June 02, 2025 Docket No. 99902078

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Approved "-A" Version of Topical Report "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-A, Revision 1

REFERENCES:

1. NRC email to NuScale, "Final Safety Evaluation for NuScale TR-141299, Rev 1, NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events defined by 10 CFR 50.155 Prop and non-Prop," dated May 16, 2025
2. NuScale letter to NRC, "NuScale Power, LLC Submittal of Topical Report "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P, Revision 1 and Docketing of Resolved Audit Responses, dated June 26, 2024 (ML24178A398)

By referenced email dated May 16, 2025 (Reference 1), the NRC issued a final safety evaluation report documenting the NRC Staff conclusion that the NuScale topical report "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299, Revision 1, is acceptable for referencing in licensing applications for the NuScale small modular reactor design. Reference 1 requested that NuScale publish the approved version of TR-141299 as soon as possible.

Enclosure 1 contains the proprietary version of the report entitled, "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P-A, Revision 1. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version of the report, TR-141299-NP-A, Revision 1.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360.0500 Fax 541.207.3928
www.nuscalepower.com

If you have any questions, please contact Kristopher Cummings at 240-833-3003 or kcummings@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 02, 2025

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P-A, Revision 1, Proprietary Version
Enclosure 2: "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-NP-A, Revision 1, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-182501

Enclosure 1:

"NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P-A, Revision 1, Proprietary Version

NuScale Power, LLC
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Enclosure 2:

"NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-NP-A, Revision 1, Nonproprietary Version

Contents

Section	Description
A	Letter from NRC to NuScale, “Final Safety Evaluation for NuScale TR-141299, Rev 1, NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events defined by 10 CFR 50.155 Prop and non-Prop,” dated May 16, 2025 (nonproprietary)
B	NuScale Topical Report: “NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155,” TR-141299-NP-A, Revision 1
C	Letter from NuScale to the NRC, Docketed Audit Responses on the NuScale Topical Report, “NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155,” TR-141299-NP, Revision 1

Section A

From: Thomas Hayden <Thomas.Hayden@nrc.gov>
Sent: Friday, May 16, 2025 9:56 AM
To: Regulatory Affairs
Cc: Griffith, Thomas; Cummings, Kristopher; Shaw, Peter; Mahmoud -MJ- Jardaneh; Getachew Tesfaye
Subject: Final Safety Evaluation for NuScale TR-141299, Rev 1, NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events defined by 10 CFR 50.155 Prop and non-Prop
Attachments: Final Safety Evaluation of the NuScale Power, LLC, Topical Report TR-141299, Revision 1, NuScale Power Plant Design Capability to MBDBE defined by 10 CFR 50 - 051625.docx; Final Safety Evaluation of the NuScale Power, LLC, Topical Report TR-141299, Revision 1, NuScale Power Plant Design Capability to MBDBE defined by 10 CFR 50 (proprietary) - 051625.docx

By letter dated June 26, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML24178A399 (Proprietary) and ML24178A398 (non-Proprietary)), NuScale Power, LLC (NuScale), submitted Topical Report (TR) TR-141299, Revision 1, "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155" to the U.S. Nuclear Regulatory Commission (NRC). The NRC staff has prepared a final safety evaluation for TR-141299, Revision 1. The non-proprietary (ML25136A034) and proprietary (ML25136A035) final safety evaluations are enclosed. The NRC staff has found that TR-141299, Revision 1, is acceptable for referencing in licensing applications for the NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the enclosed final safety evaluation.

The NRC staff requests that NuScale publish the accepted version of this TR as soon as possible following receipt of this electronic mail. The accepted version shall incorporate this electronic mail and the enclosed final safety evaluation after the title page. It must be well indexed such that information is readily located. Also, it must contain historical review information, including NRC requests for additional information and accepted responses. The accepted version of the TR shall include a "-A" (designated accepted) following the report identification number.

If the NRC's criteria or regulations change such that the NRC staff's conclusion in this electronic mail (that the TR is acceptable) is invalidated, NuScale and/or the applicant referencing the TR will be expected either to revise and resubmit its respective documentation or to submit justification for continued applicability of the TR without revision of the respective documentation.

If you have any questions or comments concerning this matter, I can be reached at (301) 415-2956 or via e-mail address at Thomas.Hayden@nrc.gov. The attached documents are both password protected. Password to follow in a separate email.

Docket No. 99902078

Sincerely,

Tommy Hayden
Project Manager

Email: thomas.hayden@nrc.gov
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

SAFETY EVALUATION BY THE U.S. NUCLEAR REGULATORY COMMISSION

TOPICAL REPORT TR-141299, REVISION 1

“NUSCALE POWER PLANT DESIGN CAPABILITY TO MITIGATE BEYOND-DESIGN-BASIS EVENTS DEFINED BY 10 CFR 50.155”

<u>NUSCALE POWER, LLC</u>

1.0 INTRODUCTION

1.1 Summary

By letter dated September 11, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23254A360), NuScale Power, LLC (NuScale), submitted, for U.S Nuclear Regulatory Commission (NRC) staff review and approval, Topical Report (TR) TR-141299-P, Revision 0, "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155" (Ref. 1). The NRC staff conducted an audit for TR-141299-P, Revision 0, starting on December 20, 2023 (Ref. 2) and concluding on March 30, 2024. On June 26, 2024, NuScale submitted Revision 1 of TR-141299-P (Ref. 3), hereafter referred to as "the TR".

1.2 Scope of the NRC Staff's Review

The purpose of the TR, as stated in its section 1.1, "Purpose," is to describe the NuScale plant design capability to mitigate beyond-design-basis events (BDBEs) as defined in Title 10 of the *Code of Federal Regulations* (CFR), section 50.155 "Mitigation of beyond-design-basis events," specifically, (i) the plant response to the loss of all alternating current power concurrent with loss of normal access to the normal heat sink and (ii) the design capability to mitigate the loss of large plant areas due to explosions or fire.

NuScale requested NRC review and approval of the TR for NuScale Power Plant capability to mitigate beyond-design-basis events and stated in TR section 1.3, "Conditions of Use," that an adopter of the TR must provide (a) plant specific design information that includes the **{{ }}** described in the TR, (b) a plant specific thermal analysis demonstrating the plant's capability to cope with BDBEs with **{{ }}** equipment identified in the TR, (c) a maintenance rule program in accordance with 10 CFR 50.65, "requirements for monitoring the effectiveness of maintenance at nuclear power plants," and (d) an emergency plan in accordance with 10 CFR 50.160, "Emergency preparedness for small modular reactors, non-light-water reactors, and non-power production or utilization facilities" or 10 CFR 50.47(b), and appendix E to Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities."

The NRC staff's review of the TR was limited to the generally non-specific information provided in the TR, and the staff's approval of the TR is subject to the limitations and conditions listed in section 5.0, "Limitations and Conditions," of this report. Accordingly, future applicants who wish to utilize the TR for their plants will be required to provide additional information, detailed in the limitations and conditions in section 5.0, in order to receive NRC approval of the use of the topical report in their applications.

2.0 BACKGROUND

2.1 Regulatory Requirements and Relevant Regulatory and Industry Guidance

The TR was developed to describe the NuScale design capabilities to mitigate beyond-design-basis events as defined by 10 CFR 50.155.

Applicable Regulations

- 10 CFR 50.155 requires applicants and licensees subject to 10 CFR Part 50, “Domestic licensing of production and utilization facilities,” and all applicants and licensees for a power reactor combined license under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” to develop, implement, and maintain strategies and guidelines to mitigate a beyond-design-basis event.

Related Guidance

The staff used the following guidance during the review of the TR:

- Regulatory Guide (RG) 1.226, “Flexible Mitigation Strategies for Beyond-Design-Basis Events,” Revision 0 (Ref. 4, ML19058A012), identifies methods and procedures to demonstrate compliance with 10 CFR 50.155
- Nuclear Energy Institute (NEI) 12-06, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” Revision 4 (Ref. 5, ML16354B421), which is endorsed by RG 1.226, provides industry guidance in meeting 10 CFR 50.155
- SECY-19-0066, “Staff Review of NuScale Power’s Mitigation Strategy for Beyond-Design-Basis External Events” (Ref. 6, ML19148A443)
- NUREG-0800, Standard Review Plan (SRP) Section 19.4, “Strategies and Guidance to Address Loss of Large Areas of the Plant due to Explosions and Fires” (ML13316B202)

2.2 Summary of Technical Information

The TR contains a description of proposed NuScale design capabilities and features to mitigate BDBEs as defined by 10 CFR 50.155. Design features that provide enhanced capabilities for coping with an extended loss of electrical power, loss of normal access to the normal heat sink, and loss of large areas due to explosions or fire are discussed within the TR. 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 serving as the ultimate heat sink for the facility. The TR specifies how these features enable a design to mitigate BDBEs using **{{ }}** plant equipment for a specified extended duration without the need for alternating current power, special equipment, or additional guidelines and strategies. NuScale states that the design features described in the TR can maintain core cooling, containment, and spent fuel pool (SFP) cooling using **{{ }}**.

The TR does not provide thermal analyses; instead, the TR states that an adopter of the TR must provide plant specific thermal analysis demonstrating its capability to maintain core cooling, containment, and SFP cooling for **{{ }}** identified in the TR.

3.0 <u>TECHNICAL EVALUATION</u>

NuScale states that design characteristics include the following:

> Containment cooling is designed to be maintained for at least **{{ }}** without pool inventory makeup or operator action. Therefore, the capability of the containment cooling supports the ability of the design to **{{ }}**

{{ }} without predetermined supplemental actions.

The installed equipment is listed in TR table 4-1, {{ }} and is described in TR section 4.0, "Plant Systems and Responses to a Loss of All Alternating Current Power Event." In TR section 4.0, NuScale also states that {{ }} those described in the TR. This statement is consistent with the docketed response for audit issue, MBDBE.LTR-8 (Ref. 7), which clarified that {{ }}. The staff understands these statements to mean {{ }} in the design bases for plants adopting this TR.

The TR does not contain any supporting analyses or references to such analysis showing the basis for the above determination. In response to staff's audit questions, NuScale revised the TR to include four conditions of its use as stated in TR section 1.3: To adopt the described methodology for the response to a beyond-design-basis external event (BDBEE), an adopter of the topical report must provide: (1) A plant specific design that {{ }} described within this report. (2) A plant specific thermal analysis demonstrating {{ }}. This analysis addresses site specific conditions, including configuration of the plant with respect to the selected number of modules and spent fuel pool capacity, for all modes of operation (normal and refueling). (3) A maintenance rule program in accordance with 10 CFR 50.65. (4) An emergency plan in accordance with 10 CFR 50.160 or 50.47(b) and 10 CFR Part 50 Appendix E describing communications and coordination with local, state, federal, and tribal agencies. As written in Limitation and Condition no. 5.1, the staff determined that these same conditions of use are applicable. Furthermore, in Limitation and Condition no. 5.1 the staff clarifies that in order to satisfy Condition of Use #1 an applicant or licensee must provide a plant specific design that includes {{ }} with the described system functions listed in TR Table 4-1 and the system design features and equipment classifications as described in TR section 4.0 – 4.15 for each system

In TR section 3.2.4, "Monitoring," NuScale states that {{ }}.

NuScale design capabilities and features to mitigate BDBEs, as defined by 10 CFR 50.155, are based on a NuScale Power Module's (NPM) ability to {{ }}. The SFP is part of the UHS.

The TR is intended to be generically applicable to NuScale reactor designs with the capabilities and features discussed in the TR. NuScale has not provided supporting analyses to demonstrate the NPM's ability to withstand beyond design basis events in the TR, and a supporting analysis to demonstrate these capabilities is therefore required to be provided by an applicant or a licensee who adopts the TR.

The potential for the future submittal of various design features as well as the need for analysis to be provided by a COL applicant has caused the staff to impose limitations and conditions for using the TR, as provided in section 5.0.

The NRC staff's evaluation of TR sections 3.0, "Plant Baseline Coping Criteria for Loss of all AC Power," through 9.0,"Spent fuel pool monitoring after final fuel removal from the reactor vessel, as required by 10 CFR 50.155(e)," is provided in sections 3.1, "Plant Baseline Coping Criteria for Loss of all AC Power," through 3.7, "SFP monitoring after final fuel removal from the reactor vessel, as required by 10 CFR 50.155(e)," respectively, of this report.

3.1 Plant Baseline Coping Criteria for Loss of all AC Power

3.1.1 Assessment of Electrical Power

In the TR, section 1.2, "Scope," NuScale states that the report is applicable to NuScale small modular reactor designs that have structures, systems and components (SSCs) capable of performing their safety functions without off-site electric power or operator actions following a BDBE.

In the TR, section 3.1.5, "Initial Event Conditions and Assumptions," NuScale states that for the baseline coping capability, 1) station batteries and associated direct current (DC) buses remain available for the designed operating time of the station batteries and 2) installed electrical distribution system, including inverters and battery chargers, remain available provided they are seismic Category I. The initial conditions and assumptions in NEI 12-06, Revision 4 (Ref. 5), and RG 1.226, Revision 0 (Ref. 4), assume that station batteries would remain available following a BDBE since they are considered robust. While the augmented direct current system (EDAS) batteries are not "safety related," NuScale states, in section 4.3.2, "Equipment Qualification," of the TR, that the EDAS SSCs are located in Seismic Class I structures. Seismic category I buildings are designed to withstand design basis external events and to provide safety margins and conservatism to make structural failure unlikely. By placing station batteries inside seismic category I structures, they will be reasonably protected against natural hazards. Therefore, the staff finds that the initial assumptions described in TR following BDBEE are consistent with the NEI 12-06, Revision 4, guidance, which is endorsed by RG 1.226.

In section 3.2, "Plant Design Capabilities" of the TR, NuScale states that the first 72 hours of a loss of all alternate current (AC) power is identical to a station blackout (SBO) and no AC power is relied upon for performing safety functions. Following a loss of all AC power event and following the automatic response of safety-related equipment, NuScale states, in section 3.2 of the TR, that **{{**

}}.

In section 4.3, "Augmented Direct Current Power System" of the TR, NuScale states that EDAS provides power to plant loads including the module protection system (MPS), plant protection system (PPS), and safety display and indication system (SDIS) and augmented direct current power system – common (EDAS-C) services plant common loads, including main control room (MCR) emergency lighting and post-accident monitoring (PAM) variable indications displayed in the MCR. An applicant must satisfy Limitation and Condition no. 5.1, which states that an applicant using the TR must meet TR Section 1.3, “Conditions of Use”. In this regard, Condition of Use #1 requires an adopter of the TR to provide a plant specific design that **{{ }}** described within the TR. Limitation and Condition no. 5.1 expands on this requirement by specifying that the adopter must also ensure the plant design includes the described system functions, system design features, and equipment qualification as described in the TR. Further, Condition of Use #2 states that an adopter of the TR must provide a plant specific thermal analysis demonstrating maintenance of core cooling, containment and spent fuel cooling **{{ }}**. By satisfying this limitation and condition the augmented quality, capacity, and capability of the EDAS to provide power for a BDBE would be demonstrated.

In section 3.2.4 of the TR, NuScale states that **{{ }}**. The staff notes that UHS and SFP level instruments can be powered by EDAS, **{{ }}** and therefore, an applicant must satisfy Limitation and Condition no. 5.2. By satisfying this condition, an applicant would be able to address **{{ }}**. Accordingly, the staff has determined that the description of the electrical power supply for the SFP level instrumentation is consistent with the guidance provided by RG 1.227, “Wide-Range Spent Fuel Pool Level Instrumentation,” Revision 0 (Ref. 8). Therefore, an applicant or licensee following this TR and satisfying Limitation and Condition no. 5.2, would be able to demonstrate that its use of the NuScale design meets 10 CFR 50.155(e) as it relates to power supplies for the first 72 hours following a BDBE.

In section 4.6.1, “[SDIS] System Design,” of the TR, NuScale states that the SDIS provides accident monitoring functions, and that electrical power is provided to the SDIS from two separate and independent divisions of EDAS-C. In section 4.3.1, “[EDAS] System Design,” of the TR, NuScale states that EDAS-C power divisions have a specified minimum battery duty cycle of 72 hours.

The NRC staff concludes that the approach in the TR, which outlines the NuScale electrical power system design, would be acceptable to ensure that electrical power is supplied to electrical equipment (e.g., instrumentation, lighting, emergency core cooling system (ECCS) solenoid valves) for a BDBE leading to a loss of all AC power, provided the applicant satisfies Limitation and Condition no. 5.1. An applicant or licensee using a NuScale design will need to implement this limitation and condition when adopting this TR to demonstrate compliance with 10 CFR 50.155(e).

3.1.2 Plant Design Capabilities

The TR describes plant design capabilities for core cooling, containment, SFP cooling, and monitoring following a loss of all AC power event. TR section 3.2 states the following:

Following a loss of all AC power event, 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 and spent fuel into a safe, stable, shutdown state with passive cooling. **{{**

}}.

Core cooling—during a loss of all AC power event, containment isolation within **{{ }}** of the event preserves the reactor coolant inventory. The decay heat removal system (DHRS) passively removes decay heat for up to the first 24 hours following a loss of all AC power event. By 24 hours, the ECCS valves automatically open and the ECCS maintains core cooling for the extended loss of all AC power event.

Containment—As stated in TR section 3.2, the safety-related containment isolation valves (CIVs) and the containment vessel (CNV) provide passive containment isolation function without operator action or electrical power. Heat removal to the UHS passively controls temperature and pressure to ensure containment integrity. Peak pressure and temperature conditions for the CNV are designed to occur early in the event when the ECCS valves open and prevent a challenge to containment integrity. Containment cooling is designed **{{ }}**.

SFP Cooling—As stated in TR section 3.2, 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 event until the UHS level lowers to the weir wall. The UHS inventory is designed to maintain passive cooling of the spent fuel in the SFP **{{ }}**.

Monitoring—As stated in TR section 3.2, **{{ }}**. However, post-accident monitoring variable indications are maintained in the main control room for at least 72 hours to provide additional assurance that systems respond as designed.

The NRC staff concludes that the approach outlined in the TR would be acceptable to ensure plant design capabilities for core cooling, containment, SFP cooling, and monitoring following a loss of all AC power event, provided the applicant satisfies Limitation and Condition no. 5.1. An applicant or licensee using a NuScale design will need to implement this limitation and condition when adopting this TR to demonstrate compliance with 10 CFR 50.155(b)(1) for core cooling, containment, and SFP cooling and 10 CFR 50.155(b)(2)(e) for spent fuel pool monitoring.

3.2 Plant Systems and Responses to a Loss of All Alternating Current Power Event

Section 4.0, “Plant Systems and Responses to a Loss of All Alternating Current Power Event,” of the TR describes individual system responses to the event in order to provide an overview of the integrated plant response. The staff reviewed the descriptions of systems in accordance with 10 CFR 50.155.

10 CFR 50.155(c) states the following:

> (c) Equipment. (1) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must have sufficient capacity and capability to perform the functions required by paragraph (b)(1) of this section.
>
> (2) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.

The system qualification and availability for the NuScale design are laid out in TR section 4.0. As described in the TR, systems used to support the mitigating strategies are protected in seismic Category I structures and include protection from other applicable design basis hazards (such as wind and flood events) and are assumed to survive the BDBEE. Table 4-1 in the TR provides a summary of the functions of the **{{ }}** and locations that are relied upon for mitigation of the BDBEE.

The NRC staff concludes that the approach in the TR, which outlines the NuScale plant systems and responses to the BDBEE, would be acceptable to ensure that the plant systems are protected from BDBEE and should continue to perform to support their safety functions, provided the applicant satisfies Limitation and Condition no. 5.1. An applicant or licensee using a NuScale design will need to implement this limitation and condition when adopting this TR to demonstrate compliance with 10 CFR 50.155(c).

3.3 Safety Functions during a Loss of All Alternating Current Power

3.3.1 Systems and Safety Functions

Section 5.0, “Safety Functions during a Loss of All Alternating Current Power,” of the TR describes the safety functions for a loss of all AC power event. Section 5.0 of the TR describes each system and the safety function that it performs. The safety functions include Core Cooling, Containment, and SFP cooling.

10 CFR 50.155(b)(1) states, in part, the following:

> (b) *Strategies and guidelines*. Each applicant or licensee shall develop, implement, and maintain;
>
> (1) Mitigation strategies for [BDBEEs]… These strategies and guidelines must be capable of being implemented site-wide and must include the following:
> (i) Maintaining or restoring core cooling, containment, and spent fuel pool cooling capabilities;

In Section 5.1, “Integrated Plant Response,” the TR explains how the NuScale design is based on passive systems whose safety functions can be performed without operator intervention when initiated from 100 percent power. The TR describes how the safety functions are met during non-power operation modes as well. Initial coping aligns with the typical plant performance during a loss of all AC power. Indefinite coping is covered in section 3.3.2 of this SE.

The core cooling safety function was discussed in TR section 5.2, "Core Cooling." NuScale states that reactor coolant inventory is maintained as the NuScale Power Module (NPM) design would be isolated and inventory would be maintained within the containment vessel. The NPM passive plant design does not include reactor coolant pumps and therefore there is no Reactor Coolant Pump (RCP) seal leakage. The isolation of the CNV allows the water level to remain above the top of the active fuel. The potential exists for the addition of water when necessary.

NuScale states that reactivity control is maintained to continue safe shutdown conditions during the event, and the NPM design allows for shutdown through reactor trip and the control rod insertion. **{{**

}}. NuScale states in the TR that "[d]uring the loss of all AC power **{{**

}}. In SECY-19-0066, the staff stated that "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, then no additional review or approval of these capabilities would be required at the COL stage." In Limitation and Condition no. 5.1, the staff incorporates TR section 3, "Conditions of Use"; TR Condition of Use #2 states that "an adopter of the TR must provide a plant specific thermal analysis demonstrating its capability to **{{**

}}." Therefore, consistent with SECY-19-0066, the staff determines that subject to Limitation and Condition no. 5.1, an applicant or licensee referencing this TR would be required to address any credible transient phenomena (e.g., return to power) that could challenge core cooling, containment, or SFP cooling.

NuScale states that decay heat removal is passively accomplished in the NPM. NuScale explains that the DHRS allows the UHS inventory to remove heat from the CNV. The DHRS works in conjunction with the ECCS to remove the decay heat. Eventually, the UHS will begin to boil. The water level will then lower in the UHS. Eventually, water can be added to the UHS, if and when needed, through protected piping.

NuScale states that SFP cooling is accomplished through heat transfer to the UHS. The heat from the spent fuel is transferred to the SFP which is initially connected to the UHS. As the UHS boils, the SFP can become disconnected from the UHS by the SFP weir wall. Eventually, if and when needed, water can be added to the UHS which also adds water to the SFP.

The NRC staff concludes that the approach outlined in TR section 5.0, which includes a summary of the core cooling, containment, and SFP cooling functions in TR Table 5-3, "Baseline Coping Capability Summary," would be acceptable to ensure the NPM would be able to perform extended coping, provided the applicant satisfies Limitation and Condition 5.1. An applicant or licensee using a NuScale design will need to implement this limitation and condition, providing a plant specific design and analysis to support the expected coping period, including all modes of operation, when adopting this TR to demonstrate compliance with 10 CFR 50.155(b)(1).

3.3.2 Indefinite maintenance of core cooling, containment, and spent fuel pool cooling capabilities

The regulation at 10 CFR 50.155(b)(1) states that each applicant or licensee shall develop, implement, and maintain:

> Mitigation strategies for beyond-design basis external events—Strategies and guidelines to mitigate beyond-design-basis external events from natural phenomena that are developed assuming a loss of all ac power concurrent with either a loss of normal access to the ultimate heat sink or, for passive reactor designs, a loss of normal access to the normal heat sink. These strategies and guidelines must be capable of being implemented site-wide and must include the following:
>
> (i) Maintaining or restoring core cooling, containment, and spent fuel pool cooling capabilities; and
>
> (ii) The acquisition and use of offsite assistance and resources to support the functions required by paragraph (b)(1)(i) of this section indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies.

The statements of consideration for 10 CFR 50.155 in the *Federal Register* (84 FR 39684; August 9, 2019) (Ref. 9) state, in part (at 39709):

> The requirement to enable "the acquisition and use of offsite assistance and resources to support the functions required by § 50.155(b)(1)(i) of this section indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies" means that licensees need to plan for obtaining sufficient resources *(e.g.,* fuel for generators and pumps, cooling and makeup water) to continue removing decay heat from the irradiated fuel in the reactor vessel and SFP as well as to remove heat from containment as necessary until an alternate means of removing heat is established.
> ...More detailed planning for offsite assistance and resources is necessary for the initial period following the event; less detailed planning is necessary as the event progresses and the licensee can mobilize additional support for recovery.

Section 3.3, "Considerations in Utilizing Off-Site Resources," of NEI 12-06, Revision 4, states, in part:

> Since site access is considered to be restored to near-normal within 24 hours, by 72 hours from the event initiation, outside resources should be able to be mobilized by that time such that a continuous supply of needed resources will be able to be provided to the site. Within these first 72 hours a site will have deployed its FLEX strategies which should result in a stable plant condition on the FLEX equipment and plans will have been established to maintain the key safety functions for the long term. Therefore, FLEX strategies and/or resources are not required to be explicitly planned in advance for the period beyond 72 hours.

> The site will need to identify staging area(s) for receipt of the off-site FLEX equipment and a means to transport the off-site equipment to the deployment location.
>
> It is expected that the licensee will ensure the off-site resource organization will be able to provide the resources that will be necessary to support the extended coping duration.
>
> In addition, the licensee will need to ensure standard connectors for electrical and mechanical FLEX equipment compatible with the site connections are provided.

Section 1.1.1.3, "Final Phase," of RG 1.226, Revision 0, states:

> The final phase will be accomplished using the onsite equipment augmented with additional equipment and consumables obtained from off-site until power, water, and coolant injection systems are restored or commissioned.
>
> Staff Position: NEI 12-06, Revision 4, Section 3.0, provides an acceptable method for determining the baseline coping capabilities for the final phase. NEI 12-06, Revision 4, Section 12.2, provides an acceptable method for establishing the capability to obtain equipment and consumables from off-site until power, water, and coolant injection systems are restored or commissioned. This provides an acceptable method to sustain the listed functions indefinitely when coupled with the restoration or commissioning of power, water, and coolant injection systems.

The NRC-endorsed guidance in NEI 12-06, Revision 4, section 3.3, recognizes that site access is expected to be restored within 24 hours from the event initiation and that off-site resources should be able to be mobilized by 72 hours such that FLEX strategies and/or resources are not required to be "explicitly planned" for the period beyond 72 hours. However, this guidance also presumes that the licensee will identify staging areas to receive off-site resources and the means to transport the equipment to areas where it is to be deployed, that the licensee will ensure the ability of an off-site organization to provide the necessary resources to support the extended coping duration, and that standard connectors for electrical and mechanical FLEX equipment that are compatible with the site connections are obtained.

The staff position in RG 1.226, section 1.1.1.3 assumes that various mitigating strategies have been implemented prior to long-term coping such that the site has established the capability to receive and utilize future, unplanned resources. Therefore, some degree of planning and preparation will be needed **{{ }}** to ensure that unplanned off-site resources can be identified, obtained, and implemented at the site **{{ }}**.

3.3.2.1 UHS Make Up

10 CFR 50.155(b)(1)(ii) requires the capability to acquire and use off-site assistance and resources to support the functions described in 10 CFR 50.155(b)(1)(i) indefinitely, or until mitigation strategies are no longer needed. The statements of consideration for 10 CFR 50.155

make it clear that licensees need to plan for obtaining sufficient resources to maintain these mitigating capabilities until alternate means of heat removal are established.

The TR executive summary states the following:

> The indefinite core cooling, containment, and spent fuel capabilities are supported by the design of the UHS [ultimate heat sink]. A plant operator has the specified timeframe after the initiation of a beyond-design-basis external event (BDBEE) to provide replenishment of the UHS water level.

The LTR describes the “specified timeframe” as **{{**

}}. Therefore, the TR submittal acknowledges that operator actions will be required **{{**

}}. Therefore, the conditions described in the TR are not consistent with the conditions assumed to be in place at the site by the statements of consideration for 10 CFR 50.155 and the staff position in RG 1.226, Section 1.1.1.3 related to the establishment of the capability to receive and implement future, unplanned resources. The NRC staff concludes that the TR does not resolve this issue and, accordingly, an applicant or licensee that references this TR must satisfy Limitation and Condition no. 5.2 to resolve this issue. Limitation and Condition 5.2 requires an applicant to provide preplanned mitigating actions to ensure that long term coping requirements regarding UHS make up are satisfied or provide a satisfactory justification showing that such actions are not required.

3.3.2.2 Control Room Egress

Per section 4.12.1, “System Design,” in the TR, breathable air is credited to be available to the control room operators for 72 hours. No other provisions are established to sustain control room operators during this initial 72-hour period. Once this breathable air is depleted, the control room will become unhabitable, and the operators will be required to egress the control room. Debris may block egress of the control room due to the BDBEE. However, no provision is described for preplanning of debris removal. Therefore, the NRC staff concludes that an applicant or licensee that references this TR must satisfy Limitation and Condition 5.3 to resolve this issue. Limitation and Condition 5.3 requires that preplanned mitigating actions are established to ensure the sustainability of control room operators for 72 hours post-BDBEE, as described in the TR, in compliance with 10 CFR 50.155(b)(1)(i), and to then egress the control room when breathable air is depleted. Alternatively, a justification must be provided that supports no preplanning for these mitigating actions.

3.3.2.3 Determining Plant Conditions Post-BDBEE

Section 5.3.2, “Containment Process Variables,” of the TR states, in part:

> Per baseline coping capability of Section 3.1.2 [,"Baseline Coping Capability Criteria, Conditions, and Assumptions,"] the instrumentation associated with each process variable is assumed to survive the BDBEE and remain fully available for a duration beyond the time necessary for the associated mitigation function to be established and monitored.

Section 5.3.2 indicates that the available duration is limited to confirming that safety systems have actuated to their passive operating configuration during the initial coping phase. Once offsite support arrives, knowledge of current plant conditions will be necessary to determine appropriate mitigating actions. Therefore, the NRC staff concludes that an applicant or licensee that references this TR must satisfy Limitation and Condition no. 5.4 to resolve this issue. Limitation and Condition no. 5.4 requires that preplanned mitigating actions are established to ensure that site support personnel can ascertain plant conditions to determine necessary plant coping requirements once on-site instrumentation power systems are depleted. Alternatively, a justification must be provided that supports not preplanning for these mitigating actions.

3.3.2.4 SFP Level Monitoring

Section 3.2.4, "Monitoring," of the TR states the following, in part:

> **{{**
>
> **}}**

Depending on the nature of the BDBEE, **{{**

}}. Therefore, the NRC staff concludes that an applicant or licensee that references this TR must satisfy Limitation and Condition no. 5.5 to resolve this issue. Limitation and Condition no. 5.5 requires that an applicant or licensee referencing this TR must address (e.g., in plant procedures) how plant operators will ensure that any required debris removal will be accomplished to allow plant access to support replacement of SFP level monitoring instrumentation power supply equipment, as described in Section 9.0 of the TR, and that a preplanned source is established to provide the power supply equipment when required. Alternatively, a justification must be provided that supports no preplanning for these mitigating actions.

3.3.2.5 Conclusion

The NRC staff concludes that the approach outlined in the TR would be acceptable to ensure long-term coping capability after a beyond-design-basis event in conjunction with the implementation of the conditions detailed above, provided the applicant satisfies the limitations and conditions established herein. An applicant or licensee using a NuScale design will need to implement the limitations and conditions mentioned in the above sections when adopting this TR to demonstrate compliance with 10 CFR 50.155(b)(1).

3.4 Capability to respond to a loss of large areas (LOLA) due to explosions or fire, as required by 10 CFR 50.155(b)(2)

10 CFR 50.155(b)(2) requires licensees to develop and implement strategies and guidelines to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with LOLA of the plant due to explosions or fire. Strategies and guidelines must address in a three-phase approach:

Phase I – Enhanced firefighting capabilities
Phase II – Measures to mitigate damage to fuel in the SFP, and
Phase III – Measures to mitigate damage to fuel in the reactor vessel and to minimize radiological release.

NuScale stated that the TR follows guidance in NUREG-0800, section 19.4, "Strategies and Guidance to Address Loss of Large Areas of the Plant due to Explosions and Fires," (ML13316B202), which directs new plants to implement guidance in the February 5, 2005 Temporary Instruction 2515/168, "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," and Nuclear Energy Institute (NEI) 06-12, "B.5.b Phase 2 & 3 Submittal Guideline," in addressing beyond-design-basis events (e.g., LOLA).

3.4.1 Phase 1 - Enhanced firefighting capabilities

NuScale stated that NEI 06-12 guidance for firefighting response to a LOLA event includes operational aspects of responding to explosions or fire including prearranging for involvement of outside organizations, planning and preparation activities (e.g., pre-positioning equipment, personnel, and materials to be used for mitigating the event), and developing procedures and training for managing the event. NuScale also stated that it incorporates the following design features to cope with potential fires that could affect module or plant safety:

- Redundant safety systems to perform safety-related functions, such as reactor shutdown and core cooling
- Physical separation between redundant trains of safety-related equipment used to mitigate the consequences of a design-basis accident
- Passive design that minimizes the need for support systems and the potential effects of "hot shorts"
- Annunciation of fire indication in the main control room to facilitate personnel response
- No electrical power requirement for mitigating design-basis events as safety systems are fail-safe on loss of power

The NRC staff concludes that the approach outlined in the TR would be acceptable to ensure enhanced firefighting capabilities to respond to a LOLA due to explosions or fire, provided the applicant satisfies Limitation and Condition no. 5.6. An applicant or licensee using a NuScale design will need to implement this limitation and condition when adopting this TR to demonstrate compliance with 10 CFR 50.155(b)(2), 10 CFR 50.48, and GDC 3.

3.4.2 Phase 2 - Measures to mitigate damage to fuel in the SFP

NuScale stated that the SFP is below grade and the walls are designed to seismic Category I requirements and are completely contained within the seismic Category I Reactor Building (RXB). NuScale also stated that all pipe connections to and from the pool are at an elevation below the normal operating level but above the minimum pool level required for SFP radiation shielding and heat removal or are protected by a siphon break which prevents inadvertent lowering of the pool level below safety limits.

The NRC staff concludes that the approach in the TR, which outlines that the SFP is below grade and designed such that the SFP cannot be drained below safety limits, would be acceptable to ensure enhanced SFP capability to mitigate damage to fuel in the SFP in response to a LOLA due to explosions or fire, or demonstrate this enhanced capability is not required, provided the applicant satisfies Limitation and Condition no. 5.1. An applicant or licensee using a NuScale design will need to implement this limitation and condition when adopting this TR to demonstrate compliance with 10 CFR 50.155(b)(2).

3.4.3 Phase 3 - Measures to mitigate damage to fuel in the reactor vessel and to minimize radiological release

NEI 06-12 guidance for extensive damage mitigation was developed based on pressurized water reactor plant key safety functions, which includes reactor coolant system (RCS) inventory control, RCS heat removal, containment isolation, containment integrity, and release mitigation. In section 6.4.1 of the TR, “Assessment of Key Safety Functions,” NuScale described the key design features to achieve the above safety functions. The assessment for each safety function is summarized as follows:

RCS Inventory Control – NuScale stated that the purpose of this key safety function is to ensure that the core is covered with water. NuScale stated that containment system is utilized as the primary means for RCS inventory control. NuScale further stated that the design does not have RCPs, and therefore, there is no potential for loss of inventory through RCP seals due to lack of seal cooling. NuScale also stated that leakage rate through CIVs is small, and makeup is not required.

RCS Heat Removal – NuScale stated that the purpose of this key safety function is to remove the decay heat from the core and transfer it the UHS. NuScale stated that the primary means for heat removal during steady state, startup and hot shutdown operations is through the steam generators. NuScale further stated that the alternate means for RCS heat removal is the passive DHRS or the ECCS and that during DHRS or ECCS operations, no electrical AC power or external feedwater injection is required.

Containment Isolation - NuScale stated that the purpose of this key safety function is to ensure no leakage paths exist that would allow gaseous and particulate radiation to escape containment. NuScale further stated that CIVs and the CNV are utilized to accomplish this function, and that the CIVs are energized open so a loss of DC power to those valves will result in their repositioning to their safe or accident response position.

Containment Integrity - NuScale stated that 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. NuScale further stated that passive heat removal to the UHS controls temperature and pressure to ensure containment integrity.

Release Mitigation - NuScale stated that the purpose of this key safety function is to minimize radiological release assuming severe core damage occurs, and a radiological release is imminent or in progress. NuScale stated that the CNV is an American Society of Mechanical Engineer (ASME) Boiler and Pressure Vessel (B&PV) Code Section III Class I pressure vessel forming a barrier to prevent uncontrolled release of radiological materials and radiological contaminants. NuScale further stated that the reactor pressure vessel is located within the CNV, and the CNV is partially immersed in the UHS, and that the UHS is the primary means to perform the release mitigation function.

The NRC staff concludes that the approach outlined in the TR would be acceptable to ensure that key safety functions to mitigate potential fuel damage and radiological release are accomplished, provided that the applicant satisfies Limitation and Condition no. 5.1. An applicant or licensee using a NuScale design will need to implement this limitation and condition when adopting this TR to demonstrate compliance with 10 CFR 50.155(b)(2).

3.4.4 Conclusion

The NRC staff concludes that the approach outlined in the TR would be acceptable to ensure adequate LOLA coping capability **{{ }}** per NEI 06-12 guidance, provided the applicant satisfies the limitations and conditions detailed above. An applicant or licensee using a NuScale design will need to implement these limitations and conditions to demonstrate compliance with 10 CFR 50.155(b)(2).

3.5 Capacity, capability, and protection of equipment associated with mitigation of events described in the rule, as required by 10 CFR 50.155(c)

10 CFR 50.155(c) states the following:

> (c) Equipment. (1) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must have sufficient capacity and capability to perform the functions required by paragraph (b)(1) of this section.
>
> (2) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.

TR section 7.0, "Capacity, capability, and protection of equipment associated with mitigation of events described in the rule, as required by 10 CFR 50.155(c)," states the following:

> **{{**
>
> **}}**

The staff has determined that future applicants that reference this TR would need to show that their proposed strategy for NPM designs provides for extended coping using **{{ }}** that is reasonably protected from the effects of natural phenomena. As stated in Limitation and Condition no. 5.1, an applicant or licensee referencing the TR must meet the "Conditions of Use" listed in Section 1.3 of the TR, which details, in part, that (1), an applicant or licensee referencing the TR must provide a plant specific design that includes the **{{ }}** [and] (2) provide a plant specific thermal analysis demonstrating **{{ }}**. Therefore, the NRC staff concludes that the approach in the TR would be acceptable, provided the applicant satisfies Limitation and Condition no. 5.1. An applicant or licensee using a NuScale design will need to implement this limitation and condition when adopting this TR to demonstrate compliance with 10 CFR 50.155(c).

3.6 Training requirements as defined by 10 CFR 50.155(d)

10 CFR 50.155(d) states "Each licensee shall provide for the training of personnel that perform activities in accordance with the capabilities required by paragraphs (b)(1) and (2) of this section."

In section 8.0 "Training requirements as defined by 10 CFR 50.155(d)" of the TR, NuScale states the following:

> **{{ }}**

As stated in Conditions of Use no. 2 of TR section 1.3, which is adopted in Limitation and Condition no. 5.1, an applicant or licensee that references the topical report must provide a plant specific thermal analysis demonstrating **{{ }}** Limitation and Condition no. 5.7 states that an applicant or licensee referencing this TR must ensure that the training program for site staff that covers **{{ }}** as described in section 9.0 of the TR. Therefore, the NRC staff concludes that the approach outlined in the TR would be acceptable to ensure adequate training requirements, provided the applicant satisfies Limitation and Condition nos. 5.1 and 5.7. An applicant or licensee using the NuScale design will need to implement these limitations and conditions when adopting this TR to demonstrate compliance with 10 CFR 50.155(d).

3.7 SFP monitoring after final fuel removal from the reactor vessel, as required by 10 CFR 50.155(e)

10 CFR 50.155(e) states:

(e) Spent fuel pool monitoring. In order to support effective prioritization of event mitigation and recovery actions, each licensee shall provide reliable means to remotely monitor wide-range water level for each spent fuel pool at its site until 5 years have elapsed since all of the fuel within that spent fuel pool was last used in a reactor vessel for power generation. This provision does not apply to General Electric Mark III upper containment pools.

RG 1.227, Rev. 0, provides guidance for satisfying the requirements of 10 CFR 50.155(e). This RG endorses, with exceptions and clarifications, the methods and procedures promulgated by NEI in NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, 'To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,'" Revision 1 (NEI 12-02) dated August 2012 (Ref. 10) as a process the NRC staff considers acceptable for meeting certain requirements in 10 CFR 50.155.

The TR indicates that the level instrumentation relative to 10 CFR 50.155(e) (four instruments) are seismically mounted, environmentally qualified, and designed to meet the guidance of NEI 12-02. The level instruments are also provided **{{**

}}.

The staff evaluated the applicant's description of the level instrumentation provided in the TR and finds that it meets the design criteria recommended in NEI 12-02. The staff also evaluated the level instrumentation power requirements and determined that **{{ }}** would provide sufficient capacity to maintain the level indication function until offsite resource availability is reasonably assured.

NEI 12-02 section 4.1 "Training," indicates that:

> "Procedures will be developed using guidelines and vendor instructions to address the maintenance, operation and abnormal response issues associated with the new SFP instrumentation."

The TR section 8.0 "Training requirements as defined by 10 CFR 50.155(d)" states that:

> **{{**
>
> **}}**

As discussed above in section 3.6, the staff finds it would be acceptable to credit the training program for site staff that covers **{{**

}}, as stated in Section 9.0 of the TR and in accordance with Limitation and Condition no. 5.7.

The staff finds that level instrumentation designed in accordance with NEI 12-02 and provided with a dedicated backup battery supply, as described in the TR, as well as the capability to allow use of procured offsite equipment, would allow a future applicant to demonstrate it meets the design criteria discussed in RG 1.227. Therefore, an applicant or licensee using this TR in a NuScale design would be required to satisfy Limitation and Condition No. 5.7 to demonstrate compliance with the requirements of 10 CFR 50.155(e).

4.0 CONCLUSION

Based upon its review as discussed above, subject to the limitations and conditions as described in section 5.0 of this SE, the NRC staff concludes that an applicant or licensee could use TR-141299-P, Revision 1 to demonstrate the NuScale plant design's capability to mitigate BDBEs as defined by 10 CFR 50.155.

If an applicant for an operating license under 10 CFR Part 50, or an applicant for a combined license under 10 CFR Part 52, is not able to demonstrate compliance with an NRC regulation when the plant specific design is complete, the applicant would be required to justify an exemption from the applicable regulatory requirement. The NRC staff will evaluate the regulatory compliance of a plant specific design during future licensing reviews conducted in accordance with 10 CFR Part 50 or 10 CFR Part 52, as applicable. As discussed in the TR, the TR could be applied generically, therefore the final design with which this TR may be utilized is currently unknown. The NRC staff will make a final determination of the acceptability of an applicant's compliance with 10 CFR 50.155 during future licensing activities when the detailed design is complete as outlined in an operating license or combined license application that references this TR.

5.0 LIMITATIONS AND CONDITIONS

The staff's approval is limited to the application of this methodology to the NuScale reactor design, specified in TR Section 1.2, "Scope," with the following limitations and conditions:

5.1 An applicant or licensee referencing the TR must meet the "Conditions of Use" listed in section 1.3. To satisfy Condition of Use #1 an applicant or licensee must provide a plant specific design that includes **{{ }}** with the described system functions listed in TR Table 4-1 and the system design features and equipment classifications as described in TR section 4.0 – 4.15 for each system.

5.2 An applicant or licensee referencing the TR must address (e.g., in plant procedures) how plant operators will ensure, during the initial coping phase, that the following actions can be achieved to provide inventory makeup to the UHS at the start of the final long-term coping phase: (1) a source of water can be identified and will be available in sufficient quantity (2) the necessary motive equipment such as pumps and generators, and the required electrical power/fuel, can be obtained, staged, and implemented and (3) any required debris removal will be accomplished to support placement of equipment and access to site connections. Alternatively, an applicant or licensee can justify why the plant-specific application requires no preplanning before a BDBEE to address these mitigating actions.

5.3 An applicant or licensee referencing the TR must address (e.g., in plant procedures) how control room operators will be sustained for 72 hours in the control room and subsequently exit the control room at 72 hours post-event once breathable air is depleted if debris blockage prevents egress from the control room. Alternatively, an applicant or licensee can justify why the plant-specific application requires no preplanning before a BDBEE to address these mitigating actions.

5.4 An applicant or licensee referencing the TR must address (e.g., in plant procedures) how site support personnel will ascertain plant conditions in order to determine necessary coping requirements during the initial phase, once on-site power systems are depleted, and at the start of the final long-term coping phase. Alternatively, an applicant or licensee can justify why the plant-specific application requires no preplanning before a BDBEE to address these mitigating actions.

5.5 An applicant or licensee referencing this TR must address (e.g., in plant procedures) how plant operators will ensure that any required debris removal will be accomplished to allow plant access to support replacement of SFP level monitoring instrumentation power supply equipment, as described in section 9.0 of the TR, and that a preplanned source is established to provide the power supply equipment when required. Alternatively, an applicant or licensee can justify why the plant specific application requires no preplanning before a BDBEE to address these mitigating actions.

5.6 An applicant or licensee referencing the TR must provide a Fire Protection Program in accordance with 10 CFR 50.48.

5.7 An applicant or licensee referencing this TR must ensure that the training program for site staff that covers installed plant equipment includes required activities related to replacement of the SFP level monitoring instrumentation power supply equipment as described in section 9.0 of the TR.

6.0 REFERENCES

1. NuScale Power, LLC, TR-141299, "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," Revision 0, September 11, 2023, (ML23254A360 (public) and ML23254A361 (non-public)).

2. Memorandum from Tesfaye, G., NRC, to Jardaneh, M., NRC “Audit Plan for the staff review of the NuScale generic licensing topical reports,” December 20, 2023 (ML23349A078).

3. NuScale Power, LLC, TR-141299, “NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155,” Revision 1, June 26, 2024, (ML24178A398 (public) and ML24178A399 (non-public)).

4. Regulatory Guide (RG) 1.226, “Flexible Mitigation Strategies for Beyond- Design-Basis Events,” Rev. 0 (ML19058A012)

5. NEI 12-06, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” Rev. 4 (ML16354B421)

6. NRC, SECY-19-0066, “Staff Review of NuScale Power’s Mitigation Strategy for Beyond-Design-Basis External Events,” June 26, 2019 (ML19148A443)

7. NuScale Power, LLC, Response to NuScale Topical Report Audit Question, A-MBDBE.LTR-8 (ML24178A401 (public) and ML24178A403 (non-public)).

8. RG 1.227, “Wide-Range Spent Fuel Pool Level Instrumentation,” issued June 2019 (ML19058A013)

9. *Federal Register* Volume 84, No. 154, “Mitigation of Beyond-Design-Basis Events,” August 9, 2019, pages 39684 - 39709

10. Nuclear Energy Institute (NEI) in document NEI 12-02, “Industry Guidance for Compliance with NRC Order EA-12-051, ‘To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,’” Revision 1, dated August 2012

Section B

Licensing Topical Report

NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155

May 2025

Revision 1

Docket: 99902078

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Department of Energy Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0008928.

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Abstract

This report describes design capabilities to mitigate beyond-design-basis events as defined by 10 CFR 50.155. This report is applicable to a NuScale design that includes the capabilities and features discussed herein.

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Executive Summary

Following the Fukushima Daichi accident in March 2011, the U.S. Nuclear Regulatory Commission (NRC) established additional requirements for nuclear power reactor licensees for the mitigation of beyond-design-basis events (MBDBE). The requirements are codified in 10 CFR 50.155, "Mitigation of beyond-design-basis events" (Reference 12.1).

NuScale seeks approval for the design capabilities a licensee will rely on for their mitigation strategies and spent fuel pool level monitoring. This report addresses the requirements of the Mitigation of Beyond-Design-Basis Events rule, 10 CFR 50.155 (Reference 12.1). Specifically, addressed are:

- response to a beyond-design-basis external event (BDBEE), assuming a loss of all alternating current (AC) power event concurrent with a loss of normal access to the normal heat sink (LNHS), as required by 10 CFR 50.155(b)(1),
- capability to respond to a loss of a large area (LOLA) due to explosions or fire, as required by 10 CFR 50.155(b)(2),
- protection of equipment associated with mitigation of a BDBEE, as required by 10 CFR 50.155(c),
- training requirements as defined by 10 CFR 50.155(d),
- spent fuel pool monitoring as required by 10 CFR 50.155(e).

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}}2(a),(c)

<u>Loss of All Alternating Current (AC) Power Event Concurrent with a Loss of Normal Access to the Normal Heat Sink</u>

10 CFR 50.155(b)(1), extensive damage mitigation guidelines, requires licensees to develop and implement strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool (SFP) cooling capabilities under the circumstances associated with a loss of all alternating current power event concurrent with a loss of normal access to the normal heat sink. Strategies and guidelines developed by a licensee must address each of the following areas:

i) Maintaining or restoring core cooling, containment, and spent fuel pool capabilities; and

ii) The acquisition and use of offsite assistance and resources to support maintaining core cooling, containment, and spent fuel pool capabilities indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies.

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The containment function is provided and maintained by plant safety-related systems. Containment function is maintained while the containment vessel (CNV) temperature and pressure are maintained within limits. Heat is passively transferred from the CNV to the ultimate heat sink (UHS). {{

}}2(a),(c)

The indefinite core cooling, containment, and spent fuel capabilities are supported by the design of the UHS. A plant operator has the specified timeframe after the initiation of a BDBEE to provide replenishment of the UHS water level.

Loss of Large Area

10 CFR 50.155(b)(2), extensive damage mitigation guidelines, requires licensees to develop and implement strategies and guidelines to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with LOLA of the plant due to explosions or fire. Strategies and guidelines developed by a licensee must address each of the following areas:

i) firefighting

ii) operations to mitigate fuel damage

iii) actions to minimize radiological release

This report specifies the response to a LOLA event. The guidance in NUREG-0800, Standard Review Plan 19.4 (Reference 12.2), directs new plants to implement guidance in the February 25, 2005, TI 2515/168, and NEI 06-12, Revision 3. The guidance in Nuclear Energy Institute (NEI) 06-12, “B.5.b Phase 2 & 3 Submittal Guideline” (Reference 12.3) is used to determine the design response to a LOLA event. The NEI 06-12 guidance describes a three-phase approach for performing an evaluation that meets 10 CFR 50.155(b)(2). The phases and associated guidance are:

- Phase 1 - Enhanced firefighting capabilities

- Phase 2 - Measures to mitigate damage to fuel in the SFP
- Phase 3 - Measures to mitigate damage to fuel in the reactor vessel and to minimize radiological release

All three phases are adequately addressed by the specified plant equipment and capabilities.

Protection of Equipment Associated with Mitigation of Events

10 CFR 50.155(c) requires the equipment relied on for the mitigation strategies and guidelines to have sufficient capacity and capability to perform the key safety functions and to be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility. {{

}}2(a),(c)

The RXB is a safety-related Seismic Category I building that houses, supports, and protects the modules, the UHS, and all the necessary fail-safe passive features to ensure safe reactor shutdown. The RXB is designed to withstand design basis natural phenomena including earthquakes, floods, high winds (including associated missiles), and extreme temperatures.

Training Requirements

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Spent Fuel Pool Monitoring After Final Fuel Removal from the Reactor Vessel

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}}2(a),(c) satisfies the requirements of 10 CFR 50.155(e).

1.0 Introduction

1.1 Purpose

This report specifies the plant design capability to mitigate beyond-design-basis events as defined by 10 CFR 50.155, specifically, (i) the plant response to the loss of all alternating current power concurrent with loss of normal access to the normal heat sink and (ii) the design capability to mitigate the loss of large plant areas due to explosions or fire.

1.2 Scope

This report describes the plant design capability to mitigate beyond-design-basis events as defined by 10 CFR 50.155 (Reference 12.1) as applicable to the design.

Specifically addressed in this report are:

- design capability to respond to a BDBEE, assuming a loss of all alternating current (AC) power event concurrent with a LNHS as required by 10 CFR 50.155(b)(1),
- capability to respond to a LOLA due to explosions or fire as required by 10 CFR 50.155(b)(2),
- protection of equipment associated with mitigation of a BDBEE as required by 10 CFR 50.155(c),
- training requirements associated with mitigation capabilities as defined by 10 CFR 50.155(d),
- spent fuel pool monitoring as required by 10 CFR 50.155(e).

Specific site characteristics that are unique in some manner must be addressed by the licensee and are not within the scope of this report.

This report is applicable to NuScale small modular reactor designs that include the following design characteristics:

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1.3 Conditions of Use

To adopt the described methodology for the response to BDBEEs, an adopter of the topical report must provide:

1. Plant specific design that {{ }}2(a)(c) described within this report.
2. A plant specific thermal analysis demonstrating {{ }}2(a)(c) This analysis addresses site specific conditions, including configuration of the plant with respect to the selected number of modules and spent fuel pool capacity, for all modes of operation (normal and refueling).
3. A maintenance rule program in accordance with 10 CFR 50.65.
4. An emergency plan in accordance with 10 CFR 50.160 or 50.47(b) and Appendix E describing communications and coordination with local, state, federal, and tribal agencies.

1.4 Acronyms and Definitions

Table 1-1 Acronyms

Term	Definition
AAPS	auxiliary AC power source
AC	alternating current
AHU	air handling unit
ASME	American Society of Mechanical Engineers
BDBE	beyond-design-basis event
BDBEE	beyond-design-basis external event
BDG	backup diesel generator
BPSS	backup power supply system
BWR	boiling water reactor
CES	containment evacuation system
CFDS	containment flooding and drain system
CFR	Code of Federal Regulations
CIV	containment isolation valve
CNTS	containment system
CNV	containment vessel
CRB	Control Building
CRDM	control rod drive mechanism
CRE	control room envelope
CRHS	control room habitability system
CVCS	chemical and volume control system
DC	direct current
DHRS	decay heat removal system
ECCS	emergency core cooling system

Table 1-1 Acronyms (Continued)

Term	Definition
EDAS	augmented DC power system
EDAS-C	EDAS - common
EDAS-MS	EDAS - module specific
EDMG	extensive damage mitigation guidelines
EDNS	normal DC power system
EHVS	13.8 kV and switchyard system
ELVS	low voltage AC electrical distribution system
EMVS	medium voltage AC electrical distribution system
ERO	Emergency Response Organization
ESF	engineered safety feature
ESFAS	engineered safety features actuation system
FPS	fire protection system
FWIV	feedwater isolation valve
GTG	NuScale Generic Technical Guideline
HVAC	heating, ventilation, and air conditioning
LOLA	loss of large areas
LNHS[a]	loss of all AC power concurrent with loss of normal access to the normal heat sink
MBDBE	mitigation of beyond-design-basis events
MCR	main control room
MPS	module protection system
MSIV	main steam isolation valve
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NMS	neutron monitoring system
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
OHLHS	overhead heavy load handling system
PAM	post-accident monitoring
PPS	plant protection system
PSCIV	primary system containment isolation valve
PWR	pressurized water reactor
RBC	Reactor Building crane
RBVS	Reactor Building HVAC system
RCCWS	reactor component cooling water system
RCS	reactor coolant system
RFP	refueling pool
RP	reactor pool
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RTB	reactor trip breaker
RTS	reactor trip system
RVV	reactor vent valve
RXB	Reactor Building
SBO	station blackout
SDIS	safety display and indication system

Table 1-1 Acronyms (Continued)

Term	Definition
SFP	spent fuel pool
SG	steam generator
SGI	Safeguards Information
SSCIV	secondary system containment isolation valve
SSC	structures, systems, and components
SSE	safe shutdown earthquake
TAF	top of active fuel
UHS	ultimate heat sink

a. The MBDBE rule defines a loss of normal access to the ultimate heat sink for a passive design, such as the plant design, as loss of normal access to the normal heat sink.

Table 1-2 Definitions

Term	Definition
Alternate AC source	An AC power source that is available to and located at or nearby a nuclear power plant and meets the following requirements- 1. Is connectable to, but not normally connected to, the offsite or onsite emergency AC power systems; 2. Has minimum potential for common mode failure with offsite power or the onsite emergency AC power sources; 3. Is available in a timely manner after the onset of station blackout (SBO); and Has sufficient capacity and reliability for operation of systems required for coping with SBO and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident).
Emergency core cooling system (ECCS) hold mode	An operating mode of augmented DC power system (EDAS) - module specific (EDAS-MS). If an ECCS actuation is not required during a loss of all AC power, EDAS will provide the electrical power to maintain the valves in the shut position.
Loss of all AC power event	A loss of all AC power concurrent with a loss of normal access to the normal heat sink.
LOLA	Refers to the unavailability of a large area of a plant due to explosion or fire.
Loss of normal access to the normal heat sink (LNHS)	Loss of ability to provide a forced flow of water to key plant systems (i.e., the pumps are unavailable and not restorable as part of the coping strategy). Normal access to the normal heat sink is lost, but the water inventory in a second heat sink, the UHS, remains available and a robust qualified makeup line to the UHS remains intact.
Module	The module is a self-contained nuclear steam supply system composed of a reactor core, a pressurizer, and two SGs integrated within the reactor pressure vessel (RPV) and housed in a compact steel containment vessel (CNV).
Post-accident monitoring (PAM)-only mode	Mode of module protection system (MPS) operation following the expiration of ECCS hold mode timers where only electrical loads required for the support of PAM variables are maintained energized.
Passive safety system	A system that does not require AC power to operate, but relies instead upon natural forces, such as gravity and natural circulation, or on sources of stored energy, such as pressurized tanks or batteries.

Table 1-2 Definitions (Continued)

Term	Definition
Robust (designs)	The design of structures, systems, and components (SSC) either meets the current plant design basis for the applicable external hazards, or the current NRC design guidance for the applicable hazard, or is shown by analysis or test to meet or exceed the current design basis.
Safe shutdown	Safe shutdown occurs when the following conditions are met- a. Reactivity is controlled such that the core is maintained subcritical. b. Core cooling is provided to assure decay heat removal.
Safe shutdown earthquake (SSE)	The design basis earthquake for commercial nuclear power plants. It is that earthquake that produces the maximum vibratory ground motion for which certain SSC, including safety-related SSC, are designed to remain functional.
Seismic Category I SSC	SSC that are designed and built to withstand the effects of the SSE and remain functional.
Seismic Category II SSC	SSC that perform no nuclear safety function and whose continued function is not required, but whose structural failure or interaction could degrade the functioning of a Seismic Category I SSC to an unacceptable safety level or could result in incapacitating injury to occupants of the control room. Seismic Category II SSC are designed so that the SSE would not cause loss of function of Seismic Category I items.
Seismic Category III SSC	All SSC that are not included in Seismic Category I or Seismic Category II. Members and structural subsystems whose failure would not impair the capability for safe shutdown or continued operation.
Ultimate Heat Sink	The ultimate heat sink is a set of safety-related pools of borated water that comprise the combined water volume of the reactor pool, refueling pool, and spent fuel pool. The UHS pools are located below grade in the Reactor Building (RXB).

2.0 Background

Following the Fukushima Daiichi accident in March 2011, the U.S. Nuclear Regulatory Commission established requirements for nuclear power reactor licensees to mitigate beyond-design-basis events. The NRC issued the regulatory requirements under several Orders that were based on task force recommendations.

The mitigation of beyond design basis events (MBDBE) rule codified in 10 CFR 50.155 (Reference 12.1) makes those orders generically applicable to holders of operating licenses and combined licenses.

2.1 Regulatory Requirements

This Topical Report documents the review of 10 CFR 50.155 requirements and their effect on licensees that reference a plant application with the required capabilities and features as specified in Section 1.2.

3.0 Plant Baseline Coping Criteria for Loss of all AC Power

This section describes coping strategies and design capabilities of a plant in a loss of all AC power event. Coping capabilities of a design is performed against the industry standard (Reference 12.4).

3.1 Coping Strategies

3.1.1 Objective

The objective of the mitigation strategies for beyond-design basis external events required by the MBDBE rule is to establish a coping capability to prevent damage to the fuel in an operating module and the spent fuel pool (SFP) and to maintain the containment function by using plant equipment and mitigation equipment during a loss of all AC power event. The mitigation strategies must enable key safety functions to be maintained or restored indefinitely, or until sufficient site functional capabilities can be maintained without the need for the mitigation strategies described in 10 CFR 50.155. Emergency planning requirements established in 10 CFR 50.160 or 50.47(b) and Appendix E ensure local, state, federal, and tribal agencies have identified responsibilities in the event of site emergencies. This is sufficient to support response to BDBEEs such that the consequences of an event will be successfully mitigated. Requirements for the conditions and the coping capabilities of the plant are described below, and are consistent with the industry standard scenario (Reference 12.4).

3.1.2 Baseline Coping Capability Criteria, Conditions, and Assumptions

1. Plant equipment protected from the effects of design basis natural phenomena is assumed to be fully available (Section 4.0).
2. Plant equipment not demonstrated as protected from the effects of design basis natural phenomena is assumed to be unavailable.
3. Procedures and equipment relied upon ensure that satisfactory performance of necessary fuel cooling[1] and containment functions are maintained.
4. The fuel in the reactor is required to remain covered at all times.
5. The fuel in the SFP is required to remain covered at all times.

3.1.3 Boundary Conditions

1. Beyond-design-basis external event impacts all modules at site.
2. All modules on-site initially operating at power[2].
3. Each module is successfully shut down when required (i.e., all rods inserted, no anticipated transient without scram).
4. On-site staff is at site administrative minimum shift staffing levels.

1.“Fuel cooling” refers to both fuel in the reactor and fuel in the SFP.
2.For conservatism, additional cases involving a module in other power stages are considered (see Section 5.5 and Section 5.6).

5. No independent, concurrent events, e.g., no active security threat.
6. Personnel on-site are available to support site response.
7. Spent fuel in dry storage is outside the scope of the event.

3.1.4 Initial Plant Conditions

The initial plant conditions are assumed to be:

1. Prior to the event, each module has been operating at 100 percent rated thermal power for at least 100 days.
2. At the time of the postulated event, the reactor and supporting systems are within normal operating ranges for pressure, temperature, and water level for the appropriate plant condition.
3. All plant equipment is either normally operating or available from the standby state as described in the plant design and licensing basis.
4. The minimum conditions for plant equipment operability or functionality do not need to be assumed in establishing the capability of that equipment to support mitigation strategies, provided there is an adequate basis for the assumed value.

3.1.5 Initial Event Conditions and Assumptions

The following conditions apply when determining the baseline coping capability of the plant (Reference 12.4):

1. No specific initiating event is considered. The initial condition is assumed to be a loss of offsite power for all modules 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.
2. Station batteries and associated direct current (DC) buses remain available for the designed operating time of the station batteries.
3. Cooling and makeup water inventories contained in systems or structures with designs that are robust for the beyond-design-basis external event are available.
4. Normal method of transferring heat to the normal heat sink is lost, but the water inventory in the ultimate heat sink (UHS) remains available.
5. Seismic Category I plant equipment contained in Seismic Category I structures is available.
6. Installed electrical distribution system, including inverters and battery chargers, remain available provided they are Seismic Category I.
7. No additional events or failures are assumed to occur immediately prior to or during the event, including security events.

3.1.6 Reactor Transient Assumptions

1. Following the loss of all AC power, all reactors automatically trip and all rods are inserted.
2. All safety-related systems operate as designed.
3. No independent failures other than those causing the simultaneous loss of all AC power and LNHS are assumed to occur in the course of the transient.

3.1.7 UHS Conditions

1. All Seismic Category I boundaries of the UHS are intact, including the liner and dry dock gate.[3]
2. Although sloshing may occur during a seismic event, the initial loss of UHS inventory does not preclude access to the refueling deck.
3. Heat load from spent fuel stored within the UHS assumes the maximum design basis heat load for the site.

3.2 Plant Design Capabilities

The first 72 hours of a loss of all AC power event is identical to a SBO. The SBO does not pose a significant challenge to the advanced passive design because it does not rely on AC power for performing safety functions. A safe and stable shutdown is automatically achieved and maintained for 72 hours without operator actions.

Following a loss of all AC power event, 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 and spent fuel into a safe, stable, shutdown state with passive cooling. {{

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3.2.1 Core Cooling

During a loss of all AC power event reactor coolant system inventory is preserved by containment isolation {{ }}2(a),(c)

The decay heat removal system (DHRS) passively removes decay heat for up to the first 24 hours following a loss of all AC power event. By 24 hours, the emergency core cooling system (ECCS) valves automatically open.

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3.2.2 Containment

Containment isolation valves (CIVs) and the CNV provide passive containment isolation function. Without operator action or electrical power, the safety-related CIVs close to isolate the CNV.

Heat removal to the UHS passively controls temperature and pressure to ensure containment integrity. Peak pressure and temperature conditions for the CNV are designed to occur early in the event when the ECCS valves open and prevent a challenge to containment integrity.

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3.2.3 Spent Fuel Pool Cooling

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 event until UHS level lowers to the weir wall.

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3.2.4 Monitoring

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Although not necessary because of the fail-safe and passive design, PAM variable indications are maintained in the main control room for at least 72 hours to provide additional assurance that systems respond as designed.

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4.0 Plant Systems and Responses to a Loss of All Alternating Current Power Event

To develop a mitigation strategy, determination of the baseline coping capability of the plant is made by evaluating the status of the three key safety functions, core cooling, containment, and SFP cooling, during the integrated plant response to a loss of all AC power event. To understand the integrated plant response, the systems that contribute to coping with the loss of all AC power event conditions are evaluated to determine their responses and availability during a loss of all AC power event. This section describes functionality of those systems and specifies their qualifications to determine availability.

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4.1 Reactor Building

The Reactor Building (RXB) is a safety-related Seismic Category I building that houses, supports, and protects the modules and the UHS, in addition to other safety-related components and systems. The RXB withstands design basis natural phenomena including earthquakes, floods, high winds (including associated missiles), and extreme temperatures.

4.1.1 Dry Dock

The RXB contains a dry dock adjacent to the UHS. A dry dock gate separates or allows passage between the dry dock and the refueling pool. When the dry dock gate is closed, the dry dock may contain no water or may contain a water level that is below the water level of the UHS. {{

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4.2 Control Building

The Control Building (CRB) is a reinforced concrete structure comprised of a Seismic Category I portion and a Seismic Category II portion. The main control room envelope and MPS components are located in the Seismic Category I portion while the Technical Support Center is located in the Seismic Category II portion. The CRB is designed for design basis environmental loads based on the designation of the building area (i.e., Seismic Category I or Seismic Category II). Environmental loads for Seismic Category I areas include earthquake, normal and tornado wind, tornado missile, flooding, rain, snow and earth pressure.

4.3 Augmented Direct Current Power System

4.3.1 System Design

The EDAS is the source of DC power to plant loads including the MPS, the plant protection system (PPS), and the safety display and indication system (SDIS).

The EDAS comprises two DC subsystems that provide a continuous, failure-tolerant source of DC power to assigned plant loads during normal plant operation and for a specified minimum duty cycle following a loss of AC power. The EDAS-common (EDAS-C) plant subsystem serves plant common loads that have functions that are not specific to a single module. These functions include main control room (MCR) emergency lighting and PAM variable indications displayed in the MCR. The EDAS-MS plant subsystem consists of separate and independent DC electrical power supply systems, one for each module.

The EDAS-MS consists of four power channels and EDAS-C consists of two power divisions. The EDAS-MS and EDAS-C are capable of providing uninterrupted power to their loads. The EDAS-MS channels A and D have a specified minimum battery duty cycle of 24 hours, and EDAS-MS channels B and C have a specified minimum battery duty cycle of 72 hours. The EDAS-C power divisions have a specified minimum battery duty cycle of 72 hours.

The 24-hour battery duty cycle of EDAS-MS channels A and D is specified to provide power to ECCS Hold Mode loads for 24 hours following a postulated loss of AC power. A 24-hour timer is initiated when AC power is lost. ECCS Hold Mode operation maintains ECCS reactor recirculation and reactor vent valves (RVVs) in a closed position, unless a valid ECCS actuation signal is received. The 72-hour battery duty cycle for EDAS-MS channels B and C and EDAS-C provides a minimum of 72 hours of DC electrical power for PAM Only Mode loads. PAM Only Mode on EDAS-MS channels B and C are initiated after the 24-hour timers time out.

4.3.2 Equipment Qualification

The EDAS-MS and EDAS-C structures, systems, and components that provide backup power meet Seismic Category I standards and are located within Seismic Category I areas of the RXB and CRB.

With the exception of cabling exiting the rooms, all major EDAS-MS components are housed in equipment rooms located in the RXB. Each EDAS-MS equipment room is partitioned into battery partition and a switchgear partition, separated by a barrier with a three-hour fire rating. Similarly, all major EDAS-C components are housed in an equipment room located in the CRB. The EDAS-C divisional equipment room is partitioned into a switchgear partition and a battery partition, separated by a barrier with a three-hour fire rating. Conditions within these rooms are limited to mild environments.

4.3.3 System Response to a Loss of All Alternating Current Power

Per initial event assumptions of Section 3.1.5, the EDAS batteries and the associated distribution system are assumed to survive the BDBEE and remain fully available during the loss of all AC power event.

The loss of the low voltage AC electrical distribution system (ELVS) at the initiation of the event results in the transfer of the module specific and common system loads to the applicable EDAS batteries. The battery chargers will cease providing output current to the DC bus, battery and connected plant loads. The EDAS battery or batteries will then immediately begin to discharge and will continue to supply connected loads. {{

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4.4 Module Protection System

4.4.1 System Design

The MPS is responsible for monitoring both nonsafety and safety-related process variables at the reactor module level and executing safety-related functions in response to out-of-normal conditions. The MPS cabinets, housing a majority of the MPS equipment, will be housed in rooms located in the reactor building designed to meet equipment separation requirements. There is one MPS per module. The MPS does not share structures, systems, and components (SSC) between modules.

The MPS also monitors the feedback valve positions of certain safety-related and nonsafety-related valves. The reactor trip system (RTS) and engineered safety features actuation system (ESFAS) are subsystems of each module’s MPS.

The MPS provides information regarding process variables to the control room operators through the module control system and the SDIS. Each module’s safety-related reactor trip circuitry is part of its MPS, powered from EDAS-MS. Included in each MPS are the safety-related circuit breakers providing electrical power to the control rod drive system (CRDS). These circuit breakers are referred to as the reactor trip breakers (RTBs).

During normal power operations, the RTS provides power to the RTBs to maintain the breakers closed. This functionality allows the ELVS to supply power to the control rod drive mechanism (CRDM) coils that, in turn, allows the control rods to remain partially or fully withdrawn from the reactor core. When a reactor trip is generated, the RTS removes power from the RTBs causing the breakers to open and interrupt ELVS power to the CRDM coils. With power removed from the CRDM coils, the control rods unlatch and fully insert into the core by gravity.

In addition to the requirements for a reactor trip for anticipated abnormal transients, the MPS provides instrumentation and controls to sense accident situations and initiate operation of necessary engineered safety features (ESFs). The ESFs include the ECCS, DHRS, secondary system isolation, containment system isolation (CSI) signal, demineralized water system isolation, chemical and volume control system isolation, pressurizer line isolation, pressurizer heater trip, and low temperature overpressure protection. Initiation of these ESFs is the function of the MPS subsystem referred to as the ESFAS.

During normal power operation, the CIVs are open and maintained in this position by energizing their valve actuator solenoids. The power used to energize these solenoids is provided by EDAS-MS and is routed through the MPS. When a containment isolation signal is generated, the ESFAS de-energizes the valve actuator solenoids causing the CIVs to close.

The DHRS is normally maintained in a standby configuration with operation prevented by closed DHRS actuation valves. These valves are maintained closed by energizing their valve actuator solenoids. Just as with the CIVs, the power used to energize these solenoids is provided by EDAS-MS and is routed through the MPS. When a DHRS actuation signal is generated, the ESFAS de-energizes the valve actuator solenoids and allows the DHRS actuation valves to open. The DHRS actuation signal also de-energizes the valve actuator solenoids for CIVs in the feedwater and main steam systems. The resultant valve positions, all occurring due to the removal of power from solenoids by the ESFAS, place the DHRS in operation.

Similar to the DHRS, the ECCS is normally maintained in standby with operation prevented by maintaining solenoids energized. This solenoid power is provided by EDAS-MS through the MPS and removal of this power from the solenoids upon an ECCS actuation signal allows the ECCS valves to open.

Execution of these safety-related functions through the removal of electric power by the MPS subsystems makes the MPS a fail-safe system. If the power needed for MPS to function is unavailable, the safety-related functions automatically occur. Because of this fail-safe manner of operation, the MPS does not require safety-related power and is powered by the nonsafety-related EDAS.

4.4.2 Equipment Qualification

The MPS is classified as a Seismic Category I system. The MPS equipment and cabling are housed in the Seismic Category I portions of the RXB and CRB. The RXB and CRB arrangement and design enable systems and components required for safe plant operation and shutdown to withstand or to be protected from the effects of sabotage, environmental conditions, natural phenomena, postulated design basis accidents, and design basis threats.

4.4.3 System Response to a Loss of All Alternating Current Power

The MPS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for mitigation strategy implementation are required.

For this event, the EDAS and the MPS remain energized. The ELVS de-energizes at event initiation and remains deenergized for the duration. The MPS monitors the voltage of the ELVS buses that provide power to the Channel B and Channel C EDAS-MS battery chargers as a method of detecting a loss of all AC power. Following a loss of AC power, the MPS automatically generates a reactor trip, DHRS actuation, pressurizer heater trip, demineralized water supply isolation, chemical and volume control system isolation, containment isolation, and starts the ECCS hold mode of operation for EDAS-MS. The effects of the 24-hour capacity of the EDAS-MS batteries are discussed in Section 4.3.1 and Section 4.10.3.

4.5 Plant Protection System

4.5.1 System Design

The PPS monitors and controls systems that are common to all reactor modules and are not specific to an individual module. The variables monitored and equipment actuated by the PPS have an augmented level of quality.

The PPS monitors process variables, including the reactor pool (RP), refueling pool (RFP), and SFP level. The PPS provides information regarding these process variables to the control room operators through the SDIS, the plant control system, or both.

The PPS consists of two divisions of equipment, each powered by the corresponding division of EDAS-C. During normal operation, the PPS provides loop power to various instruments and actuation solenoids in the control room habitability system (CRHS). The power provided to the CRHS actuation solenoids maintains the associated isolation valves closed to maintain the CRHS in a standby lineup.

Plant process variables that would require actuation of the CRHS are monitored by the PPS, including the loss of ELVS power to the EDAS-C battery chargers. When conditions requiring actuation occur, the PPS removes power from the actuation solenoids to initiate the CRHS. This de-energize-to-actuate design makes the PPS and the equipment it controls fail-safe.

4.5.2 Equipment Qualification

The PPS is classified as Seismic Category I and is housed in the Seismic Category I portion of the CRB.

4.5.3 System Response to a Loss of All Alternating Current Power

The PPS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for mitigation strategy implementation are required.

For this event, EDAS-C and the PPS remain energized. The ELVS de-energizes at event initiation and remains de-energized for the duration. As described above, the PPS monitors the voltage of the ELVS buses that provide power to the EDAS-C battery chargers as a method of detecting a loss of all AC power. With the loss of voltage on the ELVS buses, the PPS isolates the control room envelope (CRE) and initiates the CRHS.

4.6 Safety Display and Indication System

4.6.1 System Design

The SDIS is a nonsafety-related system that provides accident monitoring functions. Although the SDIS is not necessary due to the fail-safe and passive design, the post accident monitoring functions provide additional assurance that systems have responded as designed. The primary purpose of the SDIS is to display accurate, complete, and timely information provided by the MPS and the PPS.

The information displayed is provided to the SDIS from communications containing information from each separation group and each division of the MPS and PPS. This information contains the data that aid the operators in ensuring that the modules are in a safe condition following an event.

Electrical power to the SDIS is provided from two separate and independent divisions of EDAS-C. In the MCR, the SDIS provides two divisions of monitors for each module and the PPS, with both divisions of MPS and PPS data displayed on each division of the monitors. A partial list of the variables include:

- neutron flux
- core exit temperature
- wide range RCS pressure
- RPV water level
- containment water level
- wide-range containment pressure
- CIV positions
- ECCS valve positions
- DHRS valve positions
- RTB status
- SFP water level

4.6.2 Equipment Qualification

The SDIS is classified as nonsafety-related with augmented quality and design requirements. The SDIS is qualified to Seismic Category I requirements and is housed in the Seismic Category I portions of the CRB.

4.6.3 System Response to a Loss of All Alternating Current Power

The SDIS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for mitigation strategy implementation are required.

EDAS-C and the SDIS remain energized for a minimum of 72 hours following a loss of AC power. The SDIS continues to display the process variables that aid the control room operators in verifying the system configuration and the success of the strategy for maintaining core cooling, containment, and spent fuel cooling.

4.7 Containment System

4.7.1 System Design

The containment system (CNTS) is part of the module and is the containment and/or structural support for the RPV, RCS, ECCS, DHRS, CRDS, and associated components. The CNTS components include, the CNV, the CIVs, and CNTS instruments.

The CNV is a American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III Class I pressure vessel forming a barrier to prevent uncontrolled release of radioactivity and radiological contaminants. The RCS, CRDS, select primary system piping and valves, and the ECCS main valves are contained in the CNV. During normal operation, the CNV is partially immersed in the RP portion of the UHS, allowing the CNTS design to provide the function of containment heat removal. The CNV is safety-related.

The CIVs can be subdivided into two categories: the primary system containment isolation valves (PSCIVs) and secondary system containment isolation valves (SSCIVs).

The PSCIVs provide isolable connections for CES, CFDS, and chemical and volume control system (CVCS) process penetrations. The PSCIVs are a single assembly design consisting of two valves (fully independent discs, seats and actuators) within a single valve assembly. The PSCIVs are welded directly to the containment isolation test fixture valves.

One SSCIV is provided per line for the main steam lines, and feedwater lines that penetrate a CNV boundary but are neither part of the reactor coolant pressure boundary, nor connected directly to the containment atmosphere. The SSCIV and

PSCIV actuators are similar in design and their manner of operation is covered by the description of the CIV operation.

Each CIV is equipped with a stored energy device that applies a constant force to close the valve. For a CIV to be opened and remain open, its actuator solenoids must remain energized. With the solenoids energized, high-pressure hydraulic fluid overcomes the stored energy to open the valve. The power for these solenoids is provided by EDAS through the MPS.

When the ESFAS generates a containment isolation signal, power is removed from the CIV actuator solenoids. This de-energization aligns the high-pressure hydraulic fluid holding the CIVs open to a vent path and the stored energy forces the CIVs closed.

During normal operations, the CIVs associated with the containment evacuation system, reactor component cooling water system (RCCWS), main steam system, feedwater system, and CVCS are maintained open. The containment evacuation system maintains the CNV environment at a partial vacuum of less than one psia.

After a module is shut down for refueling, preparations are made for transferring the CNV from its operating bay in the RP to the RFP using the Reactor Building crane (RBC). When the RCS is sufficiently cooled, the CNV is partially filled with borated water. This water provides further cooling of the RCS by conductive heat transfer through the RPV to the water in the CNV and through the CNV to the UHS.

With the RCS cooled and partially depressurized, the ECCS valves open, allowing passive flow of borated water between containment and the RPV, until both are filled to a level near the pressurizer baffle plate. When the RCS is sufficiently cooled, all CIVs are closed, thus establishing a passively safe condition and eliminating the need for control connections to the CNV.

4.7.2 Equipment Qualification

The CNV and associated CIVs are designed and constructed to Seismic Category I requirements and are located in the Seismic Category I RXB, providing protection from non-seismic natural phenomena such as tornados, storms, and floods.

4.7.3 System Response to a Loss of All Alternating Current Power

The CNTS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for mitigation strategy implementation are required.

As discussed in Section 4.4.1, the MPS detects the loss of AC power and generates a containment isolation signal. The ESFAS then removes power from all CIV actuator solenoids causing all open CIVs to close.

4.8 Ultimate Heat Sink System

4.8.1 System Design

The purpose of the UHS during off normal operations is to serve as the main system for the removal of heat from the modules. The primary function of the UHS is the removal of heat from various sources via direct water contact including modules, spent fuel, and the DHRS passive condensers. This heat is then transferred to the pool cooling and cleanup system, or during certain conditions, directly to the surrounding environment. The plant safety-related UHS is composed of a large pool complex where the modules and spent fuel are housed. Specifically, the UHS comprises the combined volume of water in and the associated water-retaining structures and components of the RP, RFP, and SFP. The UHS system also includes the assured makeup line and the level instrumentation associated with the pools.

The modules are located in the RP during power operations and are moved to the RFP for refueling and maintenance operations. The operating modules are partially immersed in the RP, and the DHRS passive condensers are submerged in the RP.

The water volume in the RP and RFP portions of the UHS is connected with the water volume in the SFP by the space above the top of the SFP weir wall. Water level depth and temperature in the UHS are maintained during normal operations through an interface with the pool cooling and cleanup system. The pool leakage detection system also monitors for and measures leakage from the UHS to provide early indication of conditions that could reduce the water level depth.

The UHS system includes four level detectors, one each for the RP and RFP, and two for the SFP. The SFP level indicators are located at opposite ends of the SFP to ensure a single event does not cause damage to both instruments. Additionally, because all pool areas communicate while UHS water level is above the weir, each of the four instruments normally provides indication of the single unified pool level. Power is provided to the detectors from EDAS-C through the PPS and pool level is displayed in the MCR on the SDIS displays. Each level instrument {{ }}2(a),(c) capacity independent of the site distribution network, and local readout capability located in an operator-accessible area away from the pool area.

The UHS has a makeup line that meets Seismic Category I standards, and ASME B31.1 Code requirements (Reference 12.6), and is protected from external natural phenomena. (e.g., external flooding, storms such as hurricanes, high winds, and tornadoes; extreme snow, ice, and cold; and extreme heat). The UHS makeup line includes a fire protection connector that facilitates hookup of emergency sources of water for the water supply.

4.8.2 Equipment Qualification

The UHS pool wall is designed to Seismic Category I requirements and is completely contained within the Seismic Category I RXB.

The four pool level instruments are seismically mounted, environmentally qualified, and designed to meet the guidance of NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (Reference 12.7).

4.8.3 System Response to a Loss of All Alternating Current Power

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}}2(a),(c) The UHS survives the BDBEE and remains fully available during the period when the system functions necessary for mitigation strategy implementation are required.

{{

}}2(a),(c)

4.9 Decay Heat Removal System

4.9.1 System Design

Each module has a safety-related DHRS designed to passively remove decay heat in order to establish safe shutdown conditions without onsite or offsite power available. The DHRS consists of two independent trains, each providing passive cooling.

Each train consists of one passive condenser submerged in the RP, two redundant actuation valves configured in parallel above the passive condenser, and piping that connects the condenser to one of the SGs. The actuation valves are located between the module steam line connection and the upper header of the passive condenser. The connection to the steam line is between the CNV and the MSIV to allow DHRS operation following containment isolation. Piping from the lower header of the passive condenser penetrates the CNV and interfaces with the feedwater line for the associated SG.

As discussed in Section 4.4.1, the DHRS is normally maintained in standby with all four DHRS actuation valves held closed by maintaining their associated actuation solenoids energized. This condition maintains the appropriate liquid water inventory in the DHRS condensers and steam piping while allowing the MSIVs and feedwater isolation valves (FWIVs) to remain open for power production.

When the DHRS is actuated, the ESFAS removes power from the solenoids for the DHRS actuation valves, the MSIVs, and the FWIVs. This condition causes the DHRS actuation valves to open, the MSIVs and FWIVs to close, and establishes two separate, two-phase natural circulation loops. The RCS heat is transferred through the SG tubes causing the water to boil and the steam to rise up the steam piping. With the MSIVs closed and the DHRS actuation valves open, the steam then flows into the DHRS passive condenser tubes where it is condensed by the transfer of heat to the

RP through the tubes. The resultant liquid gravity drains to the feedwater lines and returns to the SG. Thus, the natural circulation loops transfer heat from the RCS to the DHRS fluid and then from the DHRS fluid to the RP water.

4.9.2 Equipment Qualification

All components of the safety-related DHRS are located within the Seismic Category I RXB, providing protection from external events. The DHRS condensers, actuation valves, and piping are Seismic Category I components.

The DHRS design internal pressure and temperature are equal to the design pressure of the RPV. Externally, the DHRS piping and condenser are designed to withstand the temperature of saturated RP water and the actuation valves are designed to open under accident conditions.

4.9.3 System Response to a Loss of All Alternating Current Power

Per baseline coping capability of Section 3.1.2, the DHRS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for mitigation strategy implementation are required.

Depending on the operating condition of a given module when the loss of all AC power occurs, the DHRS actuates as a result of the conditions in the module reaching an initiation setpoint or as a result of the MPS detecting the loss of AC power as described in Section 4.4.1.

In all cases, the DHRS actuation valves de-energize and open, and the MSIVs and FWIVs de-energize and close to place the DHRS in service.

4.10 Emergency Core Cooling System

4.10.1 System Design

Each module has a safety-related ECCS designed to provide adequate core cooling by maintaining the core covered with water during all design basis events. The ECCS is unique in that it does not include or require a source of water for injection. The system functions by releasing coolant from the RPV to the CNV to be cooled by direct heat transfer through the CNV wall to the UHS and returning the coolant to the RPV to remove heat from the core. This function is performed without the use of electric power. Instead, ECCS relies on stored energy for changing valve positions and passive motive force (hydrostatic head) for returning coolant to the RPV.

The system includes at least four main valves and their associated hydraulic lines and actuator assemblies. The main valves directly interface with the RPV and are located inside the CNV. There are at least two upper valves and at least two lower valves, the RVVs and the reactor recirculation valves (RRVs), respectively. The actuator assemblies are connected to external CNV nozzles in the vicinity of their associated main valve and are submerged in the RP.

Each main valve actuator assembly includes trip and reset pilot valves and their associated solenoids. Actuator solenoids are used to reposition the pilot valves and subsequently reposition their associated main valves. The electric power used to energize these solenoids is provided by EDAS through the MPS.

The ECCS design of some NuScale Power Module (NPMs) includes a supplemental boron feature located in the CNV. The supplemental boron feature includes solid boron oxide that is passively dissolved and mixed with liquid in the CNV during ECCS operation. The soluble boron recirculates into the RPV to maintain subcriticality. This occurs without operator action or the use of electrical power.

During normal power module operation, all main valves are closed with their pilot trip valve actuator solenoids energized and closed. When the ECCS is actuated, the ESFAS removes ECCS actuator solenoid power, the pilot trip valves are opened, and the main valves open at the appropriate time.

With all main valves open, a two-phase circulation loop is established. Saturated steam leaves the RPV through the RVV flow paths in the pressurizer space and enters the CNV where it is condensed. Saturated and sub-cooled liquid enters the RPV through the RRV flow paths above the core. In addition to mass transfer, heat is removed by conduction through the RPV wall. This heat transfer is negligible during normal operation when containment is evacuated, but during ECCS operation, the lower portions of the RPV wall are submerged and wetted by coolant on both sides, enabling heat transfer. At the same time, heat is transferred from the coolant in the CNV to the UHS.

During steady-state ECCS operation the water levels in the two vessels stabilize above the core. The RVVs have a net steam flow from the RPV to the CNV. The RRVs have a net liquid flow from the CNV to the RPV. Thus, coolant is passively transferred between the CNV and the RPV, and heat is passively transferred from the reactor core to the UHS.

4.10.2 Equipment Qualification

The ECCS valves, hydraulic lines, and actuators are part of the reactor coolant pressure boundary and all components are Seismic Category I. For applicable NPM designs, the safety related components of the ECCS supplemental boron feature are Seismic Category I. All components of the ECCS are located within the Seismic Category I RXB. The ECCS components, including instrumentation, are environmentally qualified for the moisture, chemistry, and radioactivity of expected environments, including those resulting from loss-of-coolant accidents.

4.10.3 System Response to a Loss of All Alternating Current Power

The ECCS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for mitigation strategy implementation are required.

As described in Section 4.4, the MPS detects the loss of AC power that occurs at event initiation, and, after a short time delay, starts the 24-hour ECCS timers. If plant conditions require an ECCS actuation within this 24-hour period, the ECCS valves would open as designed.

After 24 hours, the 24-hour timers expire and de-energize the ESFAS, de-energizing the ECCS trip valve solenoids. The ECCS main valves open to establish system operation.

4.11 Reactor Building Heating, Ventilation, and Air Conditioning System

4.11.1 System Design

The Reactor Building heating, ventilation, and air conditioning (HVAC) system (RBVS) consists of the air handling units (AHUs), filter units, flow dampers, cooling coils, heaters, ductwork, and fans necessary to maintain a suitable environment in the RXB spaces for equipment operation and worker habitability. The RXB spaces include the RP, SFP, and RFP areas, as well as the EDAS battery, battery charger, and module I&C rooms.

During normal operation, power for operation is supplied by the ELVS. If the ELVS is unavailable, the backup power supply system (BPSS) can be used to power the AHUs associated with the EDAS and MPS equipment rooms (battery, battery charger, and I&C rooms), with cooling provided by air-cooled direct expansion cooling coils.

In the event of loss of AC power the isolation dampers located in the normal flow path of the SFP exhaust ductwork fail to their open (i.e., safe) position. The isolation dampers located in the ductwork of the secondary flow path containing the charcoal filters fail to their closed (i.e., safe) position. This damper configuration allows a passive high-efficiency particulate air filtered vent path for the atmosphere within the RXB, providing a monitored release path to the environment for the potentially contaminated air. The stack exhaust discharge is monitored for radiation.

4.11.2 Equipment Qualification

The SSC of the RBVS whose structural failure could affect the operability of safety-related SSC are designed as Seismic Category II. The high energy line break isolation dampers are Seismic Category I.

4.11.3 System Response to a Loss of All Alternating Current Power

The loss of the ELVS at initiation of the loss of all AC power event combined with the assumption that the BPSS is unavailable makes the RBVS unavailable for cooling, heating, and humidity control for the duration of the event.

4.12 Control Room Habitability System

4.12.1 System Design

The CRHS provides breathing air to the MCR and maintains control room pressure during high radiation, release of toxic chemical, or loss of offsite power conditions. The major components used in the CRHS to perform these functions include a breathing air compressor high pressure air storage bottles, solenoid-operated CRE supply line isolation valves, and solenoid-operated CRE pressure relief valves.

The CRHS air bottles and air bottle racks are non-safety related and not risk-significant, but are designed to Seismic Category I criteria to provide reliable, clean breathing air for control room personnel under design basis seismic events and accidents. The specified capacity of the CRHS air bottles is sized such that 25 percent of the bottles may be out of service and still provide a minimum of 72 hours of breathing air. The CRHS air bottles and air bottle racks are located in the CRB just below the CRE.

The CRHS piping is provided with a main CRE supply line that has two solenoid-operated valves in parallel. The two parallel valves provide a redundant air-supply path from the air bottles to the CRE. During normal operation, the solenoids are energized from EDAS-C through the PPS. With the solenoids energized, the CRE supply valves are held closed and the air bottles are maintained in a standby condition.

The CRE pressure relief valves are provided by the CRHS to allow appropriate air flow out of the CRE during CRHS operation. These valves are solenoid operated and are normally maintained closed by maintaining the solenoids energized. The power for the solenoids is provided by EDAS-C through the PPS. When the CRHS is actuated, the PPS removes power from both CRE supply line isolation valve solenoids and both CRE pressure relief valve solenoids. This condition places the CRHS in service and allows air to flow from the air bottles into the CRE.

4.12.2 Equipment Qualification

The CRHS components are in the Seismic Category I portions of the CRB.

The control room HVAC system SSC required to provide breathing air inventory to the CRE for at least 72 hours are designed to Seismic Category I criteria. These SSC are the air storage bottles and the supply piping and components (including regulating, actuation, and isolation valves) to the CRE. The CRE pressure relief piping and components are also designed to Seismic Category I criteria.

The compressor is not required to supply the stored air inventory to the CRE; therefore, the compressor and piping from the compressor to the isolation valve that separates it from the charging header is Seismic Category III.

4.12.3 System Response to a Loss of All Alternating Current Power

Per baseline coping capability of Section 3.1.2, the CRHS air storage bottles, supply piping, regulating valves, and actuation valves are assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for mitigation strategy implementation are required.

As described in Section 4.5, the PPS monitors for and detects the loss of AC power at event initiation. As a result, the PPS removes power from both CRE supply line isolation valve solenoids and both CRE pressure relief valve solenoids. This condition places the CRHS in service and allows air to flow from the air bottles into the CRE. The CRHS continues to supply air to the CRE for a minimum of 72 hours.

4.13 Overhead Heavy Load Handling System

4.13.1 System Design

The overhead heavy load handling system (OHLHS) provides the structural support and mobility needed to lift and move loads in the RXB, including the modules, to support normal operations, maintenance, receipt of new equipment, and refueling activities. The OHLHS includes the RBC, hoists, and heavy load handling devices used in the RXB.

The RBC includes a bridge, trolley, and hoist. The RBC rails allow the bridge to travel over the pool[4] areas of the UHS, as well as the dry dock area. The RBC is designed to set the brakes and hold the load during an SSE.

If seismic activity occurs, an electrical earthquake seismic switch shuts off all power to the RBC and the auxiliary hoists. As a result, the trolley, bridge, and hoists stop and their associated brakes set to stop motion in all axes. The RBC is designed with redundant holding brakes. If one set of holding brakes fails to engage during an event, the other holding brake automatically holds the load. Both holding brake systems are designed and rated to maintain a hoisted load at the maximum allowable crane load.

The trolley includes seismic restraints to ensure it remains on the bridge and the bridge includes seismic restraints to ensure it remains on the runway. This combination of design features ensures a module in transition continues to be suspended by the RBC during and after a seismic event.

The RBC is normally powered by the ELVS. A loss of power to the RBC has the same effect as a seismic event: all RBC motion is stopped and brakes are set on the trolley, bridge, and hoist. Dual brakes are set to ensure the load stays at its current position.

If power cannot be restored, manual RBC operations may be used to position the crane and place a suspended load in a safe location after a seismic event or loss of

4.The RBC system includes limit switches to restrict operations over certain portions of the RFP and SFP.

power. Manual operations of the RBC are accessible from the platforms or decks of the RBC. Manual operations include:

- hoist load may be manually lowered from the trolley machinery deck
- bridge and trolley manual operation by a secondary emergency drive system located in line with the normal drive machinery

4.13.2 Equipment Qualification

The RBC, including the auxiliary hoists, is designed as a single-failure-proof crane in accordance with the requirements of ASME NOG-1 (Reference 12.8) for Type I cranes.

The RBC is Seismic Category I equipment. The RBC is located within the Seismic Category I RXB. The brakes used to maintain the RBC load suspended are designed to withstand the environmental conditions that exist in the RXB during a loss of all AC power.

4.13.3 System Response to a Loss of All Alternating Current Power

Per baseline coping capability of Section 3.1.2, the RBC is assumed to survive the BDBEE and remain capable of maintaining a design capacity load suspended for the duration of the loss of all AC power event.

When the loss of ELVS power occurs at event initiation, the RBC is de-energized. As a result, RBC motion is stopped and brakes are set on the trolley, bridge, and hoist.

4.14 Communications System

The communications system will include at least one on-site and one off-site communications system capable of remaining functional during a loss of all AC power event to include the loss of local area communications infrastructure due to a BDBEE. Fixed and portable satellite communications will be provided to meet this requirement. Portable satellite communications devices, batteries, and battery chargers will be designated for this specific purpose by the licensee.

4.15 {{ }}2(a),(c)

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}}2(a),(c)

Table 4-1 {{ }}2(a),(c)

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}}2(a),(c)

5.0 Safety Functions during a Loss of All Alternating Current Power

In addition to analyzing the integrated plant response and key safety functions for a loss of all AC power event with all modules initially operating, the effects of a loss of all AC power event on the key safety functions for a module transitioning to refueling and for a module being refueled is also analyzed.

5.1 Integrated Plant Response

The following description of the overall plant response is based on the descriptions of the individual system responses provided in Section 4.0 and is intended to facilitate the discussions regarding the key safety functions in this section. {{

}}2(a),(c)

Per Section 3.1.5 initial event conditions and assumptions, the initiating event is an unspecified external event that results in the loss of offsite power and the loss of all site AC power. For the design, the initiating event results in the loss of all power, except for that power provided by the EDAS.

At event initiation, the MPS and the PPS detect the loss of AC power. The MPS starts the delay timer for the loss of AC power actuations. The PPS isolates the CRE and actuates the CRHS to pressurize the CRE from the breathing air bottles. The loss of feedwater caused by the loss of power combined with the steam turbine generator trip causes a significant reduction in heat removal by the secondary system, and RCS temperature and pressure rise.

{{

}}2(a),(c) At this point in the event, the reactor is subcritical and remains subcritical during cooldown, the CNV is isolated, and the DHRS is passively providing decay heat removal. With these conditions established, the module passively establishes safe shutdown.

During the next 24 hours, RCS pressure and temperature decrease as decay heat continues to be transferred to the UHS by DHRS operation. During this cooldown and depressurization of the RCS, the differential pressure between the RPV and the CNV decreases. The MPS actuates ECCS automatically prior to 24 hours if necessary. If not already actuated at 24 hours, the ECCS hold mode timers expire and the ESFAS de-energizes. This de-energization results in all the ECCS main valves opening to establish passive decay heat removal via the ECCS.

The opening of the ECCS valves causes a rise in CNV pressure and temperature, as designed. The condensing of the reactor coolant steam on the inner surface of the CNV and the subsequent heat transfer to the UHS through the CNV limits the pressure transient and subsequently reduces CNV pressure.

At this point in the event, all functions required for coping with the loss of all AC power event have been completed or are passively occurring. {{

}}2(a),(c)

The UHS temperature rises at event initiation because of the loss of power to the pool cooling systems followed by the initiation of the DHRS for all modules. {{

}}2(a),(c)

5.2 Core Cooling

Adequate core cooling is provided by plant safety-related systems following initiation of a loss of all AC power event. {{

}}2(a),(c)

5.2.1 Reactor Coolant System Inventory

The design of the module and its safety-related core cooling systems relies on coolant inventory control rather than coolant inventory makeup to ensure the core remains covered and adequate core cooling is provided in a loss of all AC power event. During normal operation, RCS inventory is sufficient for ECCS operation and plant cooldown. {{

}}2(a),(c)

NEI 12-06 (Reference 12.4) provides a list of five reactor coolant inventory loss sources for pressurized water reactors (PWRs) and boiling water reactors (BWRs). Because the NuScale design has unique aspects that differ from traditional PWR and BWR designs, each of the five listed sources below is evaluated for applicability. The following list includes the expected leakage source followed by an explanation of why the listed source {{

}}2(a),(c)

5.2.1.1 Normal system leakage

During normal operation, the CVCS is the only system in direct communication with the RCS that includes a pathway for coolant outside of the CNV. {{

}}2(a),(c)

{{

}}2(a),(c)

Per technical specifications, a module may operate at full power with a certain allowable amount of primary-to-secondary leakage in each SG. As described in Section 4.7 and Section 4.9 of this report, the MSIVs and FWIVs close {{ }}2(a),(c)

Other potential leakage paths that could be considered "normal system leakage" would result in leakage into the CNV. This result is important because, as described in Section 4.10 of this report, the CNV is part of the normal ECCS flow path and function. Thus, leakage from these sources would be preserved, as designed, for ECCS operation and would not result in the need for inventory addition.

5.2.1.2 Losses from letdown unless automatically isolated or until isolation is procedurally directed.

As described in Section 3.2.1 and Section 5.1, the CVCS is automatically isolated by the closure of safety-related CIVs and eliminates this potential leakage pathway outside of containment {{ }}2(a),(c)

5.2.1.3 Losses due to reactor coolant pump seal leakage (rate is dependent on the reactor coolant pump seal design).

The NuScale design does not include reactor coolant pumps or similar components for which seal leakage would reduce inventory. Therefore, there are no losses due to reactor coolant pump seal leakage.

5.2.1.4 Losses due to BWR recirculation pump seal leakage

The NuScale design does not include recirculation pumps or similar components for which seal leakage would reduce inventory. Therefore, there are no losses due to recirculation pump seal leakage.

5.2.1.5 BWR inventory loss due to operation of steam-driven systems, safety-relief valve cycling, and RPV depressurization.

The NuScale design does not include steam-driven systems that use RCS inventory.

The safety-relief valve cycling in a BWR is similar to reactor safety valve cycling for a module. However, when a reactor safety valve cycles, the RCS inventory exiting through the valve is condensed and collected in the CNV. The NuScale design preserves the RCS inventory for ECCS operation and does not result in the need for inventory addition. The plant does not depressurize by reducing RCS inventory. Actuation of ECCS valves reduces RCS pressure, but is not a reduction in RCS inventory. Therefore, there is no inventory loss due to operation of the reactor safety valves.

5.2.1.6 Containment Leakage

In addition to the five potential sources of leakage described above, containment leakage is evaluated as a possible path for the loss of RCS inventory in a loss of all AC power event scenario, because the RCS partially transfers to the CNV following ECCS actuation. {{

}}2(a),(c)

5.2.1.7 Conclusion

The functions necessary to ensure adequate RCS inventory {{

}}2(a),(c)

5.2.2 Reactivity Control

To establish and maintain safe shutdown conditions during a loss of all AC power event, the module is made subcritical and then maintained subcritical during the cooldown that follows.

Per boundary conditions in Section 3.1.3, when the reactor trip occurs in the loss of all AC power event, all control rods fully insert. This action achieves initial subcriticality; however, depending on the time in the life cycle, for some module designs, the control rods alone may not provide sufficient negative reactivity to compensate for the positive reactivity added as the RCS cools.

During the loss of all AC power {{

}}2(a),(c)

5.2.3 Decay Heat Removal

As detailed in Section 4.9, the safety-related DHRS actuates without operator action after the loss of all AC power event and, although one train has the capacity to remove enough decay heat to establish safe shutdown conditions, both fully independent trains of the DHRS are placed into operation on each module. Safe shutdown conditions are established with both trains of the DHRS in service and passively transferring RCS heat to the UHS. The DHRS continues to passively reduce RCS temperature and perform the core cooling function until ECCS actuates by the 24-hour mark.

With the DHRS in operation, RPV pressure and temperature are reduced. Within 24 hours the ECCS valves actuate. The natural circulation flow path established by the actuation of the ECCS causes most of the reactor coolant to bypass the SGs. As a result, the decay heat removal provided by the DHRS is reduced and the ECCS assumes the core cooling function.

Both the DHRS and the ECCS utilize UHS inventory to perform their core cooling safety function. As described in Section 4.8, the modules are partially immersed and the DHRS passive condensers are submerged in the UHS. With both systems transferring RCS heat from the modules to the UHS and the addition of decay heat from the spent fuel in the SFP, the UHS temperature rises and begins to boil. It is at this point that UHS level begins to lower. Eventually, the UHS level lowers to the top of the DHRS passive condensers. The design ensures the DHRS can provide adequate core cooling beyond the first 24 hours of the loss of all AC power event.

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}}2(a),(c)

5.2.4 Core Cooling Process Variables

The key safety function {{

}}2(a),(c). However, the process variables listed in Table 5-1 are available to assure control room operators that the safety-related systems have performed as designed.

Per baseline coping capability of Section 3.1.2, the instrumentation associated with each process variable is assumed to survive the BDBEE and remain fully available for a duration beyond the time necessary for the associated function to be established and monitored.

Table 5-1 Core Cooling Key Process Variables

Function	Process Variables for Assuring the Function is Established
RCS inventory control	RPV water level Containment isolation valve positions
Reactivity control	Neutron flux Core exit temperature RTB status CVCS containment isolation valve positions
DHRS decay heat removal	DHRS valve positions Core exit temperature
SSCIV positions	MSIV bypass isolation valve position FWIV positions MSIV position
ECCS decay heat removal	ECCS valve positions Containment water level RPV water level Core exit temperature SFP level[1]

[1] Spent fuel pool level provides indication of UHS level when UHS level is above the SFP weir.

5.3 Containment

The containment function is provided and maintained by plant safety-related systems {{

}}2(a),(c)

As previously discussed, the CNV, in addition to providing the containment function, also functions to support core decay heat removal. As part of this heat removal function, containment temperature and pressure are maintained within their design limits. During a loss of all AC power event, {{

}}2(a),(c)

5.3.1 Containment Temperature and Pressure

Rather than relying on an active containment heat removal system (e.g., fan cooler system, spray system), containment heat removal is ensured passively as an inherent consequence of the physical configuration wherein each CNV is partially immersed in the reactor pool portion of the UHS.

During the loss of all AC power event, mass and energy are released into the CNV when the ECCS valves open. This release represents the highest pressure and temperature conditions for the CNV during the event. After the ECCS valves open, the reactor coolant water is accumulated in the CNV and is passively returned to the reactor vessel by natural circulation. Under these conditions, the CNV provides an interfacing medium for core decay and containment heat removal. Specifically, the steel walls of the CNV, together with the heat transfer medium surrounding the CNV (i.e., the UHS), serve as a passive means to remove heat from the high-energy fluid released into the CNV. This passive heat transfer ensures that CNV temperature and pressure do not approach values that could adversely affect the integrity of the CNV.

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}}2(a),(c)

5.3.2 Containment Process Variables

The key safety function of containment is established and maintained {{
}}2(a),(c). However, the process variables listed in Table 5-2 are available to assure the control room operators that the safety-related systems have performed as designed.

Per baseline coping capability of Section 3.1.2, the instrumentation associated with each process variable is assumed to survive the BDBEE and remain fully available for a duration beyond the time necessary for the associated mitigation function to be established and monitored.

Table 5-2 Containment Key Process Variables

Function	Process Variables for Assuring the Function is Established
Containment isolation	CIV positions
Containment heat removal	Wide range containment pressure
	SFP level[1]

[1]Spent fuel pool level provides indication of UHS level when UHS level is above the SFP weir.

5.4 Spent Fuel Cooling

Spent fuel in the SFP is {{

}}2(a),(c)

5.4.1 Spent Fuel Pool Level

As described in Section 4.8, the SFP is part of the UHS and normally communicates with the RFP and RP through the space above the SFP weir wall. As such, the pools respond as a single volume during a loss of all AC power event until UHS level lowers below the weir wall.

With the loss of pool cooling systems at the initiation of the event, the transfer of heat from the modules due to the loss of all AC power event and the decay heat addition from spent fuel in the SFP, UHS temperature rises and the pool begins to boil. At this point, UHS level begins to decrease, and without inventory addition, reaches the top of the SFP weir wall.

With the SFP now hydraulically separated from the other pools in the UHS, level reduction continues based on spent fuel decay heat. {{

}}2(a),(c)

5.4.2 Spent Fuel Pool Cooling Process Variables

The key safety function of SFP cooling {{
}}2(a),(c)

Per baseline coping capability of Section 3.1.2, the SFP level instruments are assumed to survive the BDBEE and remain fully available for a duration beyond the time necessary for the associated strategy to be established and monitored.

5.5 Transition Mode (MODE 4)

As described in Section 4.7.1, a module is transferred from its operating bay in the RP to the RFP to perform refueling activities. In preparation for the transfer, the CNV is flooded with borated water to the pressurizer baffle plate level, the module is cooled down, the ECCS valves are opened, and the CIVs are closed. In this condition, decay heat is passively transferred from the RCS to the UHS through the CNV, and safe shutdown conditions are maintained.

The module is then lifted above the RP floor using the RBC, and transported to the CNV flange tool in the RFP. At the CNV flange tool, the module is lifted above the RP floor to clear the top of the tool and allow the module to be positioned for placement in the tool.

5.5.1 Core Cooling

Although the module conditions established in preparation for refueling, including placement in the CNV flange tool, ensure adequate core cooling is maintained, the most restrictive core cooling conditions for a module in transition occur when the module is lifted to the maximum lift height. This is limiting because the maximum lift height represents the minimum CNV immersion in the UHS.

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}}2(a),(c)

5.6 Refueling Mode (MODE 5)

During refueling operations, the module is disassembled, and the bottom head of the RPV, that includes the reactor core, is located in the RPV flange tool. Refueling is then conducted by transferring fuel between the reactor core and the SFP fuel storage racks over the weir in the SFP weir wall. Because the top of the RPV flange tool and the top of the spent fuel storage racks are below the height of the SFP weir wall, the highest lift point during fuel movement occurs as the fuel passes over the weir. {{

}}2(a),(c)

5.6.1 Dry Dock

The RXB contains a dry dock adjacent to the UHS. During refueling, the dry dock is utilized to inspect the upper section of a module. A dry dock gate separates or allows passage between the dry dock and the refueling pool. When the dry dock gate is closed, the dry dock may contain a water level below the water level of the UHS. Failure of the dry dock gate results in a decrease in the water level of the UHS as water transfers from the UHS into the dry dock. Design of the dry dock gate as safety-related Seismic Category I ensures that the dry dock gate is available during the MBDBE. {{

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5.7 Baseline Coping Capability

Based on an analysis of the key safety functions coping capability provided by installed plant equipment, a baseline coping capability can be established. The baseline capability is summarized in Table 5-3.

Table 5-3 Baseline Coping Capability Summary

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5.8 Indefinite maintenance of core cooling, containment, and spent fuel pool cooling capabilities

As required by 10 CFR 50.155(b)(1)(ii), the strategies and guidelines for loss of all AC power concurrent with loss of normal access to the normal heat sink must include support for maintaining core cooling, containment, and spent fuel pool cooling capabilities indefinitely or until sufficient site functional capabilities can be maintained without the need for mitigation strategies. The requirements of 10 CFR 50.155(b)(1)(ii) {{

}}2(a),(c)

In the Fukushima Daiichi accident, personnel began adding water to the spent fuel pool of Unit 4 via fire and concrete pump trucks nine days after the tsunami, and began injecting water via the fuel pool cooling system 14 days after the tsunami (Reference 12.5). Offsite AC power was restored to Units 1 and 2 and Station Blackout conditions ended after nine days. AC power was restored to Units 3 and 4 and Station Blackout conditions ended after 15 days. Water injection through a stationary motor driven pump connected to off-site power was established within 23 days (Reference 12.5). These actions occurred with no pre-planning or pre-staging of resources, without the benefit of a hardened pool makeup connection, and despite widespread destruction limiting access to offsite resources.

The Fukushima response time is {{ }}2(a),(c) of the design to maintain a passive safe state of the modules and spent fuel pool. Therefore, this timeframe provides more than a sufficient amount of time {{

}}2(a),(c) Through the passive systems of the plant design, sufficient core cooling, containment, and spent fuel pool cooling capabilities are maintained during the BDBEE. {{

}}2(a),(c)

6.0 Capability to respond to a LOLA due to explosions or fire, as required by 10 CFR 50.155(b)(2).

This section assesses the requirements and associated guidance to meet 10 CFR 50.155(b)(2), extensive damage mitigation guidelines, for the design of the plant. 10 CFR 50.155(b)(2) states:

> *"Extensive damage mitigation guidelines - Strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant impacted by the event, due to explosions or fire, to include strategies and guidelines in the following areas:*
>
> *i) Firefighting*
>
> *ii) Operations to mitigate fuel damage*
>
> *iii) Actions to minimize radiological release."*

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Each area of consideration has expectation elements that must be addressed in the appropriate regulatory submittal (i.e., topical report, design certification application, or combined license application). This topical report addresses the applicable expectation elements.

6.1 Limited Impacts of LOLA on the Module

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Exterior concrete walls and multiple interior walls provide further protection from an external threat. The RXB exterior walls are designed to preclude aircraft perforation in the aircraft impact assessment and therefore would prevent significant damage to components inside the RXB from an external explosion or fire.

Personnel access restrictions, robust module bay walls, and biological shield design also provide adequate protection from an internal threat. The horizontal portion of the bioshield is reinforced concrete encased in stainless steel. The bioshield reinforced concrete slab encloses the top of the CNV. The module bay walls enclose the CNV on three sides. On the last side (facing the center of the UHS), the vertical face of the biological shield extends below the UHS nominal water level to prevent a reactor pool surface fire from entering under the biological shield. The vertical face is constructed of a steel member frame with radiation panels. The radiation panels are arranged in an overlapping pattern: the front face panel and rear face panel alternate to provide coverage such that air is vented from the operating bay both during normal operations and if there is a high energy line break in the bay. The overlap is staggered to provide a torturous path for fire and smoke, and to minimize potential projectiles entering the operating bay through the bioshield. A fire under the bioshield that can damage redundant safe shutdown equipment is not credible based on the limited combustible loading, the robust component design, and the circuit designs.

Given the access restriction and the design of the bioshield structure, the components under the bioshield (e.g., on the top of the module), the CNV itself, and the components located within are not subject to direct impact from LOLA events.

6.2 Phase 1 - Enhanced firefighting capabilities

The guidance for firefighting response to a LOLA event includes operational aspects of responding to explosions or fire including prearranging for involvement of outside organizations, planning and preparation activities (e.g., pre-positioning equipment, personnel, and materials to be used for mitigating the event), and developing procedures and training for managing the event.

The design features fire detection, notification, and suppression systems, as designed, installed, and maintained in accordance with applicable codes and standards. The design incorporates the following features to cope with potential fires that could affect module or plant safety:

- redundant safety systems to perform safety-related functions, such as reactor shutdown and core cooling
- physical separation between redundant trains of safety-related equipment used to mitigate the consequences of a design-basis accident
- passive design that minimizes the need for support systems and the potential effects of “hot shorts”
- annunciation of fire indication in the main control room to facilitate personnel response

- no electrical power requirement for mitigating design-basis events as safety systems are fail-safe on loss of power

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6.3 Phase 2 - Measures to mitigate damage to fuel in the SFP

The SFP is described in Section 4.8. NEI 06-12 Section 2.1, SFP Strategies, states in part:

> *"NOTE: These strategies are not required for sites that have spent fuel pools that are below grade and can not be drained."*

The SFP is below grade and cannot be drained. The pool walls are designed to Seismic Category I requirements and are completely contained within the Seismic Category I RXB. All pipe connections to and from the pool are at an elevation below the normal operating level but above the minimum pool level required for SFP radiation shielding and heat removal or are protected by a siphon break. This configuration prevents inadvertent lowering of the pool level below safety limits. Therefore, as stated in NEI 06-12 Section 2.1, SFP mitigation strategies are not required for the plant. A licensee who adopts this topical report will not need to describe spent fuel cooling strategies in response to LOLA events.

6.4 Phase 3 - Measures to mitigate damage to fuel in the reactor vessel and to minimize radiological release

Phase 3 consists of two parts: measures to enhance command and control, and a set of PWR-specific reactor and containment mitigation strategies. The command and control measures are aimed at improving initial site operation response before the Emergency Response Organization is fully activated. NEI 06-12, Section 3.2.1 lists 12 basic assumptions for developing extensive damage mitigation guidelines, including no warning of imminent threat, loss of access to the control room, and the loss of all AC and DC power. {{

}}2(a),(c)

The generic list of PWR-specific key safety functions identified in NEI 06-12 were developed based on a traditional PWR plant design and are identified below:

- RCS inventory control
- RCS heat removal
- Containment isolation
- Containment integrity
- Release mitigation

NEI 06-12, Section 4.2.3, states in part:

> *"It is recognized that new plants typically have more safety trains that are more spatially separated than for current US 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 identified in Sections 3.3 and 3.4 or may need additional strategies to satisfy the key safety functions."*

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6.4.1 Assessment of Key Safety Functions

RCS Inventory Control

The purpose of this key safety function is to ensure that the core is covered with water. If the core remains covered with water, core damage will be precluded. Traditional PWR designs accomplish this function by injecting water into the RPV using systems such as safety injection. The module 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 and is accomplished by CNTS (Section 4.7). The CVCS is a non-safety alternate means to perform this function. When DHRS (Section 4.9) or ECCS (Section 4.10) are actuated, the core will remain covered.

The design does not have reactor coolant pumps; therefore, there is no potential for loss of inventory through reactor coolant pumps seal leakage due to lack of seal cooling, as may occur in a station blackout event for traditional PWR designs. Leakage could occur through the CIVs, but the leakage rate is small and RCS makeup is not required.

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RCS 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 primary means for heat removal during steady state, startup and hot shutdown operations is through the steam generators. The alternate means for RCS heat removal is the DHRS or the ECCS when the ECCS valves open. During DHRS or ECCS operations, external feedwater injection is not required. No portable SG level measurement is required. Decay heat from the core can be transferred to the UHS through the isolated containment as demonstrated in Section 5.0.

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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. Traditional PWR designs accomplish this function by using CIVs. The plant also relies on CIVs to accomplish this function. The Containment System design and isolation is discussed in Section 4.7.

The CIVs for interfacing systems, with the exception of feedwater and main steam (which are not part of the reactor coolant pressure boundary), have dual valve, single body CIVs outside of the CNV. This isolation capability is located under the bioshield. This dual containment isolation capability is both the primary and alternate means of performing this function. CIVs are energized open so a loss of DC power to those valves will result in their repositioning to their safe or accident response position.

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}}2(a),(c)

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. Traditional PWR designs accomplish this function through the use of containment sprays to control containment pressure and temperature below applicable limits as well as quenching debris that relocates outside of the RPV during a severe accident.

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}}2(a),(c)

Release Mitigation

The purpose of this key safety function is to minimize a radiological release assuming severe core damage occurs and a radiological release is imminent or in progress. Traditional PWRs perform this function by scrubbing the containment atmosphere using containment sprays. However, a release that is in progress may not be susceptible to the scrubbing effects, and a release from containment or the RXB may nonetheless occur.

This same phenomenon is applicable for a release from the CNV to the environment. The RPV is located within the CNV. The CNV is partially immersed in the UHS. The UHS is the primary means to perform this function.

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}}2(a),(c)

An analysis of beyond-design-basis core damage events provides reasonable assurance that, even in the extremely unlikely event of a severe accident and release, the design features and site characteristics provide adequate protection of the public when compared to the acceptance criteria of 10 CFR 52.137(a)(2)(iv).

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}}2(a),(c) to satisfy 10 CFR 50.155(b)(2)(iii).

6.5 LOLA Summary and Conclusions

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7.0 Capacity, capability, and protection of equipment associated with mitigation of events described in the rule, as required by 10 CFR 50.155(c).

This section assesses the requirements and associated guidance to meet 10 CFR 50.155(c) for the design of the plant. 10 CFR 50.155(c) states:

> *(c) Equipment (1) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must have sufficient capacity and capability to perform the functions required by paragraph (b)(1) of this section.*
>
> *(2) The equipment relied on for the mitigation strategies and guidelines required by paragraph (b)(1) of this section must be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.*

10 CFR 50.155(c) requires the equipment relied on for the mitigation strategies and guidelines to have sufficient capacity and capability to perform the key safety functions and to be reasonably protected from the effects of natural phenomena that are equivalent in magnitude to the phenomena assumed for developing the design basis of the facility.
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}}2(a),(c)

8.0 Training requirements as defined by 10 CFR 50.155(d)

This section assesses the requirements and associated guidance to meet 10 CFR 50.155(d) for the design of the plant. 10 CFR 50.155(d) states:

> *"Each licensee shall provide for the training of personnel that perform activities in accordance with the capabilities required by paragraphs (b)(1) and (2) of this section."*

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9.0 Spent fuel pool monitoring after final fuel removal from the reactor vessel, as required by 10 CFR 50.155(e).

This section assesses the requirements and associated guidance to meet 10 CFR 50.155(e) for the design of the plant. 10 CFR 50.155(e) states:

> *"In order to support effective prioritization of event mitigation and recovery actions, each licensee shall provide reliable means to remotely monitor wide-range water level for each spent fuel pool at its site until 5 years have elapsed since all of the fuel within that spent fuel pool was last used in a reactor vessel for power generation."*

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}}2(a),(c)

Permanently installed water level instrumentation in the SFP is capable of monitoring water level from the normal UHS level to the top of the stored spent fuel in the SFP. The instrumentation is described in Section 4.8 as part of the UHS. The SFP level instrumentation design meets the guidance of NEI 12-02, Section 3 (Reference 12.7), as endorsed by NRC Regulatory Guidance.

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The SFP level instrumentation is part of the permanently installed plant equipment and is capable of providing SFP level indication for the specified timeframe.

10.0 Emergency Planning

A combined or operating license applicant provides a proposed emergency plan description along with the schedule for implementation for combined licenses. 10 CFR 50.160 or 50.47(b) and Appendix E establish the emergency plan content for small modular reactors that will be used by an applicant. These requirements include paragraph 50.160(b)(1)(iii) roles for emergency response performance including communications and command and control. Additionally, preplanning activities for emergencies as described in 50.160(b)(1)(iv)(A) includes established plans for direction of effective control during an emergency, and maintenance of the plan including contacts and arrangements.

Emergency planning accounts for coordination with local, state, federal, and tribal agencies for eventualities that require emergency action. Compliance with this regulation in conjunction with the aforementioned installed plant equipment accomplishes the goal of sufficient response to BDBEEs. An adopter of this topical report will provide with their application their emergency plan as part of the submittal. The established contingencies in that plan account for the necessary preplanning for a NuScale facility to respond to a BDBEE {{

}}2(a)(c)

11.0 Conclusion

The module design incorporates design features that provide enhanced capabilities for coping with an extended loss of electrical power, loss of normal access to the normal heat sink, and loss of large areas due to explosions or fire. {{

}}2(a),(c)

The requirements of 10 CFR 50.155 (b) through (e) are satisfied based on the following provisions:

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}}2(a),(c)

12.0 References

12.1 U.S. Nuclear Regulatory Commission, “Mitigation of Beyond-Design-Basis Events; Final Rule,” (10 CFR 50.155), Federal Register, Vol. 84, August 8, 2019, pp.39684-79722.

12.2 U.S. Nuclear Regulatory Commission, “Strategies and Guidance to Address Loss of Large Areas of the Plant Due to Explosions and Fires,” NUREG-0800, Section 19.4, Revision 0, June 2015.

12.3 Nuclear Energy Institute, “B.5.b Phase 2 & 3 Submittal Guideline,” NEI 06-12, Revision 3, July 2009, Agencywide Document Access and Management System (ADAMS) Accession No. ML092890400.

12.4 Nuclear Energy Institute, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” NEI 12-06, Revision 4, December 2016.

12.5 IAEA “The Fukushima Daiichi Accident” Technical Volume 1/5 Description and Context of the Accident” Sept 30, 2012.

12.6 American Society of Mechanical Engineers, *Power Piping*, ASME Code for Pressure Piping, B31, ASME B31.1, New York, NY, 2018.

12.7 Nuclear Energy Institute, “Industry Guidance for Compliance with NRC Order EA-12-051, To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,” NEI 12-02, Revision 1, August 2012, Agencywide Document Access and Management System (ADAMS) Accession No. ML1224A307.

12.8 American Society of Mechanical Engineers, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), ASME NOG-1-2010, New York, NY.

Section C

Audit Question Number	NuScale Letter Number
A-MBDBE.LTR-8	LO-170692 (ML24178A398)

LO-170692

June 26, 2024

Docket No. 99902078

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Topical Report "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P, Revision 1 and Docketing of Resolved Audit Responses

REFERENCES: 1. Letter from NuScale to NRC, LO-141315, "NuScale Power, LLC Submittal of Topical Report 'NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155,' TR-141299, Revision 0" dated September 11, 2023

NuScale Power, LLC (NuScale) hereby submits Revision 1 of the "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P. The purpose of this submittal is to provide the updated revision of the topical report for development of the NRC Safety Evaluation Report. Additionally, NuScale is providing responses to select audit questions to support development of the NRC Safety Evaluation Report.

Enclosure 1 contains the proprietary version of the report entitled, "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P, Revision 1. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 5) supports this request. Enclosure 2 contains the nonproprietary version of the report (TR-141299-NP).

Enclosure 3 contains the proprietary version of NuScale's individual response to audit questions requested by the NRC to be docketed. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 5) supports this request. Enclosure 4 contains the nonproprietary version of the response.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

NuScale Power, LLC
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If you have any questions, please contact Kristopher Cummings at 240-833-3003 or kcummings@nuscalepower.com.

Sincerely,

Thomas Griffith
Manager, Licensing
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, New Reactor Licensing Branch Chief, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Thomas Hayden, Project Manager, NRC

Enclosure 1: "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P, Revision 1, proprietary version
Enclosure 2: "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-NP, Revision 1, nonproprietary version
Enclosure 3: NuScale audit response, proprietary version
Enclosure 4: NuScale audit response, nonproprietary version
Enclosure 5: Affidavit of Mark W. Shaver, AF-170693

Enclosure 4:

NuScale audit response, nonproprietary version

Response to NuScale Topical Report Audit Question

Question Number: A-MBDBE.LTR-8

Receipt Date: 03/04/2024

Question:

Clarify whether any active components credited in the TR for BDBEE have performance requirements that exceed the safety-related design and performance criteria established in the design bases.

Response:

The performance requirements for active components credited for beyond-design-basis events in the topical report are consistent with the safety-related design and performance criteria established in the design bases. Section 5.0 describes the safety functions during the shutdown of a NuScale Power Module. These systems initiate a safe-shutdown condition as required by their design functions and corresponding basis. After the initiation of safe-shutdown in response to an event, no additional active systems are required to maintain a safe condition as a result of a beyond-design-basis event. {{

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No changes to the LTR are necessary.

Enclosure 3:

Affidavit of Mark W. Shaver, AF-182501

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairsof NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the process by which NuScale develops its NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155.

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

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

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

(4) The information sought to be withheld is in the enclosed report entitled, "NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," TR-141299-P-A, Revision 1. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 02, 2025.

Mark W. Shaver

Mark W. Shaver