staff confirmed that the capability and capacity of the NuScale electrical power system design is adequate to supply power to electrical equipment during a BDBEE, providing additional assurance that systems can respond as designed. In contrast, the staff concludes that the electrical power system design for SFP level instrumentation conforms to the guidance in RG 1.227, Revision 0, on power supplies and satisfies 10 CFR 50.155(e), as it relates to the capability and capacity of the NuScale electrical power system design to support SFP monitoring for the first 72 hours following a BDBEE.

*20.1.4.3 Mitigating Strategies for a Loss-of-All-Alternating-Current-Power Event*

The staff reviewed the following regulatory requirements in this section of the SER:

- The provisions in 10 CFR 50.155(b)(1) require applicants and licensees to develop, implement, and maintain strategies and guidelines to mitigate BDBEEs from natural phenomena. These strategies and guidelines are developed assuming a loss of all ac power concurrent with (for passive reactor designs) a loss of normal access to the normal heat sink. The provisions of 10 CFR 50.155(b)(1) require that the strategies and guidelines be capable of being implemented sitewide and include maintaining or restoring core cooling, containment, and SFP cooling capabilities (see 10 CFR 50.155(b)(1)(i)).

- The provisions in 10 CFR 50.155(c)(1) require that equipment relied on for the mitigation strategies and guidelines of 10 CFR 50.155(b)(1) must have sufficient capacity and capability to perform the functions required by 10 CFR 50.155(b)(1).

- The provisions in 10 CFR 50.155(e) require that each licensee shall provide reliable means to remotely monitor the wide-range water level for each SFP at its site until 5 years have elapsed since all of the fuel within that SFP was last used in a reactor vessel for power generation.

*20.1.4.4 Three Key Safety Functions*

In TR-0816-50797, Section 5.0, the applicant described that, to develop a mitigation strategy, the baseline coping capability of the NuScale plant design must be determined. The determination is made by evaluating the status of the three key safety functions stated in 10 CFR 50.155(b)(1)(i) during the integrated plant response to a loss of all ac power.

The staff's review of the three key safety functions of core cooling, containment, and spent fuel cooling is in the following SER sections: Section 20.1.4.4.1 for core cooling, Section 20.1.4.4.2 for containment capability, and Section 20.1.4.4.3 for SFP cooling. Specifically, the staff reviewed whether the design capacities and capabilities of the permanently installed SSCs in the NuScale design, as described in the FSAR, are capable of providing adequate core cooling, containment, and SFP cooling consistent with the requirements of 10 CFR 50.155(b)(1)(i) and (c)(1) for 72 hours following a BDBEE.

20.1.4.4.1 Core Cooling

The regulations in 10 CFR 50.155 require, in part, development of guidance and strategies to maintain or restore core cooling following a BDBEE. The strategies must be capable of mitigating a loss of all ac power.

NEI 12-06, Table 3-2, "PWR FLEX Baseline Capability Summary," provides some examples of acceptable methods for demonstrating the baseline coping capability of the reactor core strategies to maintain reactor core cooling safety functions during a loss-of-all-ac-power event.

NuScale DCA Part 2, Tier 2, Section 20.1.3, states the plant response to a loss-of-all-ac-power event for the first 72 hours is identical to an SBO event and is described in DCA Part 2, Tier 2, Section 8.4, “Station Blackout.” NuScale used the NRELAP5 code to predict the plant response to the SBO event and described it in DCA Part 2, Chapter 15, “Transient and Accident Analyses.” The staff reviewed the SBO assumptions, as described in NuScale’s DCA Part 2, Chapter 8, in Chapter 8 of this report. SBO transient analysis assumptions are consistent with NEI 12-06 initial conditions, such as all 12 modules operating at full power, loss of all ac power, dc power available, and no manual operator actions. As described in DCA Part 2, Tier 2, Section 8.4 and Section 20.1, during a loss of all ac power, the reactor is automatically tripped with all control rods fully inserted upon receipt of an MPS actuation signal due to high pressurizer pressure. After the CIVs close, the reactor core cooling is maintained by the automatic opening of the DHRS valves. For approximately 24 hours, reactor core decay heat is removed by natural circulation through the steam generator (SG) and DHRS heat exchanger and rejected to the reactor pool, which serves as the UHS. DHRS cooling results in a continuing decrease in reactor coolant system (RCS) pressure and temperature. At approximately 24 hours, the ECCS is actuated by the MPS timer, which removes EDSS power to the ECCS trip valves, which results in the subsequent opening of the ECCS reactor vent valves and reactor recirculation valves. As a result, reactor pressure vessel (RPV) pressure rapidly decreases as the containment pressure increases until equilibrium is reached and the ECCS becomes the primary method of decay heat removal from the RPV to the containment, where heat is transferred through the containment wall to the UHS.

The staff conducted an audit of documents that supported the applicant’s conclusions in DCA Part 2, Tier 2, Chapter 20, and TR-0816-50797, related to extended long-term core cooling as it relates to core cooling (ADAMS Accession No. ML19151A658) under both DHRS and ECCS cooling. The staff also conducted a follow-on audit to better understand the detailed calculations, analyses, and bases underlying NuScale’s SBO transient analysis and thermal-hydraulic parameters for specific time frames during the first 72 hours (see the audit plan at ADAMS Accession No. ML18348B076). The staff confirmed that the DHRS and ECCS provide adequate core cooling, including during the timeframe prior to ECCS actuation when the riser level falls below the DHRS inlet. The applicant’s analysis results for DHRS thermal-hydraulic performance, including the potential for RCS natural circulation oscillations, is presented in DCA Part 2, Tier 2, Section 5.4.4, “Decay Heat Removal System.”

Sections 15.0.5 and 15.0.6 of this SER evaluate the potential for boron redistribution due to boron volatility and potential stratification in the core region. Boron redistribution due to extended DHRS or ECCS operation without operator action to inject boron via the chemical and volume control system (CVCS) could result in a subsequent return to power, which could exacerbate boron redistribution, relocate boron outside the core region over time, and challenge core cooling while on extended ECCS or DHRS operation. This issue is described in Sections 15.0.5 and 15.0.6 of this SER using Chapter 15 methodologies, assumptions, and acceptance criteria for long-term cooling. Although all control rods are assumed to insert into the core in response to BDBEE conditions, boron redistribution is a physical phenomenon that could potentially cause recriticality, even with nominal assumptions with no operator action, as discussed in DCA Part 2, Tier 2, Section 15.0.6, “Evaluation of a Return to Power.” Review of the net margin to criticality in DCA Part 2, Tier 2, Section 4.3, Table 4.3-4, “Reactivity Requirements for Long Term Shutdown Capability,” which assumes all rods are inserted, demonstrates margin to a recriticality. This subcritical margin is sufficient to assure that boron redistribution over a 72-hour timeframe will not cause a core recriticality with all rods fully inserted. As discussed in Section 20.1.1 of this report, the staff is limiting its review to the first 72 hours of the applicant’s mitigation strategy, and any credible transient phenomena beyond 72 hours following a BDBEE that could challenge core cooling must be addressed by the COL applicant.

Regarding core cooling instrumentation, NuScale DCA Part 2, Tier 2, Section 20.1.3, states that monitoring instrumentation remains available in the MCR for at least 72 hours to provide additional assurance that systems have responded as designed. Consistent with RG 1.226, monitored parameters that provide additional assurance to a plant operator that core cooling is established, as indicated in DCA Part 2, Tier 2, Table 20.1-1, “Core Cooling Parameters,” and includes DHRS and ECCS valve position, RPV water level, core exit temperature, and containment water level. The instrumentation and display equipment associated with core cooling are reasonably protected consistent with RG 1.226 and designed to the environmental conditions of a loss-of-ac-power event (see the “Assessment of Electrical Power” subsection discussed above in this report). Therefore, the staff expects that core cooling instrumentation will remain available for the initial 72 hours and provides additional assurance that the systems have responded as designed.

Based on the discussion above, the staff finds that the applicant’s approach to core cooling in response to a loss-of-all-ac-power event for 72 hours following a BDBEE is consistent with staff guidance contained in RG 1.226 and is therefore acceptable. Specifically, the applicant’s design and approach to maintain core cooling has sufficient capacity and capability for 72 hours using installed equipment. Therefore, the staff finds the applicant’s design and approach are capable of providing adequate core cooling using installed equipment, consistent with the provisions of 10 CFR 50.155(b)(1)(i) and 10 CFR 50.155(c)(1), for 72 hours following a BDBEE. The period beyond 72 hours following a BDBEE will be more heavily dependent on provisions contained in operating programs (e.g., continuing the development of strategies and guidelines associated with 10 CFR 50.155(b)) and thus more appropriate for a COL applicant to address.

20.1.4.4.2 Containment

The regulation in 10 CFR 50.155 requires, in part, that an applicant or a licensee develop, implement, and maintain strategies and guidelines for maintaining or restoring containment capabilities following a BDBEE. The strategies and guidelines are developed assuming a loss of all ac power concurrent with a loss of normal access to the normal heat sink for passive reactor designs.

In RG 1.226, the staff endorsed industry guidance, NEI 12-06. Within NEI 12-06, Table 3-2 provides examples of acceptable approaches for demonstrating the capability of containment strategies to maintain containment during an extended loss of ac power. One such approach is to perform an analysis demonstrating that containment pressure control is not challenged.

In NuScale DCA Part 2, Tier 2, Section 20.1, the applicant described that the containment function is automatically established and passively maintained by safety-related equipment.

In NuScale DCA Part 2, Tier 2, Section 20.1, the applicant referenced TR-0816-50797. TR-0816-50797 provides information that supports the containment function described in DCA Part 2, Tier 2, Section 20.1. TR-0816-50797 is also incorporated by reference into the DCA, as identified in DCA Part 2, Tier 2, Table 1.6-2, “NuScale Referenced Technical Reports.”

In TR-0816-50797, Section 5.3, “Containment,” the applicant described the containment response following an extended loss of all ac power. TR-0816-50797 describes that the containment function is established and maintained by plant safety-related systems for many days following an extended loss of all ac power without operator action. In addition, TR-0816-50797 describes that containment temperature and pressure do not approach values that could adversely affect the integrity of containment. The justification for stating that the containment function is maintained over the long term is based on the analysis of the cooling capability of the ECCS, which includes the containment vessel walls and the reactor pool (i.e., UHS). The staff's review of the core cooling capability of the ECCS is discussed above in the "Core Cooling" subsection of this report.

In the NuScale DCA Part 2, Tier 2, Section 20.1, the applicant described the first 72 hours of an extended loss of ac power as identical to an SBO. As discussed in the "Core Cooling" subsection of this report, the applicant's SBO analysis is referenced by calculations supporting the conclusions in DCA Part 2, Tier 2, Section 20.1. In NuScale DCA Part 2, Tier 2, Section 8.4, the applicant described an integrated plant response to a loss of all ac power with assumptions that are consistent with NEI 12-06 initial conditions (e.g., all reactors operating at power) and NuScale's mitigating strategies in TR-0816-50797. Specifically, during a loss of all ac power, the CIVs automatically close following receipt of an MPS actuation signal, which establishes the containment of reactor coolant. The closing of safety-related CIVs occurs without operator action and requires no electrical power. CIV valve position indication is available in the control room to verify valve closure. Indication is powered from the EDSS. Additional discussion regarding the capability of CIVs (e.g., seismic, environmental) can be found in the "Equipment" subsection discussed above in this report.

In TR-0816-50797, the applicant described the containment response to a loss of all ac power, which results in high-energy fluid (i.e., reactor coolant) being released into the NuScale CNV when the ECCS valves open. The ECCS valves open at approximately 24 hours after a loss of all ac power due to MPS actuation. During the 24-hour period before the ECCS valves opening, a significant amount of cooling occurs by passive heat transfer between the RCS and the UHS (i.e., reactor pool) through operation of the DHRS. Therefore, by the time the ECCS valves open at approximately 24 hours, the pressure in the reactor vessel is significantly reduced. The results of the applicant's containment analysis during a loss of all ac power is depicted in DCA Part 2, Tier 2, Section 8.4, Figure 8.4-3, "Station Blackout Containment Vessel Pressure," and Figure 8.4-4, "Station Blackout Containment Vessel Temperature." As shown in these figures, NuScale's peak containment pressure and temperature in response to a loss of all ac power occurs when the ECCS valves open. These peak containment conditions are well below the containment design limits and continue to decrease over time as the natural circulation process is established through operation of the ECCS. Because the analysis of a loss of all ac shows the containment pressure and temperature conditions remaining well below containment design limits, the applicant concluded that containment pressure control is not challenged, and the containment function is maintained.

Regarding containment instrumentation, NuScale DCA Part 2, Tier 2, Section 20.1.3, states that monitoring instrumentation remains available in the MCR for at least 72 hours to provide additional assurance that systems have responded as designed. Consistent with RG 1.226, monitored containment parameters provide additional assurance to a plant operator that the containment function is established, as indicated in DCA Part 2, Tier 2, Table 20.1-2, "Containment Parameters," and include CIV position indication and containment pressure. Moreover, TR-0816-50797, Section 4.9, "Safety Display and Indication System," also describes that installed instrumentation can provide 72 hours of module monitoring, which provides additional ability to verify that containment functions are established following a BDBEE. In the NuScale design, instrumentation associated with CIV position and containment pressure are safety related and can be expected to remain available after an event that leads to the loss of all ac power. Although containment instrumentation is not relied upon for the mitigation of beyond-design-basis events for the initial 72 hours, containment instrumentation is available and provides additional assurance that systems have responded as designed. Information on containment key parameters (e.g., containment valve position) is available in the control room as long as electrical power is available. Electrical power is expected to be available for a minimum of 72 hours (see the “Assessment of Electrical Power” subsection discussed above in this report).

Based on the discussion above, the staff finds that the applicant’s approach to maintaining containment capabilities in response to a loss-of-all-ac-power event for 72 hours following a BDBEE is consistent with staff guidance contained in RG 1.226 and is acceptable because the applicant’s design and approach have sufficient capacity and capability to maintain containment for 72 hours using installed plant equipment. Specifically, the analyzed peak containment pressure and temperature conditions remain below the containment design limits using installed plant equipment. Therefore, the staff finds the applicant’s design and approach are capable of maintaining containment using installed plant equipment, as required by the provisions of 10 CFR 50.155(b)(1)(i) and 10 CFR 50.155(c)(1), for 72 hours following a BDBEE. The period beyond 72 hours following a BDBEE will be more heavily dependent on provisions contained in operating programs (e.g., continuing the development of strategies and guidelines associated with 10 CFR 50.155(b)) and thus more appropriate for a COL applicant to address.

#### 20.1.4.4.3 Spent Fuel Pool Cooling

The regulation in 10 CFR 50.155 requires, in part, that an applicant or a licensee develop, implement, and maintain strategies and guidelines for maintaining or restoring SFP cooling capabilities following a BDBEE. The strategies and guidelines are developed assuming a loss of all ac power concurrent with a loss of normal access to the normal heat sink for passive reactor designs.

DCA Part 2, Tier 2, Section 20.1.3, states the following:

> The SFP cooling function is maintained by submergence of the spent fuel in the UHS.
>
> - The SFP, as part of the UHS, communicates with the refueling pool and reactor pool above the SFP weir wall. As such, the pools respond as a single volume during a loss of all ac power until UHS level lowers below the weir wall.
> - The UHS inventory maintains passive cooling of the spent fuel in the SFP for more than 150 days following initiation of a loss of all ac power without pool inventory makeup or operator action.

DCA Part 2, Tier 2, Section 9.2.5.2.3, states the following:

> During an event where loss of electric power occurs, the volume of water already in the pool provides the inventory for the necessary heat removal. Upon loss of power, the reactor pool cooling and SFP cooling systems shut down. The UHS water expands as it heats and eventually begins to boil. Heat continues to be removed from the pool through boiling and evaporation, removing enough heat to maintain the spent fuel and fuel in the NPMs sufficiently cool to prevent fuel

damage. The design is such that UHS water boil-off will continue to remove heat from the power modules and spent fuel.

DCA Part 2, Tier 2, Section 20.1.4, provides information on level instruments in the SFP.

TR-0816-50797, Section 4.11, states that Section 9.2.5 of the FSAR describes the UHS system and summarizes that description. The staff's evaluation of the heat removal functional capability of the UHS is in Section 9.2.5 of this SER, where the thermal analysis of the UHS for extended loss of ac power is discussed with the design-basis assumptions (such as initial pool level and temperature) being more limiting than the analysis for a loss of all ac power. As discussed in SER Section 9.2.5 for design-basis accidents, the UHS is designed with sufficient water inventory to remove the heat from the power modules and spent fuel for more than 72 hours by water boiling off without the need for operator action, makeup water, or electric power. For a loss of all ac power, all NuScale Power Modules (NPMs) are in orderly shutdown and cooldown, which means less heat load as compared to the heat load being assumed in SER Section 9.2.5. Accordingly, the staff finds that the heat load resulting from a loss of all ac power is bounded by the accident being evaluated in SER Section 9.2.5 for the first 72 hours following the event. The SFP cooling capability of the permanently installed SSCs in the NuScale design for a loss of all ac power is adequate for 72 hours without water makeup. Therefore, the NuScale design is consistent with the requirements of 10 CFR 50.155(b)(1) with respect to SFP cooling capability for 72 hours.

In SER Section 9.2.5, as related to the conformance of GDC 2, the staff reviewed and concluded that the SFP as a portion of the UHS is adequately designed to protect against the effects of external events such as earthquakes, tornados, hurricanes, and floods to remain functional following the events. The SFP makeup capability included in the design is the UHS makeup line. The applicant stated that the makeup line is designed to be seismic Category I and is protected from external natural phenomena. Based on the above, the staff finds that the NuScale SFP design provides reasonable protection of the equipment for SFP cooling in accordance with the requirements of 10 CFR 50.155(c).

The only plant parameter used to ensure SFP cooling is maintained is SFP level indication. The staff determined that, even though the thermal analysis indicates there will be sufficient water in the UHS to remove the heat load as designed for an extended period, monitoring capability of the SFP level is required at all times during a loss of all ac power to confirm and ensure the cooling function is being performed. The review of the SFP level instrumentation relative to 10 CFR 50.155(e) is documented in Section 20.1.4.6 of this SER.

Beyond 72 hours after a BDBEE, the staff notes that the plant operators will need to replace batteries for the SFP level instrument and to supply makeup water for the SFP before the SFP water level is insufficient to maintain spent fuel cooling. The COL applicant will need to provide procedures and operator training according to 10 CFR 50.155(b)(1)(ii).

#### *20.1.4.5 Ventilation Capability*

As discussed in Section 20.1.4.3 of this SER, the staff finds that the NuScale design has sufficient capacity and capability to maintain core cooling, containment, and SFP cooling for 72 hours after a BDBEE using installed equipment and without operator action. However, while not relied on for mitigation strategies and guidelines, NuScale DCA Part 2, Tier 2, Section 20.1.3, states that monitoring instrumentation (the SDIS) is maintained in the MCR for 72 hours to provide additional assurance that systems have responded as designed. Therefore, the staff reviewed the ventilation design description outlined in DCA Part 2, Tier 2, Section 6.4, “Control Room Ventilation,” and Section 9.4.1, “RXB Ventilation,” to confirm that the monitoring functions will be available for 72 hours following an extended loss of all ac power.

The staff reviewed the applicant’s information for consistency with NEI 12-06, Revision 4, which is endorsed by RG 1.226. The guidance in NEI 12-06, Revision 4, Section 3.2.1.8, “Effects of Loss of Ventilation,” is to verify that the effects of loss of heating, ventilation, and air conditioning in an extended loss-of-all-ac-power event can be addressed consistent with NUMARC 87-00, “Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors,” issued August 1991, or by plant-specific thermal-hydraulic calculations. NUMARC 87-00, Section 2.7, “Effects of Loss of Ventilation,” discusses technical bases for equipment operability outside containment and control room habitability. The staff audited NuScale’s EC-B060-4543, “GOTHIC Passive Cooling of NuScale Control Room Building,” as well as TR-0816-50797, and confirmed that equipment qualification is not challenged under the loss of all ac power or SBO scenario (ADAMS Accession No. ML19308A061). Specifically, the staff reviewed the input to the code and confirmed that the input was in the appropriate ranges.

At the initiation of a loss-of-all-ac-power event, the control room ventilation system stops performing heating, ventilation, and air conditioning functions, the control room envelope (CRE) is isolated, and control room habitability system (CRHS) actuates to pressurize the MCR. As a result, the environmental conditions in the MCR are maintained by passive cooling and the flow of breathing air from the CRHS.

The CRHS SSCs that provide breathing air inventory to the CRE for 72 hours are specified to be designed to seismic Category I criteria. These SSCs are the air storage bottles and the supply piping and components (including the regulating valves and actuation valves) to the CRE. The CRE isolation dampers and pressure relief piping and components are also specified to be designed to seismic Category I criteria.

As stated in DCA Part 2, Tier 2, Section 6.4.2.3, “Off-Normal Operation,” there is an external air connection point that will allow the connection of a post-72-hour air supply from offsite air bottles to supply air and pressurization to the CRE for extended accident conditions if needed.

Based on the above discussion, the staff determined that the MCR is habitable for the first 72 hours following a BDBEE, which enables operator monitoring of instrumentation from the MCR. The staff finds that the applicant’s approach described above is consistent with the guidance found in NEI 12-06, as endorsed by RG 1.226, and, therefore, the staff finds the ventilation capability adequate to support monitoring functions for 72 hours following an extended loss of ac power.

#### *20.1.4.6 Spent Fuel Pool Level Instrumentation*

RG 1.227, Revision 0, endorses, with exceptions and clarifications, the methods and procedures promulgated by the NEI in document NEI 12-02, Revision 1, issued August 2012, as a process the NRC staff considers acceptable for meeting certain regulations in 10 CFR 50.155.

NEI 12-02, Section 2.1, defines three water levels trained personnel shall reliably identify:

(1) level that is adequate to support operation of the normal fuel pool cooling system

(2) level that is adequate to provide substantial radiation shielding for a person standing on the SFP operating deck

(3) level where fuel remains covered and actions to implement makeup water addition should no longer be deferred

In DCA Part 2, Tier 2, Section 20.1.4, the applicant discussed how the UHS level instruments meet the requirements of 10 CFR 50.155(e). DCA Part 2, Tier 2, Section 20.1.4.1, "Design Bases," describes the design basis of the four UHS level instruments. It also states that all four instruments are capable of monitoring Levels 1 and 2. The two SFP level instruments are capable of monitoring Level 3, if water level drops below the weir wall elevation. The level instrument in the reactor pool and the level instrument in the refueling pool areas are capable of monitoring the level of the water above the fuel in the reactor core when the NPM is disassembled in the refueling pool during refueling.

To identify the three key water levels in NEI 12-02, DCA Part 2, Tier 2, Section 20.1.4.1, defines the three water levels and references Table 9.2.5-1, "Relevant Ultimate Heat Sink Parameters," for further details. The staff reviewed the definition of the three levels and DCA Part 2, Tier 2, Table 9.2.5-1, and confirmed that Section 20.1.4.1 identifies the three key water levels as described in NEI 12-02 and therefore satisfies 10 CFR 50.155(e) in regard to ensuring the level instruments provide full coverage indication over the three significant levels identified.

20.1.4.6.1 Instruments

Section 3 of NEI 12-02, Revision 1, describes the types of instruments that can be used to monitor the SFP water level. NEI 12-02 states that a reliable level indication shall be provided for each SFP that can be used in responding to BDBEEs.

DCA Part 2, Tier 2, Section 20.1.4.2, states that the UHS pools are provided with four (two in the SFP, one in the refueling pool, and one in the reactor pool) wide-range instruments capable of monitoring the water level from the top of the stored fuel to the operating deck.

The staff noted that the two SFP instrument channels are designed to monitor the SFP level from the top of the stored fuel to the operating deck. The level instrument channels located in the refueling pool and the reactor pool are able to monitor the SFP water level from the top of weir (connecting the SFP to the other UHS pools) up to the operating deck. The staff evaluated the applicant's instrument descriptions and determined that crediting these four permanently installed instruments as primary and backup channels follows the guidance provided by NEI 12-02 and satisfies 10 CFR 50.155(e) in regard to providing at least one primary and one backup instrument channel.

20.1.4.6.2 Arrangement

NEI 12-02, Section 3.2, indicates that the arrangement recommendations provide reasonable measure for separation and missile protection for permanently installed instrumentation.

In DCA Part 2, Tier 2, Section 20.1.4.2, the applicant stated that SFP level instruments are separated to reduce the potential for falling debris or missiles affecting both channels of instrumentation. In addition, the other two instruments in the UHS are located in separate pools, which also provides missile protection.

The staff evaluated the applicant’s description of the level instruments and determined that the equipment description would ensure that the SFP level instruments are arranged in a manner that provides reasonable protection against missiles; therefore, the staff concludes that these features follow the guidance provided by NEI 12-02 and satisfies 10 CFR 50.155(e) in regard to arrangement.

20.1.4.6.3 Mounting

NEI 12-02 states that the mounting of permanently installed instruments shall be designed consistent with the highest seismic or safety classification of the SFP.

In DCA Part 2, Tier 2, Section 20.1.4.1, the applicant indicated that the level instruments are designed to seismic Category I standards and the mounting and associated cabling are installed as seismic Category I.

The staff evaluated the applicant’s description of the level instruments and determined that the equipment description and the seismic classification of the components are acceptable to assure that the SFP level instruments are mounted in a manner that provides reasonable protection against seismic events; therefore, the staff concludes that these features follow the guidance in NEI 12-02 and satisfies 10 CFR 50.155(c) in regard to mounting.

20.1.4.6.4 Qualification

NEI 12-02, Section 3.4, states that the instrument channel reliability shall be demonstrated via an appropriate combination of design, analyses, operating experience, and testing of channel components.

In DCA Part 2, Tier 2, Section 20.1.4.2, the applicant indicated that the level instruments and the associated cabling are qualified to perform their functions under the following environmental conditions:

- safe-shutdown earthquake seismic event (seismic Category I)
- concentrated borated water environment
- maximum temperature of approximately 100 degrees Celsius (212 degrees Fahrenheit ) and 100 percent relative humidity
- boiling water or steam environment
- radiological conditions existing from a normal refueling with a freshly discharged fuel batch that remains covered with SFP water (Level 3)

In TR-0816-50797, Revision 2, Section 4.9.2, “Equipment Qualification,” the applicant stated that the SDIS is qualified to seismic Category I requirements and is housed in the concrete, seismic Category I portions of the CRB. In TR-0816-50797, Revision 2, Section 4.11.2, “Equipment Qualification,” the applicant stated that the four pool level instruments are seismically mounted, environmentally qualified, and designed to meet the guidance of NEI 12-02. NEI 12-02 provides guidance for SFP instrumentation design features to ensure that reliable level indication is provided for each SFP in response to BDBEEs.

The pool instruments and cabling are subject to a harsh environment as shown in DCA Part 2, Tier 2, Table 3.11-1, "List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments." Additionally, DCA Part 2, Tier 2, Section 3.11.2.1, "Environmental Qualification of Electrical Equipment," states, "Electrical equipment identified to be in a harsh location…are [sic] environmentally qualified by type testing or type testing and analysis using the guidance of IEEE Std. 323-1974."

Additionally, DCA Part 2, Tier 2, Table 3.2-1, states that the pool level instruments are classified as augmented quality (AQ-S) in accordance with IEEE Std. 497-2002 CORR 1, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," which incorporates IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." The staff evaluated the applicant's description of the level instruments provided in the DCA and determined that the qualification measures applicable to the pool level instruments presented are consistent with the criteria discussed in the guidance and that the instruments will be qualified in accordance with IEEE Std. 323-1974, which is endorsed by the NRC in RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 1, issued June 1984. Therefore, the staff concludes that these qualification measures follow the guidance provided by NEI 12-02 and satisfies 10 CFR 50.155(e) in regard to qualification.

#### 20.1.4.6.5 Independence

NEI 12-02, Section 3.5, states that independence of permanently installed instrumentation and primary and backup channels is obtained by physical and power separation commensurate with the hazard and electrical isolation needs.

DCA Part 2, Tier 2, Section 20.1.4, states that all the instrument channels are physically and electrically independent. The instrument channels are permanently installed and power supplies to these instruments are addressed in DCA Part 2, Tier 2, Section 8.3.2. The staff finds that the design description follows the guidance provided by NEI 12-02 and satisfies 10 CFR 50.155(e) in regard to independence.

#### 20.1.4.6.6 Power Sources

NEI 12-02, Section 3.6, states that the normal electrical power supply for each channel shall be provided by different sources such that the loss of one of the channels' primary power supply will not result in a loss of the power supply function to both channels of SFP level instrumentation. All channels of SFP level instrumentation shall provide the capability of connecting the channel to a source of power (e.g., portable generators or replaceable batteries) independent of the normal plant ac and dc power systems.

As discussed previously, in Section 20.1.4.2 of this report, the staff evaluated the level instrument power supply and connections and determined that the power supply for the SFP level instrument channel follows the guidance provided by NEI 12-02 and satisfies 10 CFR 50.155(e) in regard to power supply.

#### 20.1.4.6.7 Accuracy

NEI 12-02, Section 3.7, states that the instrument channels shall maintain their designed accuracy following a power interruption or change in power source without recalibration.

DCA Part 2, Tier 2, Section 20.1.4.2, the applicant stated that the level instruments are designed to maintain the minimum accuracy following a power interruption or change in power source without recalibration.

The staff evaluated the applicant's description of the level instruments and determined that the level instruments were designed to retain calibration following a power interruption; therefore, the staff concludes that this feature follows the guidance provided by NEI 12-02 and the level instruments are in conformance with 10 CFR 50.155(e) in regard to accuracy.

#### 20.1.4.6.8 Testing

NEI 12-02, Section 3.8, indicates that static or nonactive installed (fixed) sensors should be designed such that testing and calibration can be performed in situ.

DCA Part 2, Tier 2, Section 20.1.4.2, indicates that the level instruments are designed to allow for testing and calibration in situ. The applicant also proposed COL Item 20.1-8, which states that the COL applicant will develop procedures and a training and qualification program for operations, maintenance, testing, and calibration of UHS level instrumentation.

The staff reviewed the applicant's system description and noted that the permanently installed instrument channels are normally used to monitor the SFP level and will be subject to routine testing and calibration in accordance with plant procedures. The staff evaluated the applicant's proposed COL Item 20.1-8 and agrees that COL applicants must develop a procedure for the maintenance, testing, and calibration of the level instruments to ensure each instrument is maintained reliable during the plant life. Accordingly, the staff concludes that these design features follow the guidance provided by NEI 12-02 and, thus, satisfies 10 CFR 50.155(e) in regard to testing.

#### 20.1.4.6.9 Display

NEI 12-02 states that the SFP level indication from the installed channel shall be displayed in the control room, at the alternate shutdown panel, or another appropriate and accessible location.

DCA Part 2, Tier 2, Section 20.1.4.2, indicates that the level instruments provide display in the MCR and the remote shutdown station. The instruments also initiate high- or low-level alarms, both locally and in the MCR, to alert operators of pool level conditions.

The staff evaluated the applicant's description of the level instruments and determined that the instruments are designed to provide indication of the pool water level in the MCR and the remote shutdown station; therefore, the staff concludes that these features follow the guidance provided by NEI 12-02 and satisfies 10 CFR 50.155(e) in regard to displays.

*20.1.4.7 Conclusion for Spent Fuel Pool Level Instrumentation*

The staff evaluated the information provided in the applicant's DCA Part 2, Tier 2, Section 20.1.4, and TR-0816-50797, Revision 1, related to the SFP water level instrumentation and, for the reasons stated above, determined that the proposed level instruments are designed in accordance with the guidance provided in NEI 12-02 and will have the capacity to monitor SFP water level for 72 hours after a BDBEE. In addition, as described above, these instruments are reliable, able to withstand beyond-design-basis natural phenomena, and capable of monitoring key SFP level parameters following a BDBEE, in accordance with the guidance in RG 1.227. In view of the above, the staff concludes that the SFP level instrumentation satisfies 10 CFR 50.155(e) for 72 hours following a BDBEE.

### 20.1.5 Combined License Information Items

NEI 12-02, Section 4, addresses the training, procedures, testing, and calibration of the SFP level instrument channels. DCA Part 2, Tier 2, Section 20.1, proposed COL Item 20.1-8 for a COL applicant to address the development of procedures, training, and qualification program for operations, maintenance, testing, and calibration of UHS level instruments.

**Table 20.1-2 NuScale Combined License Information Items for DCA Part 2, Tier 2, Section 20.1**

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| 20.1-8 | A COL applicant that references the NuScale Power Plant design certification will develop procedures, training, and a qualification program for operations, maintenance, testing, and calibration of ultimate heat sink level instrumentation to ensure the level instruments will be available when needed and personnel are knowledgeable in interpreting the information as addressed in NEI 12-02, Revision 1, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation." | 20.1 |

The staff reviewed the COL item and determined that developing a qualification program for operations, maintenance, testing, and calibration of UHS level instrumentation is the responsibility of the COL applicant and that the COL applicant will address these requirements of 10 CFR 50.155(e).

### 20.1.6 Conclusion

The staff finds, as documented in the review above, that the applicant has provided sufficient information to demonstrate that the design is capable of providing adequate core cooling, containment, SFP cooling, and SFP level instrumentation for 72 hours following a BDBEE and meets the requirements of 10 CFR 50.155(b)(1)(i) and 10 CFR 50.155(c) regarding these design capacities and capabilities, and 10 CFR 50.155(e) for SFP monitoring, for 72 hours following a BDBEE.

## 20.2 <u>Loss of Large Areas of the Plant due to Explosions and Fires</u>

### 20.2.1 Introduction

This section documents the staff's evaluation of how the design and mitigation strategies for the NuScale Power Plant meet the requirements of 10 CFR 50.155(b)(2).

### 20.2.2 Summary of Application

**DCA Part 2, Tier 1:** There is no Tier 1 information for this area of review.

**DCA Part 2, Tier 2:** DCA Part 2, Tier 2, Section 20.2, "Loss of Large Areas of the Plant due to Explosions and Fires," describes the results of the NuScale Power Plant response to a LOLA event given in TR-0816-50796 with no security-related information.

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

**Technical Specifications:** There are no generic TS for this area of review.

**Technical Reports:** TR-0816-50796 documents an assessment evaluating the NuScale Power Plant response to a LOLA event using the guidance in NEI 06-12, "B.5.b Phase 2 and 3 Submittal Guideline," Revision 3, issued 2009. The report defines LOLA criteria and identifies the design features that meet those criteria and expected COL applicant requirements.

### 20.2.3 Regulatory Basis

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

In 10 CFR 50.155(b)(2), the NRC requires each licensee to develop and implement guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with LOLAs of the plant due to explosions or fire and to include strategies in the following areas:

- firefighting
- operations to mitigate fuel damage
- actions to minimize radiological release

The guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 19.4, "Strategies and Guidance to Address Loss of Large Areas of the Plant due to Explosions and Fires," lists acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

### 20.2.4 Technical Evaluation

The staff reviewed the DCA in accordance with SRP Section 19.4. The staff considers conformance with the guidance in NEI 06-12, Revision 3, as an acceptable method in satisfying the Commission's requirements in 10 CFR 50.155(b)(2).

*20.2.4.1 Identification of Key Safety Functions*

The applicant stated that the generic pressurized-water reactor (PWR) key safety functions identified in NEI 06-12, Section 4.2.3.1, "Identification of Key Safety Functions," are all applicable to the NuScale Power Plant. The PWR key safety functions are as follows:

- RCS inventory control
- RCS heat removal
- containment isolation
- containment integrity
- release mitigation

The applicant did not identify any new key safety functions for the NuScale Power Plant. NEI 06-12 states that, for each safety function, the applicant should identify the minimal set of equipment for both a primary and an alternate means of satisfying the key safety function.

The applicant stated that the NuScale design maintains RCS inventory, RCS heat removal, containment isolation, and containment integrity without additional strategies. The applicant stated further that the primary and alternate means to maintain these functions are spatially separated in accordance with the guidance of NEI 06-12.

Evaluation of Key Safety Function—Reactor Coolant System Inventory Control

The purpose of this safety function is to ensure that the core remains covered with water. The applicant stated that the NPM does not require water injection to keep the core covered during transient or accident scenarios. Containment isolation is the primary means for RCS inventory control. During a loss of the reactor pressure barrier, RCS inventory is maintained within the CNV if containment is isolated. When the ECCS or DHRS are actuated, the core will remain covered.

The ECCS functions by recirculating coolant condensed in containment and returning it to the RPV utilizing natural recirculation. The ECCS is a safety-related system available to provide heat removal from the RPV. The ECCS can provide decay heat removal for at least 72 hours without modulation, automatic or manual, after the initiating event. All safety-related functions of ECCS are designed using passive operating principles and fail in an actuated configuration upon initiation signal or loss of dc power. No electrical power or support systems are required for successful operation of the ECCS.

The DHRS is a passive core cooling safety system consisting of two independent and redundant trains. Each train alone has the capability to provide sufficient heat removal to satisfy the safety function. The DHRS functions by utilizing the SGs to remove heat from the primary coolant. The generated steam is condensed and returned to the SG as subcooled liquid. Flow is driven by natural circulation and heat is rejected to the UHS.

An alternate means is the use of the CVCS. The CVCS is a means of RCS inventory control during normal operation, startup, and shutdown and is described in DCA Part 2, Tier 2, Section 9.3.4, "Chemical and Volume Control System." Success of this system to perform this key function requires at least one CVCS makeup pump running. The CVCS makeup pumps for NPMs 1 through 6 are located in the north gallery space, and the CVCS makeup pumps for NPMs 7 through 12 are located in the south gallery space of the RXB at elevation10.7 meters (m) (35 feet (ft)). The RXB elevation of 10.7 m (35 ft) is underground and protected from external events. The CVCS makeup pumps are provided with backup power via the backup power supply system (described in DCA Part 2, Tier 2, Section 8.3.1.1.2, “Backup Power Supply System”). Accordingly, the CVCS is acceptable as an alternate means of providing this safety function.

The staff finds this acceptable because the applicant identified the primary and alternate means of meeting this safety function consistent with NEI 06-12.

Evaluation of Key Safety Function—Reactor Coolant System Heat Removal

The purpose of this key safety function is to remove the decay heat from the core and transfer it to the UHS. The applicant stated that the primary means for heat removal during steady-state, startup, and hot-shutdown operations are the energy conversion systems.

The energy conversion systems, which includes condensate and feedwater, main steam, and turbine generator systems, are the means of heat removal during normal plant operations. Successful operation requires an open flowpath with at least one of three condensate pumps and one of three feedwater pumps operating. An open flowpath means steam flows from the SGs to the condenser, as well as condensate flows from the condenser back to the SGs. A containment isolation signal will isolate this flowpath. Supporting systems include condenser air removal, instrument air, circulating water, and site cooling water systems.

The condensate and feedwater pumps and associated electrical and instrumentation wiring for NPMs 1 through 6 are located in the north turbine generator building and for NPMs 7 through 12 in the south turbine generator building. Both turbine generator buildings are located less than 91 m (100 yards) away from the RXB. Support systems are also located in the turbine generator building, security owner-controlled area, and protected area. The safety-related feedwater and main steam isolation valves are located under the biological shield on top of the CNV.

The alternate means for RCS heat removal is the DHRS, followed by the ECCS, when the ECCS valves open. Evaluation of the DHRS and ECCS is described above.

The staff finds this acceptable because the applicant identified the primary and alternate means of meeting this safety function consistent with NEI 06-12.

Evaluation of Key Safety Function—Containment Isolation

The purpose of this key safety function is to ensure no leakage paths exist that would allow gaseous and particulate radiation to escape containment. The applicant stated that the NuScale Power Plant relies on containment isolation to accomplish this function. Containment isolation is performed by the containment system. The CIVs for interfacing systems, with the exception of feedwater and main steam, have dual-valve, single-body CIVs outside of the containment. This isolation capability is located under the biological shield. This dual containment isolation capability may be considered the primary and alternate means of performing this function. A loss of dc power to those valves will result in their repositioning to their safe or accident response position.

The staff finds this acceptable because the applicant identified the primary and alternate means of meeting this safety function consistent with NEI 06-12.

Evaluation of Key Safety Function—Containment Integrity

The purpose of this key safety function is to ensure the containment fission product barrier is maintained to minimize or prevent radiological release outside containment. The applicant stated that the CNV is designed for high pressures and passive heat transfer between the CNV shell and the UHS. The CNV temperature and pressure are maintained due to the partial immersion of the CNV in the UHS. Severe accident analyses have shown that, even during a severe accident where the core relocates to the bottom of the RPV, the RPV will not be breached and the CNV remains intact due to the passive cooling capabilities of the RPV and CNV containing the inventory from ECCS initiation or safety relief actuation. This function is accomplished passively by heat exchange between the RPV, CNV, and immersion in the UHS.

The guidance in NEI 06-12, Chapter 4, "Actions for New Plants," states that it is recognized that new plants typically have more safety trains that are more spatially separated than for current U.S. operating plants. Additionally, some new designs employ passive features that may be more or less susceptible to damage from the effects of large fires and explosions. Therefore, new plants may not need all of the mitigation strategies or may need additional strategies to satisfy the key safety functions. The staff reviewed information on the UHS and containment systems provided in NuScale's TR-0816-50796 and in DCA Part 2, Tier 2, Section 9.2.5, "Ultimate Heat Sink," and Section 6.2, "Containment Systems." The staff finds this acceptable because the applicant provided information on how this key safety function is accomplished by passive SSCs, which are less susceptible to damage than those in current operating plants.

Evaluation of Key Safety Function—Release Mitigation

The purpose of this key safety function is to minimize a radiological release, assuming severe core damage has occurred and a radiological release is imminent or in progress. The applicant stated that the RPV is located within the CNV and that the CNV is partially immersed in the UHS. The UHS is the primary means to perform this function. There is no explicit alternate means to perform this function.

The applicant stated that a release below the UHS water level will be scrubbed by the UHS. There are penetrations below the water level and a release at such a location is bounded by a release above the water line. A more likely and more conservative release point that would require mitigation is above the UHS water line, more specifically on the top of the CNV where there are numerous penetrations. These penetrations are located underneath the biological shield. If the release were to extend into the RXB, a release could occur through one of several RXB exterior openings.

Because the success criteria for meeting this safety function are not satisfied by existing redundant spatially separated equipment, the applicant provided a mitigation strategy compliant with NEI 06-12 guidance in Section 4.3.3.1 of TR-0816-50796, Revision 0. This strategy on portable sprays is evaluated in Section 20.2.4.2.3.7 of this SER.

The staff finds this acceptable because the applicant identified the primary, the UHS, and alternate means (portable sprays) of meeting this safety function consistent with NEI 06-12.

#### *20.2.4.2 Mitigation Strategies*

##### 20.2.4.2.1 Phase 1—Enhanced Firefighting Capabilities

NEI 06-12 guidance lists 31 Phase 1 firefighting and operational strategies that should be considered by an applicant when developing its mitigative strategies to meet the requirements of 10 CFR 50.155(b)(2). The applicant stated that all Phase 1 strategies listed in NEI 06-12, except the enhancement for supplying the fire protection ring header, will be COL items.

Evaluation—Supplying the Fire Protection Ring Header

One of the Phase 1 firefighting mitigative strategies listed in NEI 06-12 is to develop a means for an alternate water supply feed for the fire protection yard main loop in the event that the normal water supply source is lost.

The applicant stated that the NuScale Power Plant features that address the firefighting capabilities for a LOLA event are included in the fire protection system (FPS) design. Specifically, the FPS includes an underground yard fire main loop. Hydrants are provided on the yard fire main loop in accordance with National Fire Protection Association (NFPA) Standard 24, "Standard for the Installation of Private Fire Service Mains and Their Appurtenances," at intervals up to 76 m (250 ft) and located on all four sides of the RXB. The lateral to each hydrant is controlled by an isolation valve. The NuScale design will successfully support supplying the underground fire water ring main using a portable diesel-driven pump. External water sources available for makeup to the yard fire main loop are the two fire protection supply tanks that each contain at least 1.14 x $10^6$ liters (300,000 gallons) of water. There are several connections in the yard main that can support supplying the yard main using a portable diesel-driven pump and valves that can isolate damaged section(s) when required. The staff finds this acceptable because it follows the guidance in NEI 06-12.

20.2.4.2.2 Phase 2—Measures to Mitigate Damage to Fuel in the Spent Fuel Pool

NEI 06-12, Section 2.0, "Spent Fuel Pool Strategies," states that the SFP strategies are not required for sites that have SFPs that are below grade and cannot be drained. The applicant stated that the NuScale SFP is located below grade and cannot be drained.

The applicant stated that the bottom elevation of the SFP is located at RXB elevation 7.6 m (25 ft), and the SFP operating deck is located at RXB elevation 30.5 m (100 ft). Grade elevation for the RXB is elevation 30.5 m (100 ft). Therefore, the SFP is located below grade. The SFP has a normal water level of approximately RXB elevation 28.7 m (94 ft). If the SFP is breached, the SFP water inventory would drain into the gallery rooms outside the SFP at RXB elevation 7.3 m (24 ft) and above. An analysis shows that the minimum water level in the SFP after a maximum postulated drain-down event is approximately 15.2 m (50 ft) RXB elevation, which results in approximately 7.6 m (25 ft) of water in the SFP. This level adequately covers the spent fuel assemblies and provides a margin of coverage above the top of spent fuel. The minimum level in the pool for adequate spent fuel dose rate shielding is 6.1 m (20 ft).

Because the SFP is below grade and cannot be drained below a level that does not adequately cover the spent fuel, the staff finds that the SFP strategies are not required for the NuScale design.

20.2.4.2.3 Phase 3—Measures to Mitigate Damage to Fuel in the Reactor Vessel and to Minimize Radiological Release

The NEI 06-12 guidance lists eight Phase 3 strategies that should be considered by an applicant when developing its mitigative strategies to meet the requirements of 10 CFR 50.155(b)(2). Phase 3 mitigative strategies are intended to restore or maintain core cooling to mitigate potential damage to fuel in the reactor system and to mitigate potential radiological releases through the containment. The applicant stated that six of the strategies do not pertain to the NuScale design. The applicant has provided design enhancements to support the portable spray strategy, as described in Section 20.2.4.2.3.7 of this SER. The applicant stated that the strategy for “Command and Control EDMG” will be a COL item.

#### 20.2.4.2.3.1 Evaluation—Makeup to Refueling Water Storage Tank to Supply Emergency Core Cooling System Long Term

As described in Section 3.3.1 of NEI 06-12, the objective of this strategy is to provide a large-volume makeup source to the reactor water storage tank (or equivalent) to supply the ECCS for the long term.

The applicant stated that the NuScale design does not require RCS makeup. The NuScale design does not use refueling water storage tanks or equivalent. All safety-related functions of the ECCS are designed using passive operating principles and fail in an actuated configuration upon an initiation signal or loss of dc power. No electrical power or support systems are required for successful operation of the ECCS. Because the NuScale design does not require RCS makeup, the staff finds that the applicant does not need a strategy for this item.

#### 20.2.4.2.3.2 Evaluation—Manually Depressurize Steam Generators to Reduce Inventory Loss

As described in Section 3.3.2 of NEI 06-12, the objective of this strategy is to provide a power-independent means to depressurize SGs by locally, manually opening atmospheric dump valves (or SG-operated relief valves) to reduce SG pressure and RCS temperature and pressure.

The applicant stated that the NuScale design does not depressurize the SGs during an accident. Adequate heat removal is achieved using passive systems. Because the NuScale design does not require depressurization of SGs to reduce inventory loss, the staff finds that the applicant does not need a strategy for this item.

#### 20.2.4.2.3.3 Evaluation—Manual Operation of Turbine-Driven Pumps

As described in Section 3.3.3 of NEI 06-12, the objective of this strategy is to provide a power-independent means to provide core cooling and prevent or delay core damage.

The applicant stated that the NuScale design does not include an auxiliary feedwater system. No turbine-driven, safety-related pumps are included in the design. Adequate heat removal is achieved using passive systems. Because the NuScale design does not and need not include an auxiliary feedwater system or turbine-driven, safety-related pumps to provide adequate heat removal, the staff finds that the applicant does not need a strategy for this item.

#### 20.2.4.2.3.4 Evaluation—Manually Depressurize Steam Generators and Use Portable Pump

As described in Section 3.3.4 of NEI 06-12, the objective of this strategy is to provide a low-pressure makeup source to provide SG makeup and core cooling.

The applicant stated that the NuScale design does not depressurize the SGs during an accident. Adequate heat removal is achieved using passive systems. Because the NuScale design does not require depressurization of SGs to reduce inventory loss, the staff finds that the applicant does not need a strategy for this item.

20.2.4.2.3.5 Evaluation—Makeup to Condensate Storage Tank/Auxiliary Feedwater Storage Tank

As described in Section 3.3.5 of NEI 06-12, the objective of this strategy is to provide a makeup source to the condensate storage tank (CST)/auxiliary feedwater storage tank (AFWST) to supply auxiliary feedwater for the long term.

The applicant stated that the NuScale design does not include a CST or an AFWST. Adequate heat removal is achieved through passive systems and does not require the addition of water to the SGs. Because the NuScale design does not need a CST or an AFWST to provide makeup water, the staff finds that the applicant does not need a strategy for this item.

20.2.4.2.3.6 Evaluation—Containment Flooding with Portable Pump

As described in Section 3.3.6 of NEI 06-12, the objective of this strategy is to provide a power-independent means to inject water into the containment to flood the containment floor and cover core debris.

The intent of this strategy is to flood containment after severe core damage and RPV failure and is typically a backup strategy for containment sprays or ECCS injection systems. The applicant stated that the NuScale design does not require a containment spray system because the containment floods as a result of ECCS actuation or the lifting of safety relief valves. The CNV is partially immersed in the UHS. The amount of inventory in the UHS and the prevention of pool draining inherent in the plant design preclude the need for an additional mitigation strategy. Because the containment floods as a result of ECCS actuation or lifting of safety relief valves, and since the CNV is partially immersed in the UHS, the UHS inventory is very large, and the design prevents pool draining, the staff finds that the applicant does not need a strategy for this item.

20.2.4.2.3.7 Evaluation—Portable Sprays

As described in Section 3.3.7 of NEI 06-12, the objective of this strategy is to provide a means to reduce the magnitude of any fission product releases by spraying.

The applicant indicated that the FPS has standpipe hose connections in accordance with NFPA Standard 14, “Standard for the Installation of Standpipe and Hose Systems,” and RG 1.189, “Fire Protection for Nuclear Power Plants,” Revision 2, issued October 2009. The applicant described the FPS as follows. Standpipes are installed within each stairway and exit corridors. Also, standpipes, hose connections, and hydrants are provided for manual firefighting in areas containing equipment required for safe plant shutdown and in yard areas of the plant. Therefore, the applicant expected that the standpipe connections may be used to supply water to portable monitor nozzles for use in spraying potential release points to reduce fission product releases. The applicant stated that a portable pump will also be available and will be used to spray a radiological release, including plant structures that cannot be sprayed from the FPS due to physical layout or equipment limitations. The FPS is described in DCA Part 2, Tier 2, Section 9.5.1, “Fire Protection Program.” Procedures and guidance for this strategy will be developed by the COL applicant to support implementation. The staff finds the standpipes, hose connections, hydrants, and portable pumps acceptable because the applicant’s strategy follows the guidance in NEI 06-12, Section 3.3.7.

### 20.2.5 Combined License Information Items

Table 20.2-1 lists the COL information item numbers and descriptions related to a LOLA, from DCA Part 2, Tier 2, Table 1.8-2.

**Table 20.2-1 NuScale Combined License Information Items for DCA Part 2, Tier 2, Section 20.2**

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| 20.2-1 | A COL applicant that references the NuScale Power design certification will develop enhanced firefighting capabilities by implementing the guidance in NRC guidance document "Developing Mitigating Strategies/Guidance for Nuclear Power Plants to Respond to Loss of Large Areas of the Plant in Accordance with B.5.b of the February 25, 2002, Order" dated February 25, 2005 (Reference 20.2-3). The enhanced firefighting capabilities should address the expectation elements listed in Section 4.1.3 of the Technical Report TR-0816-50796. | 20.2 |
| 20.2-2 | A COL applicant that references the NuScale Power design certification will provide a means for water spray scrubbing using fog nozzles and the availability of water sources, and address runoff water containment issues (sandbags, portable dikes, etc.) as an attenuation measure for mitigating radiation releases outside containment. | 20.2 |

### 20.2.6 Conclusion

Based on the staff's review of the information provided by the applicant, the staff concludes that the applicant has adequately followed the guidance of SRP Section 19.4 and NEI 06-12. The staff finds, as documented in the review above, that the applicant has provided sufficient information regarding Phase 1, 2, and 3 design enhancements at this stage that could be used by a COL applicant in developing its mitigative strategies to meet the requirements of 10 CFR 50.155(b)(2).