Mitigation Strategies for Extended Loss of AC Power Event

November 2016

Revision 0

Docket: PROJ0769

NuScale Power, LLC

1100 NE Circle Blvd., Suite 200

Corvallis, Oregon 97330

www.nuscalepower.com

Department of Energy Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0000633.

CONTENTS

TABLES

FIGURES

Abstract

This report provides an assessment of the NuScale Power Plant response to an extended loss of AC power (ELAP) event and addresses the COL applicant requirements and guidance for developing the site-specific FLEX mitigating strategies. The report also addresses how the Near Term Task Force (NTTF) recommendations and subsequent lessons learned that resulted from the Fukushima Dai-ichi events are dispositioned for the NuScale design certification application (DCA).

Executive Summary

After the Fukushima Dai-ichi events, the Nuclear Regulatory Commission (NRC) established a senior-level task force to conduct a systematic review of NRC regulations and processes to determine if the agency should make safety improvements. Commission Paper SECY-11-0093 published the recommendations of the task force, which was referred to as the Near-Term Task Force (NTTF). These recommendations resulted in NRC Orders, 10 CFR 50.54(f) letters, rulemaking, and industry guidance. The Orders, letters, rulemaking, and industry guidance do not impose any requirements to a design certification applicant, but do impose requirements and guidance for existing licensees and combined license (COL) applicants.

This report provides an assessment of the NuScale Power Plant response to an extended loss of AC power (ELAP) event and addresses the COL applicant requirements and guidance for developing the site-specific FLEX mitigating strategies. The report also addresses how the NTTF recommendations and subsequent lessons learned that resulted from the Fukushima Dai-ichi events are dispositioned for the NuScale design certification application (DCA).

The NuScale Power Plant is designed to maintain core cooling, spent fuel pool cooling, and containment integrity, independent of alternating current (AC) or direct current (DC) power sources, for an extended duration.

The industry guidance document for developing FLEX mitigating strategies for an ELAP event, NEI 12-06 (Reference 13.1.16), is based on specific assumptions and design features relevant to existing light water reactor plant designs that were not designed to maintain these key safety functions without AC or DC power sources for extended durations. NuScale has an inherently different design that does not entirely align with these assumptions or design features. For example, the NuScale Power Plant was designed to utilize installed plant systems and components to cope with an ELAP event. The NuScale Power Plant was also designed to minimize, to the extent practical, operator actions required to cope indefinitely with an ELAP. This design results in a significantly longer coping period (greater than 72 hours) before portable FLEX equipment may be required and a reduction in the size and number of total portable equipment needed. Thus, many of the strategies detailed in NEI 12-06 are not required and do not have a significant improvement in safety for the NuScale design. This report provides an evaluation of the NEI 12-06 guidance, including applicability determinations for each method and performance attribute associated with a Pressurized Water Reactor, and verification that the intended purpose of each applicable recommendation is met either through the design of the NuScale Power Plant or through a FLEX mitigating strategy.

1.0 Introduction

1.1 Purpose

The purpose of this report is to provide an assessment of the NuScale Power Plant response to an extended loss of AC power (ELAP) event and to address the COL applicant requirements and guidance for developing the site-specific FLEX mitigating strategies. The report also addresses how the NTTF recommendations and subsequent lessons learned that resulted from the Fukushima Dai-ichi events are dispositioned for the NuScale DCA.

1.2 Scope

The background, regulatory requirements, and guidance associated with the lessons learned from the Fukushima Dai-ichi events are described in Section 2 of this report. The NuScale approach to determining the applicable external hazards for FLEX strategies is described in Section 3. Section 4 of the report summarizes the assumptions, general criteria, and prescriptive guidance on initial plant conditions from NEI 12-06 that should be utilized in developing FLEX mitigating strategies. A discussion of the applicable SSC in the NuScale Power Plant design that are credited for coping with an ELAP is included in Section 5 of the report. This section includes a description of the design, qualification, and the NuScale Power Plant response to an ELAP. Additional description of the expected environmental conditions for buildings and rooms that contain equipment credited for FLEX strategies is provided in Appendix B. Section 6 of the report describes the integrated response of the NuScale Power Module to an ELAP including analysis results. Section 7 summarizes these strategies and defines the coping durations for each phase (i.e., Phase 1, 2 and 3). Sections 8 through 10 describe additional lessons learned guidance such as configuration management, portable FLEX equipment, procedures and personnel needed to support a beyond-design-basis event. Appendix A of this report includes an evaluation of the applicability determinations for each method and performance attribute described in Appendix D of NEI 12-06, as well as verification that the intended purpose of each applicable recommendation is met either through the design of the NuScale Power Plant or through a FLEX strategy.

1.3 Definitions, Acronyms and Abbreviations

Table 1-1 Abbreviations

Term	Definition
AAPS	auxiliary AC power supply
AC	alternating current
AHU	air handling unit
BDBEE	Beyond design basis external event
BDG	backup diesel generator
BPSS	backup power supply system

Term	Definition
BWR	boiling water reactor
CFR	Code of Federal Regulations
CIV	containment isolation valve
CNTS	containment system
CNV	containment vessel
COL	combined license
CRB	Control Building
CRDM	control rod drive mechanism
CRE	control room envelope
CRHS	control room habitability system
CRVS	control room HVAC system
CVCS	chemical and volume control system
DC	direct current
DCA	Design Certification Application
DHRS	decay heat removal system
ECCS	emergency core cooling system
EDMG	extensive damage mitigation guidelines
EDNS	normal DC power system
EDSS	highly reliable dc power system
EDSS-C	EDSS – common
EDSS-MS	EDSS – module specific
EHVS	13.8kV and switchyard system
ELAP	extended loss of alternating current power
ELVS	low voltage AC electrical distribution system
EMVS	medium voltage AC electrical distribution system
EOP	emergency operating procedure
ERDS	emergency response data system
ESF	engineered safety feature
ESFAS	engineered safety features actuation system
FLEX	Diverse and flexible coping strategies based on NRCs Fukushima task force recommendations
FSG	FLEX Support Guideline
FWIV	feedwater isolation valve
HPAC	high pressure air compressor
HVAC	heating, ventilation, and air conditioning
IAB	inadvertent actuation block
IEEE	Institute of Electrical and Electronics Engineers
I/O	input/output
LUHS	loss of normal access to the ultimate heat sink
MLA	module lifting adapter
MPS	module protection system

Term	Definition
MSIV	main steam isolation valve
NEI	Nuclear Energy Institute
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
NTTF	Near-Term Task Force
PAM	post-accident monitoring
PCS	plant control system
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
RG	Regulatory Guide
RP	reactor pool
RPCS	reactor pool cooling system
RPV	reactor pressure vessel
RTB	reactor trip breaker
RTS	reactor trip system
RVV	reactor vent valve
RXB	Reactor Building
SAMG	severe accident mitigation guidelines
SBO	station blackout
SCWS	site cooling water system
SDIS	safety display and indication system
SECY	Secretary of the Commission, Office of the NRC
SFP	spent fuel pool
SFPCS	spent fuel pool cooling system
SG	steam generator
SRV	safety relief valve
SSCIV	secondary system containment isolation valve
SSC	structures, systems, and components
SSE	safe shutdown earthquake
UHS	ultimate 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 4. Has sufficient capacity and reliability for operation of all systems required for coping with SBO and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident).
Class 1E	The safety classification of the electrical equipment and systems essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal or otherwise essential in preventing significant release of radioactive material to the environment. Class 1E is a functional term. Equipment and systems are to be classified Class 1E if they fulfill the functions listed in the definition. Identification of systems or equipment as Class 1E based on anything other than their function is an improper use of the term and should be avoided.
Emergency core cooling system (ECCS) hold mode	The operating mode of EDSS – module specific (EDSS-MS) for a period of up to 24 hours after commencing loss of AC power operation. Power is supplied to the ECCS valves, allowing them to remain closed until a valid initiation signal is received or 24 hours elapses.
Extended loss of alternating current power (ELAP)	The loss of offsite power, backup diesel generators (BDGs), and any alternate AC source as defined in 10 CFR 50.2, but not the loss of AC power from buses fed by station batteries through inverters.
Harsh environment	An environment resulting from a design basis event, i.e., loss-of-coolant accident, high-energy line break, or main steam line break.
Loss of AC power (EDSS)	The operation of highly reliable DC power system (EDSS) following a complete loss of AC electric power to the essential and nonessential switchgear buses for a specified period of time.

Term	Definition
Loss of normal access to the ultimate heat sink (LUHS)	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 ultimate heat sink (UHS) is lost, but the water inventory in the UHS remains available and robust piping connecting the UHS to plant systems remains intact. The motive force for UHS flow, i.e., service water or circulating water pumps, is assumed to be lost with no prospect for recovery.
Mild environment	An environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.
NuScale Power Module (NPM)	The NPM is a self-contained nuclear steam supply system composed of a reactor core, a pressurizer, and two steam generators (SGs) integrated within the reactor pressure vessel (RPV) and housed in a compact steel containment vessel (CNV).
PAM-only mode	The operating mode of the EDSS-MS during loss-of-AC-power operation upon completion of ECCS hold-mode operation. The EDSS – common (EDSS-C) operates in this mode for the entire 72 hours. During this mode, power is supplied only to equipment required for display of post-accident monitoring (PAM) variables.
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.
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 (e.g., Regulatory Guide 1.76, Revision 1), or has been 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 sufficient to maintain the reactor coolant system (RCS) less than or equal to 420F.
Safe shutdown earthquake (SSE)	The design basis earthquake for commercial nuclear power plants. It is that earthquake which 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.

Term	Definition
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.

2.0 Background

The Fukushima Dai-ichi accident in March 2011 resulted from a tsunami that exceeded the station's design basis, and flooded the site's emergency power supplies and electrical distribution system. This extended loss of power severely compromised the key safety functions of core cooling and containment integrity that ultimately led to core damage in three reactors. The loss of power also impaired the spent fuel pool (SFP) cooling function, but the pools retained sufficient water inventory to preclude fuel damage from loss of cooling. The inability to determine the status of the SFP diverted operator attention that could have been used for reactor core cooling and containment integrity.

The events at Fukushima Dai-ichi highlight the possibility that extreme natural phenomena can challenge the prevention, mitigation, and emergency preparedness defense-in-depth layers (i.e., a low probability but high consequence event). At Fukushima Dai-ichi, limitations in time and unpredictable conditions associated with the accident significantly challenged attempts by the responders to preclude core damage and containment failure. During the events at Fukushima Dai-ichi, the plant operators faced challenges beyond any faced previously at a commercial nuclear reactor. Mitigation of the BDBEE required additional actions. The diverse and flexible coping strategies based on NRC's Fukushima NTTF recommendations (FLEX) methodology increase defense-in-depth for beyond design basis scenarios such as an ELAP and LUHS occurring simultaneously at all units on a site.

2.1 Regulatory Requirements and Guidance

Following the earthquake and tsunami at the Fukushima Dai-ichi nuclear power plant, the NRC established a senior-level task force referred to as the NTTF. The NTTF conducted a systematic review of NRC regulations and processes to determine if the agency should make safety improvements in light of the events in Japan. The NTTF's report, "Recommendations for Enhancing Reactor Safety in the 21st Century," was provided for Commission consideration in a Commission Paper, SECY-11-0093 (Reference 13.1.1). Then SECY-11-0124 (Reference 13.1.1) and SECY-11-0137 (Reference 13.1.2) established the NRC staff's recommendations and prioritization of the recommendations. Table 2-1 provides a listing of the NTTF recommendations, their status, and their applicability to the NuScale Power Plant design.

Table 2-1 Near-Term Task Force recommendations

Number	Recommendation	Comment
1	The Task Force recommends establishing a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	General recommendation.
1.1	Draft a Commission policy statement that articulates a risk-informed defense-in-depth framework that includes extended design basis requirements in the NRC's regulations as essential elements for ensuring adequate protection.	NRC action, not applicable to NuScale DCA.
1.2	Initiate rulemaking to implement a risk-informed, defense-in-depth framework consistent with the above recommended Commission policy statement.	NRC action, not applicable to NuScale DCA.
1.3	Modify the Regulatory Analysis Guidelines to more effectively implement the defense-in-depth philosophy in balance with the current emphasis on risk-based guidelines.	NRC action, not applicable to NuScale DCA.
1.4	Evaluate the insights from the IPE and IPEEE efforts as summarized in NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," issued December 1997, and NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," issued April 2002, to identify potential generic regulations or plant-specific regulatory requirements.	NRC action, not applicable to NuScale DCA.
2	The Task Force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design basis seismic and flooding protection of SSC for each operating reactor.	General recommendation.
2.1	Order licensees to reevaluate the seismic and flooding hazards at their sites against current NRC requirements and guidance, and if necessary, update the design basis and SSC important to safety to protect against the updated hazards.	Existing licensee requirement, not applicable to NuScale DCA. The COL applicants will use the latest seismic and flooding information for the site location.

Number	Recommendation	Comment
2.2	Initiate rulemaking to require licensees to confirm seismic hazards and flooding hazards every 10 years and address any new and significant information. If necessary, update the design basis for SSC important to safety to protect against the updated hazards.	NRC action, not applicable to NuScale DCA.
2.3	Order licensees to perform seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features such as watertight barriers and seals in the interim period until longer term actions are completed to update the design basis for external events.	Existing licensee requirement, not applicable to NuScale DCA.
3	The Task Force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	NRC action, not applicable to NuScale DCA.
4	The Task Force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design basis and BDBEE.	General recommendation.
4.1	Initiate rulemaking to revise 10 CFR 50.63 to require each operating and new reactor licensee to (1) establish a minimum coping time of 8 hours for a loss of all AC power, (2) establish the equipment, procedures, and training necessary to implement an “extended loss of all AC” coping time of 72 hours for core and SFP cooling and for RCS and primary containment integrity as needed, and (3) preplan and prestage offsite resources to support uninterrupted core and SFP cooling, and RCS and containment integrity as needed, including the ability to deliver the equipment to the site in the time period allowed for extended coping, under conditions involving significant degradation of offsite transportation infrastructure associated with significant natural disasters.	10 CFR 50.63 is not being revised; new regulation 10 CFR 50.155 is in the rulemaking process (Reference.13.1.3). This technical report addresses the proposed regulation.

Number	Recommendation	Comment
4.2	Order licensees to provide reasonable protection for equipment currently provided pursuant to 10 CFR 50.54(hh)(2) from the effects of design basis external events and to add equipment as needed to address multiunit events while other requirements are being revised and implemented.	NRC Order EA-12-049 (Reference 13.1.4) and JLD-ISG-2012-01 (Reference 13.1.5) issued. This technical report addresses the Order requirements and related guidance.
5	The Task Force recommends requiring reliable hardened vent designs in boiling water reactor (BWR) facilities with Mark I and Mark II containments.	General recommendation.
5.1	Order licensees to include a reliable hardened vent in BWR Mark I and Mark II containments.	Not applicable to NuScale Power Plant design.
5.2	Reevaluate the need for hardened vents for other containment designs, considering the insights from the Fukushima accident. Depending on the outcome of the reevaluation, appropriate regulatory action should be taken for any containment designs requiring hardened vents.	Not applicable to NuScale Power Plant design.
6	The Task Force recommends, as part of the longer term review, that the NRC identify insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	NRC action – closed, no additional requirements (Reference 13.1.6).
7	The Task Force recommends enhancing SFP makeup capability and instrumentation for the SFP.	General recommendation.
7.1	Order licensees to provide sufficient safety-related instrumentation, able to withstand design basis natural phenomena, to monitor key SFP parameters (i.e., water level, temperature, and area radiation levels) from the control room.	NRC Order EA-12-051 (Reference 13.1.7) and JLD-ISG-2012-03, Revision 0, (Reference 13.1.8) were issued. This technical report addresses the Order requirements and related guidance.
7.2	Order licensees to provide safety-related ac electrical power for the SFP makeup system.	Consolidated with NTTF Recommendation 4 (Reference 13.1.10) and incorporated in proposed regulation (Reference 13.1.3). This technical report addresses the proposed requirements and guidance.

Number	Recommendation	Comment
7.3	Order licensees to revise their technical specifications to address requirements to have one train of onsite emergency electrical power operable for SFP makeup and SFP instrumentation when there is irradiated fuel in the SFP, regardless of the operational mode of the reactor.	Consolidated with NTTF Recommendation 4 (Reference 13.1.10) and incorporated in proposed regulation (Reference 13.1.3). This technical report addresses the proposed requirements and related guidance.
7.4	Order licensees to have an installed seismically qualified means to spray water into the SFP including an easily accessible connection to supply the water (e.g., using a portable pump or pumper truck) at grade outside the building.	Consolidated with NTTF Recommendation 4 (Reference 13.1.10) and incorporated in proposed regulation (Reference 13.1.3). This technical report addresses the proposed requirements and related guidance.
7.5	Initiate rulemaking or licensing activities or both to require the actions related to the SFP described in detailed recommendations 7.1–7.4.	Consolidated with NTTF Recommendation 4 (Reference 13.1.10) and incorporated in proposed regulation (Reference 13.1.3). This technical report addresses the proposed requirements and related guidance.
8	The Task Force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, severe accident mitigation guidelines (SAMG), and EDMG.	General recommendation.
8.1	Order licensees to modify the EOP technical guidelines (required by Supplement 1, “Requirements for Emergency Response Capability,” to NUREG-0737, issued January 1983 (GL 82-33), to (1) include EOPs, SAMG, and EDMG in an integrated manner, (2) specify clear command and control strategies for their implementation, and (3) stipulate appropriate qualification and training for those who make decisions during emergencies.	Incorporated in proposed rule 10 CFR 50.155 (Reference 13.1.3). Partially addressed in this technical report. COL applicant item.
8.2	Modify Section 5.0, “Administrative Controls,” of the standard technical specifications for each operating reactor design to reference the approved EOP technical guidelines for that plant design.	Recommendation was not pursued.

Number	Recommendation	Comment
8.3	Order licensees to modify each plant's technical specifications to conform to the above changes.	Recommendation was not pursued.
8.4	Initiate rulemaking to require more realistic, hands-on training and exercises on SAMG and EDMG for all staff expected to implement the strategies and those licensee staff expected to make decisions during emergencies, including emergency coordinators and emergency directors.	Incorporated in proposed rule 10 CFR 50.155 (Reference 13.1.4). Partially addressed in this technical report. COL applicant item.
9	The Task Force recommends that the NRC require that facility emergency plans address prolonged SBO and multiunit events.	General recommendation.
9.1	Initiate rulemaking to require emergency planning (EP) enhancements for multiunit events in the following areas: • personnel and staffing • dose assessment capability • training and exercises • equipment and facilities	Incorporated in proposed rule 10 CFR 50.155 (Reference 13.1.3). Partially addressed in this technical report. COL applicant item.
9.2	Initiate rulemaking to require EP enhancements for prolonged SBO in the following areas: • communications capability • ERDS capability • training and exercises • equipment and facilities	Incorporated in proposed rule 10 CFR 50.155.

Number	Recommendation	Comment
9.3	Order licensees to do the following until rulemaking is complete: • determine and implement the required staff to fill all necessary positions for responding to a multiunit event • add guidance to the emergency plan that documents how to perform a multiunit dose assessment (including releases from spent fuel pools) using the licensee's site-specific dose assessment software and approach • conduct periodic training and exercises for multiunit and prolonged SBO scenarios. Practice (simulate) the identification and acquisition of offsite resources, to the extent possible • ensure that EP equipment and facilities are sufficient for dealing with multiunit and prolonged SBO scenarios • provide a means to power communications equipment needed to communicate onsite (e.g., radios for response teams and between facilities) and offsite (e.g., cellular telephones, satellite telephones) during a prolonged SBO • maintain ERDS capability throughout the accident	March 12, 2012, 10 CFR 50.54(f) letter issued to existing licensees. This technical report partially addresses the letter requirements and guidance that will be codified in the proposed rule 10 CFR 50.155.
9.4	Order licensees to complete the ERDS modernization initiative by June 2012 to ensure multiunit site monitoring capability.	NuScale ERDS is discussed in DCA Part 2, Tier 2, Section 7.2.
10	The Task Force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	General recommendation.
10.1	Analyze current protective equipment requirements for emergency responders and guidance based upon insights from the accident at Fukushima.	NRC action, not applicable to NuScale DCA.
10.2	Evaluate the command and control structure and the qualifications of decision makers to ensure that the proper level of authority and oversight exists in the correct facility for a long-term SBO or multiunit accident or both.	NRC action, not applicable to NuScale DCA. Incorporated in proposed rule 10 CFR 50.155.

Number	Recommendation	Comment
10.3	Evaluate ERDS to do the following: • determine an alternate method (e.g., via satellite) to transmit ERDS data that does not rely on hardwired infrastructure that could be unavailable during a severe natural disaster • determine whether the data set currently being received from each site is sufficient for modern assessment needs	NRC action, not applicable to NuScale DCA.
11	The Task Force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision making, radiation monitoring, and public education.	General Recommendation.
11.1	Study whether enhanced onsite emergency response resources are necessary to support the effective implementation of the licensees' emergency plans, including the ability to deliver the equipment to the site under conditions involving significant natural events where degradation of offsite infrastructure or competing priorities for response resources could delay or prevent the arrival of offsite aid.	NRC action, not applicable to NuScale DCA. Incorporated in proposed rule 10 CFR 50.155 (Reference 13.1.3).
11.2	Work with the Federal Emergency Management Agency, States, and other external stakeholders to evaluate insights from the implementation of EP at Fukushima to identify potential enhancements to the U.S. decision making framework, including the concepts of recovery and reentry.	NRC action, not applicable to NuScale DCA.
11.3	Study the efficacy of real-time radiation monitoring onsite and within the emergency planning zones (including consideration of ac independence and real-time availability on the Internet).	NRC action, not applicable to NuScale DCA.
11.4	Conduct training, in coordination with the appropriate Federal partners, on radiation, radiation safety, and the appropriate use of KI in the local community around each nuclear power plant.	NRC action, not applicable to NuScale DCA.

Number	Recommendation	Comment
12	The Task Force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the regulatory oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	NRC action, not applicable to NuScale DCA.
12.1	Expand the scope of the annual regulatory oversight process self-assessment and biennial regulatory oversight process realignment to more fully include defense-in-depth considerations.	NRC action, not applicable to NuScale DCA.
12.2	Enhance NRC staff training on severe accidents, including training resident inspectors on SAMG.	NRC action, not applicable to NuScale DCA.

In order to address the NTTF recommendations, the following guidance documents were developed based on involvement with stakeholders and nuclear industry representatives:

- NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities" (Reference 13.1.14)
- 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 13.1.15)
- NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (Reference 13.1.16)
- NEI 13-06, "Enhancements to Emergency Response Capabilities for Beyond Design Basis Events and Severe Accidents" (Reference 13.1.17)
- NEI 14-01, "Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents" (Reference 13.1.18)

The guidance documents listed above do not impose any design requirements or a response by a design certification applicant; however, this technical report discusses how the NuScale Power Plant is designed to mitigate an ELAP and addresses the requirements and guidance for the combined license (COL) applicant. The NTTF recommendations most relevant to the design of a new nuclear power plant are Recommendations 4.2 and 7.1 (see Table 2-1).

2.1.1 Near-Term Task Force Recommendation 4.2

The NRC determined that NTTF Recommendation 4.2 concerning mitigation strategies was a high-priority action. Based upon the NTTF recommendation, the NRC issued Order EA-12-049 (Reference 13.1.4). This Order required provisions for mitigation strategies for BDBEEs.

The Order "*requires a three-phase approach for mitigating beyond-design-basis external events. The initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling. The transition phase requires providing sufficient, portable, onsite equipment and consumables to maintain or restore these functions until they can be accomplished with resources brought from off site. The final phase requires obtaining sufficient offsite resources to sustain those functions indefinitely*" (Reference 13.1.4). These phases are shown below.

Initial Phase (Phase 1):	initially cope by relying on installed plant equipment.
Transition Phase (Phase 2):	transition from installed plant equipment to on-site FLEX equipment.
Final Phase (Phase 3):	obtain additional capability and redundancy from off-site equipment until power, water, and coolant injection systems are restored or commissioned.

The industry developed NEI 12-06 (Reference 13.1.16) to provide guidelines for nuclear power plants to assess extreme external event hazards and implement the mitigation strategies specified in NRC Order EA-12-049. The NRC subsequently issued Interim Staff Guidance JLD-ISG-2012-01 (Reference 13.1.5), which endorsed NEI 12-06 with clarifications on determining baseline coping capability and equipment qualification.

The consequences of postulated BDBEEs that are most impactive to reactor safety are loss of power and loss of the UHS. NEI 12-06 outlines an approach for adding diverse and flexible mitigation strategies—or FLEX— to increase defense-in-depth for beyond design basis scenarios to address an ELAP and LUHS occurring simultaneously at all units on a site.

FLEX strategies consist of the following elements:

- portable equipment that provides means of obtaining power and water to maintain or restore key safety functions for all reactors at a site
- reasonable staging and protection of portable equipment from BDBEEs applicable to a site
- procedures and guidance to implement FLEX
- programmatic controls that ensure the continued viability and reliability of FLEX

FLEX strategies consist of an on-site component using equipment stored at the plant site and an off-site component for providing additional materials and equipment for longer-term response.

The purpose of NEI 12-06 is to outline the process that can be used by individual licensees to define and implement site-specific diverse and flexible mitigation strategies that reduce the risks associated with beyond design basis conditions.

2.1.2 Near-Term Task Force Recommendation 7.1

The NRC also determined that NTTF Recommendation 7.1 concerning capability to ensure reliable spent fuel pool level instrumentation was a high-priority action. Based upon the NTTF recommendation, the NRC issued Order EA-12-051 (Reference 13.1.7). This Order requires provisions for reliable spent fuel pool level instrumentation.

NEI 12-02 (Reference 13.1.14) provides guidelines for the design, qualification, installation, and testing, for reliable spent fuel pool level instrumentation to meet NRC Order EA-12-051. The NRC subsequently issued Interim Staff Guidance JLD-ISG-2012-03 (Reference 13.1.8), which endorsed NEI 12-02 as an acceptable method for satisfying NRC Order EA-12-051.

2.1.3 Rulemaking

Subsequent to the NRC Orders, 10 CFR 50.155 was proposed to codify the requirements in the NRC Orders and associated information requests issued under 10 CFR 50.54(f). The proposed rule, as published in the Federal Register on November 13, 2015 (Reference 13.1.3), objectives are to:

- make the requirements in Order EA–12–049 and Order EA–12–051 generically applicable, giving consideration to lessons learned from implementation of the Orders,
- establish new requirements for an integrated response capability,
- establish new requirements for actions that are related to onsite emergency response, and
- address a number of petitions for rulemaking submitted to the NRC following the March 2011 Fukushima Dai-ichi event.

Although the proposed regulation is in the rulemaking process the regulation is discussed in this Technical Report to document the review of the proposed regulation and effect on the NuScale Power Plant design certification application[1].

2.1.3.1 Rulemaking Scope

The proposed rulemaking addresses, either in requirements or through supporting implementation guidance, all of the recommendations in NTTF Recommendations 4, 7, 8, 9.1, 9.2, 9.3 (with one exception: emergency response data system (ERDS) modernization is addressed, but maintenance of ERDS capability throughout the accident is not addressed), 10.2, and 11.1.

This proposed rulemaking requires the integration of the following guideline sets with the emergency operating procedures (EOPs):

- guidelines in response to Order EA-12-049, FLEX Support Guidelines (FSGs)

[1] The latest revisions of the NEI documents, not necessarily the revision referenced in the rulemaking, were used for this review

- guidelines in response to 10 CFR 50.54(hh)(2)[2] referred to as extensive damage mitigation guidelines (EDMG) through most of the industry

Additional requirements in 10 CFR 50 Appendix E for expanded onsite response (site-wide) for communications and staffing, and capability to assess multiple source term dose is also included in the proposed rulemaking package (Reference 13.1.4).

The proposed rulemaking is applicable to holders of an operating license for a nuclear power reactor under Part 50 or COL under Part 52. The rule is not applicable to design certifications submitted under 10 CFR 52.47.

2.1.3.2 NRC Draft Guidance

The NRC developed the following draft Regulatory Guides (RGs) in support of the proposed rulemaking for 10 CFR 50.155:

- DG-1301 "(Proposed New Regulatory Guide 1.226), Flexible Mitigation Strategies for Beyond-Design-Basis Events" (Reference 13.1.19). The RG endorses, with clarification, NEI 12-06 (Reference 13.1.16) as acceptable to meet these portions of proposed regulation (10 CFR 50.155).
- DG-1317 "(Proposed New Regulatory Guide 1.227), Wide-Range Spent Fuel Pool Level Instrumentation" (Reference 13.1.20). The RG endorses, with exception and clarification, NEI 12-02 (Reference 13.1.14) as acceptable to meet these portions of proposed regulation (10 CFR 50.155).
- DG-1319 "(Proposed New Regulatory Guide 1.228), Integrated Response Capabilities For Beyond-Design-Basis Events" (Reference 13.1.21). The RG endorses NEI 12-01 (Reference 13.1.14), NEI 13-06 (Reference 13.1.16), and NEI 14-01 (Reference 13.1.17) with clarification as acceptable to meet these portions of proposed regulation (10 CFR 50.155).

---

[2] Regulation 10 CFR 50.54(hh)(2) may be moved to new regulation 10 CFR 50.155, presently in rulemaking process reference Federal Register notice Vol. 80, No. 219 pages 70610-701647

3.0 Applicable Hazards

For determining the applicable external hazards, the approach outlined in this report is to perform a generic assessment of the capability of a standard plant design licensed under 10 CFR 52.47 and establish a set of bounding design criteria for the specific external hazards. NEI 12-06 (Reference 13.1.16) states that the following five classes of extreme external hazards need to be addressed:

- seismic events
- external flooding
- storms such as hurricanes, high winds, and tornadoes (including missiles)
- snow and ice storms
- extreme cold and heat

The NEI 12-06 guidance also states that all sites should evaluate seismic events and address the impact of high temperatures on the storage, deployment, and operation of FLEX equipment. The applicability of the other external hazards is determined on a site-specific basis by a licensee.

A COL applicant that references the NuScale Power Plant design certification will utilize a similar approach for determining applicable external hazards as outlined in Appendix F of NEI 12-06 for the AP1000 plant designs. A COL applicant that utilizes the NuScale Power Plant design will review the proposed site for applicability of each hazard and address the expected hazard compared to the NuScale design criteria.

The NuScale standard plant design for site parameters (see Table 2-1, Site Parameters, in the NuScale DCA Part 2, Tier 2) demonstrates the wide range of extreme environmental conditions covered by the design. Because of the conservatisms that are incorporated into the selection of these site environmental conditions, these conditions are expected to bound extreme site-specific values.

The design criteria for the above external hazards are discussed in more detail in the following NuScale DCA sections:

- seismic events: addressed in DCA Part 2, Tier 2, Section 3.7.1
- external flooding: addressed in DCA Part 2, Tier 2, Section 3.4.2
- storms such as hurricanes, high winds, and tornadoes (including missiles) : addressed in DCA Part 2, Tier 2, Sections 3.3 and 3.5
- snow and ice storms: addressed in DCA Part 2, Tier 2, Sections 3.8.4
- extreme cold and heat: addressed in DCA Part 2, Tier 2, Table 2.0-1

The COL applicant will review the proposed site for applicability of each hazard. If the design criteria specified in the NuScale DCA does not bound the expected site-specific hazards, then the COL applicant will need to address the expected external hazard in its FLEX strategies.

Portable FLEX equipment that is stored far enough from the site (i.e., an offsite storage location) such that it would not be subjected to the hazard that affected the site need not be designed or qualified for any of the assumed hazards. In addition, the storage arrangements (building, etc.) would not be required to be designed or qualified to survive the external hazard. As described in this report, the inherent coping time of the NuScale Power Plant design is sufficiently long (at least greater than 72 hours), that there is significant time to transport equipment from an off-site storage location. Use of more than one storage location is not necessary as long as the storage site is far enough away from the affected site(s) that the same external hazard could not affect both the operating units at the plant(s) and the FLEX equipment storage location. In this way, the storage location would not be required to be built to survive the external hazard protection described in Section 4 of NEI 12-06. This approach is reasonable considering the amount of time available to procure the equipment from an offsite or remote storage location.

4.0 Nuclear Energy Institute 12-06 Summary

4.1 Diverse and Flexible Coping Strategies

4.1.1 Objective

The objective of the FLEX mitigating strategies is to establish an indefinite coping capability to prevent damage to the fuel in the reactor and SFPs and to maintain the containment function by using plant equipment and FLEX equipment during an ELAP coincident with a LUHS.

4.1.2 Strategies

The FLEX strategies are focused on maintaining or restoring the key plant safety functions of core cooling, containment, and spent fuel cooling, and are not specific to a damage state or external event type. In some cases, additional hazard-specific boundary conditions are applied in order to cause the implementation strategies to be focused on the likely challenges for that specific external event. A safety function-based approach facilitates the utilization of the FLEX strategies in support of the operating and emergency response network of procedures and guidance.

The strategies for coping with these conditions involve a three-phase approach:

Phase 1 - Initially cope by relying on plant equipment
Phase 2 - Augment or transition from plant equipment to on-site FLEX equipment and consumables to maintain or restore key functions
Phase 3 - Obtain additional capability and redundancy from off-site FLEX equipment until power, water, and coolant injection systems are restored or commissioned

Development of FLEX strategies begins with establishing the plant’s ability to cope with the baseline conditions for simultaneous ELAP and LUHS events. This allows the duration of each phase to be determined by the:

- on-site availability of equipment
- resources required for the deployment and activation of the equipment consistent with the established coping timeline
- site conditions expected following the BDBEE
- ability of the site to receive resources and equipment from off-site

4.1.3 Boundary Conditions

The following general boundary conditions apply to the establishment of FLEX strategies (Reference 13.1.14):

1. BDBEE occurs impacting all units at site[3]
2. All reactors on-site initially operating at power, unless the site has procedural direction to shut down due to the impending event[4],
3. Each reactor 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.
5. No independent, concurrent events, e.g., no active security threat.
6. All personnel on-site are available to support site response.
7. Spent fuel in dry storage is outside the scope of FLEX.

4.2 Baseline Coping Capability Criteria, Conditions, and Assumptions

The baseline coping capability is built upon strategies that address a simultaneous ELAP and LUHS caused by an unspecified event. To support the development of these strategies, baseline assumptions have been established on the presumption that, other than the ELAP and the LUHS, the following two basic conditions are true (Reference 13.1.16):

1. Plant equipment that is designed to be robust with respect to design basis external events is assumed to be fully available.
2. Plant equipment that is not robust is assumed to be unavailable.

4.2.1 General Criteria

The following general criteria are to be used in establishing the baseline coping capability (Reference 13.1.16):

- Procedures and equipment relied upon should ensure that satisfactory performance of necessary fuel cooling[5] and containment functions are maintained.
- The fuel in the reactor is required to remain covered at all times.
- The fuel in the SFP is required to remain covered at all times.

4.2.2 Initial Plant Conditions

The initial plant conditions are assumed to be the following (Reference 13.1.16):

1. Prior to the event the reactor has been operating at 100 percent rated thermal power for at least 100 days.

---

[3] For the NuScale design, this means all 12 NPMs are impacted.
[4] For the NuScale design, this means all 12 NPMs are initially operating at 100% power.
[5] “Fuel cooling” refers to both the fuel in the reactor core and the fuel in the SFP.

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 FLEX strategies, provided there is an adequate basis for the assumed value.

4.2.3 Initial Event Conditions and Assumptions

NEI 12-06 provides a generic list of event initial conditions and assumptions to apply while determining the baseline coping capability. The following are those conditions and assumptions (Reference 13.1.16):

1. No specific initiating event is used. The initial condition is assumed to be a loss of offsite power 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. The loss of offsite power is assumed to affect all units at a plant site.
2. All design basis installed sources of emergency onsite ac power and SBO alternate ac power sources are assumed to be not available and not imminently recoverable.
3. Station batteries and associated dc buses along with ac power from buses fed by station batteries through inverters remain available.
4. Cooling and makeup water inventories contained in systems or structures with designs that are robust for the applicable hazards are available.
5. Normal access to the UHS is lost, but the water inventory in the UHS remains available and robust piping connecting the UHS to plant systems remains intact. The motive force for UHS flow, i.e., service water or circulating water pumps, is assumed to be lost with no prospect for recovery. Fire or other pumps may be available provided they are robust for the applicable hazard.
6. Fuel for FLEX equipment stored in structures with designs that are robust for the applicable hazard remains available.
7. Plant equipment that is contained in structures with designs that are robust for the applicable hazard is available.
8. Other equipment, such as portable ac power sources, portable back up dc power supplies, spare batteries, and loss-of-large-area equipment, may be used as onsite FLEX equipment provided it is reasonably protected from the applicable external hazards and has predetermined hookup strategies with appropriate procedures or guidance and the equipment is stored in a relative close vicinity of the site.
9. Installed electrical distribution system, including inverters and battery chargers, remain available provided they are protected consistent with current station design.
10. No additional events or failures are assumed to occur immediately prior to or during the event, including security events.
11. The fire protection system ring header as a water source is acceptable if the header is robust for the applicable hazards.

4.2.4 Reactor Transient Assumptions

NEI 12-06 specifies the following additional boundary conditions for the reactor transient following the BDBEE (Reference 13.1.16):

1. Following the loss of all ac power, the reactor automatically trips and all rods are inserted.
2. The main steam system valves (such as main steam isolation valves (MSIVs), turbine stops, atmospheric dumps, etc.) necessary to maintain decay heat removal functions operate as designed.
3. Safety relief valves (SRVs) or power operated relief valves (PORVs) initially operate in a normal manner if conditions in the RCS so require. Normal valve reseating is also assumed.
4. No independent failures other than those causing the simultaneous ELAP and LUHS events are assumed to occur in the course of the transient.

4.2.5 Reactor Coolant Inventory Loss Assumption

Because the document is written to existing plant designs, NEI 12-06 states that reactor coolant inventory loss is expected and provides a list of sources for the leakage. As a result of this expected leakage, RCS makeup capability is assumed to be required at some point during the ELAP. The following is a list of the given potential leakage sources that apply to pressurized water reactors (PWRs) (Reference 13.1.16):

1. normal system leakage
2. losses from letdown unless automatically isolated or until isolation is procedurally directed
3. losses due to reactor coolant pump seal leakage

4.2.6 Spent Fuel Pool Conditions

The following conditions are established for the SFP in NEI 12-06 for determining baseline coping capabilities (Reference 13.1.16):

1. All boundaries of the SFP are intact, including the liner, gates, and transfer canals.
2. Although sloshing may occur during a seismic event, the initial loss of SFP inventory does not preclude access to the refueling deck around the pool.
3. SFP cooling system is intact, including attached piping.
4. SFP heat load assumes the maximum design basis heat load for the site.

4.3 Shutdown Modes

With existing plant designs in mind, NEI 12-06 states, “Due to the small fraction of the operating cycle that is spent in an outage condition, generally less than 10%, the probability of a BDBEE occurring during any specific outage configuration is very small.” Given this assumption, the combination of the existing requirements for a systematic approach to shutdown safety risk identification and planning and the availability of the FLEX equipment is viewed as the most effective way of enhancing safety during shutdown, rather than the development of FLEX strategies explicitly designed for outage conditions.

Guidance is provided for the enhancement of the shutdown risk process and procedures through incorporation of the FLEX equipment, and the following provisions for shutdown modes should be included (Reference 13.1.16):

1. Primary and alternate connection points for core cooling.
2. Core cooling pumps are sized to provide core cooling for outage conditions.
3. Identify a source of borated water for core cooling (the borated water source does not need to be robust for all external events). The key is to have sufficient water sources. If the borated water source is not robust for an external hazard applicable to the site, other water sources robust for that hazard should be identified to back it up. If the backup water source is not borated, then consideration should be given to controlled use to minimize dilution.
4. A means to remove heat from containment, e.g., venting.

NEI 12-06 also provides the clarification that analyses are only needed to support provisions 1, 2, and 4 above, and not for the purposes of determining the sequence of events of an ELAP during shutdown conditions.

5.0 NuScale Power Plant Systems and Responses to an Extended Loss of Alternating Current Power

To develop a FLEX strategy, the baseline coping capability of the NuScale Power Plant design must be determined. This determination is made by evaluating the status of the three key safety functions during the integrated plant response to an ELAP. To understand the integrated plant response, the systems that either contribute to the ELAP conditions for which coping is necessary or function to cope with those conditions must be evaluated to determine their responses and availability during an ELAP. This section describes the relevant functionality of those systems, evaluates their qualifications for the purpose of determining availability during an ELAP, and describes the designed response to an ELAP.

5.1 Reactor Building

The Reactor Building (RXB) is a safety-related, Seismic Category I building that houses, supports, and protects the NPMs and the UHS, in addition to other safety-related components and systems.

The RXB is designed to withstand design basis natural phenomena including earthquakes, floods, high winds (including associated missiles), and extreme temperatures.

5.2 Control Building

The Control Building (CRB) contains safety-related, Seismic Category I areas up to and below elevation 120.0 ft. The portions of the CRB above elevation 120.0 ft are nonsafety-related and Seismic Category II. The safety-related portions include the main control room (MCR) located on elevation 76'-6" of the CRB.

All elevations of the CRB are designed for design basis natural phenomena loads based on the designation of the building area (i.e., Seismic Category I or Seismic Category II).

5.3 Alternating Current Distribution Systems

5.3.1 System Design

During normal operation, onsite AC electrical power distribution is provided by three systems; the 13.8kV and switchyard system (EHVS), the medium voltage AC electrical distribution system (EMVS), and the low voltage AC electrical distribution system (ELVS).

The EHVS buses normally receive electric power from the steam turbine generators and can be powered by the auxiliary AC power supply (AAPS) or the offsite power network during abnormal conditions. Power is then distributed to the EMVS buses for 4160 VAC loads and to supply the 480 VAC buses in the ELVS.

5.3.2 System Response to an Extended Loss of Alternating Current Power

The EHVS, EMVS, and ELVS systems are assumed to be unavailable without the prospect for recovery per initial event assumption number 1 of Section 4.2.3.

5.4 Backup Power Supply System

5.4.1 System Design

The backup power supply system (BPSS) consists of two backup diesel generators (BDGs), a site-specific auxiliary AAPS, and the support equipment required to operate and distribute the power from these components.

The primary function of the BDGs is to provide 480 VAC power to certain loads, including the highly reliable DC power system (EDSS) battery chargers, in the post-72 hour period following an SBO. However, the BDGs are designed to automatically start approximately 30 seconds after sensing a loss of power to all EHVS buses, and to be available for loading approximately two minutes later. The output breakers for the BDGs do not automatically close, but can be closed remotely by the control room operators or locally by field operators.

The AAPS type is site-specific, but regardless of type it is used to supply 13.8kV power to the site when no other sources of AC power are available.

Like the BDGs, the AAPS receives an automatic start signal approximately 30 seconds after sensing a loss of power to all EHVS buses, but does not automatically restore power. The AAPS and its output breaker can be controlled locally or remotely to energize the EHVS buses.

5.4.2 Equipment Qualification

Although the functions performed by the BDGs are nonsafety-related and not risk significant, the BDGs are designed to meet Seismic Category II requirements. The AAPS is not safety-related or risk significant, and therefore, Seismic Category III.

5.4.3 System Response to an Extended Loss of Alternating Current Power

The BDGs and the AAPS receive automatic start signals 30 seconds after the initiation of the event. However, because they are not seismically robust, the BDGs and the AAPS are considered unavailable during the ELAP per baseline coping capability assumption number 2 of Section 4.2.

5.5 Normal Direct Current Power System

5.5.1 System Design

The normal DC power system (EDNS) uses batteries, battery chargers, and inverters to supply power to nonsafety-related plant DC and AC loads. These loads include[6]:

- the module control system
- the plant control system (PCS)
- electrical distribution control power
- control rod latching power

The EDNS consists of 14 subsystems with the configuration and equipment requirements for each determined by the loads that it supplies. Although multiple subsystems only provide power for DC loads, the RXB and CRB subsystems supply both DC and AC loads. As a result, they include inverters in addition to batteries and battery chargers.

Normally, EDNS battery chargers receive power from the ELVS to meet the load demands and maintain the system's batteries charged. If ELVS power is interrupted or lost, the EDNS battery chargers cease to provide power and the batteries automatically provide the required power to the system loads without interruption. The EDNS batteries are sized to supply the continuous full load for a runtime period of no less than 40 minutes.

5.5.2 Equipment Qualification

The EDNS equipment is designed such that its failure due to seismic events does not degrade the operation of any safety-related system, but is not designed for continued operation following an SSE.

5.5.3 System Response to an Extended Loss of Alternating Current Power

Because the EDNS equipment is not seismically robust, the system is assumed to be unavailable for the implementation of the FLEX strategy. However, if the EDNS batteries in the RXB and CRB subsystems were to continue to provide power, they would contribute to conditions required to be addressed by the plant design. Specifically, continued operation of the RXB EDNS subsystem allows the control rods to remain withdrawn until a valid reactor trip signal is generated as described in Section 5.7.1, and operation of both subsystems results in heat addition to their surroundings. Therefore,

6 This list does not represent a complete list of EDNS loads.

for analysis purposes it is conservative to assume EDNS remains functional for the duration of the EDNS batteries' capacities during an ELAP, while not crediting any of the system's functionality for coping.

5.6 Highly Reliable Direct Current Power System

5.6.1 System Design

The EDSS is the source of 125 VDC power to plant loads such as[7]:

- the module protection system (MPS)
- the plant protection system (PPS)
- the neutron monitoring system
- the safety display and indication system (SDIS)
- MCR emergency lighting

The EDSS design consists of two separate and independent portions. One portion, referred to as the EDSS-C, serves plant common loads (i.e., loads with functions not specific to any single NPM). This includes loads such as the PPS and control room heating, ventilation, and air conditioning (HVAC) air duct radiation monitors

The other portion, referred to as the EDSS-MS, consists of 12 separate and independent DC electrical power supply systems, one for each of the 12 NPMs. For a given NPM, an EDSS-MS provides electrical power for the MPS and other safety-related loads associated with that NPM, including the neutron monitoring system and PAM equipment for display and indication of NPM-specific safety parameters.

The EDSS-C portion of the system contains two separate divisions. As seen in Figure 5-1, both divisions of EDSS-C contain two batteries, each designed to supply the required loads for a minimum of 72 hours. Thus, if one battery is not functional or is taken out of service for maintenance, the other redundant battery is capable of supplying the required power for at least 72 hours.

7 This list does not represent a complete list of EDSS loads.

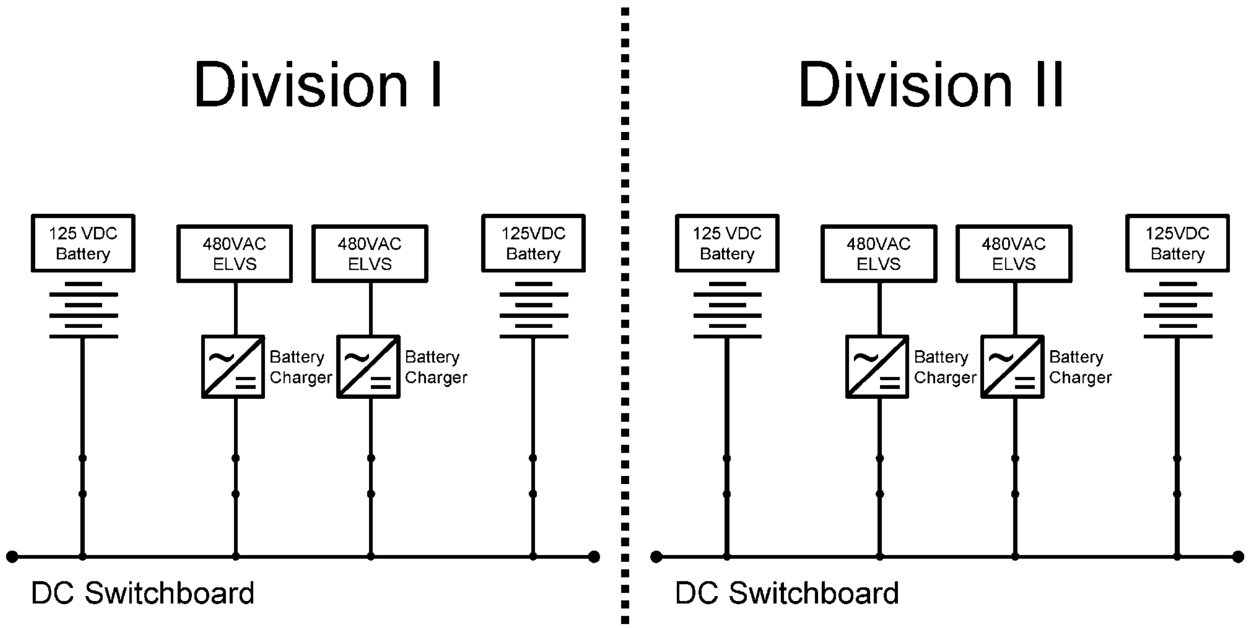

Figure 5-1 Highly reliable direct current power system-common simplified overview

Each of the 12 EDSS-MS portions of the system also contains two separate divisions. Each of these divisions are further divided into two channels per division, with Division I composed of Channels A and C and Division II composed of Channels B and D. Channels A and D, one channel from each division, both contain two batteries, each designed to supply required loads for a minimum of 24 hours. Channels B and C, again one channel from each division, both contain two batteries, each designed to supply required loads for a minimum of 72 hours. This results in redundancy similar to that found in the EDSS-C portion of the system, except that the EDSS-MS possesses this redundancy at the channel level rather than the division level. Thus, as seen in Figure 5-2, the EDSS-MS for each NPM contains four 24-hour minimum capacity batteries and four 72-hour minimum capacity batteries.

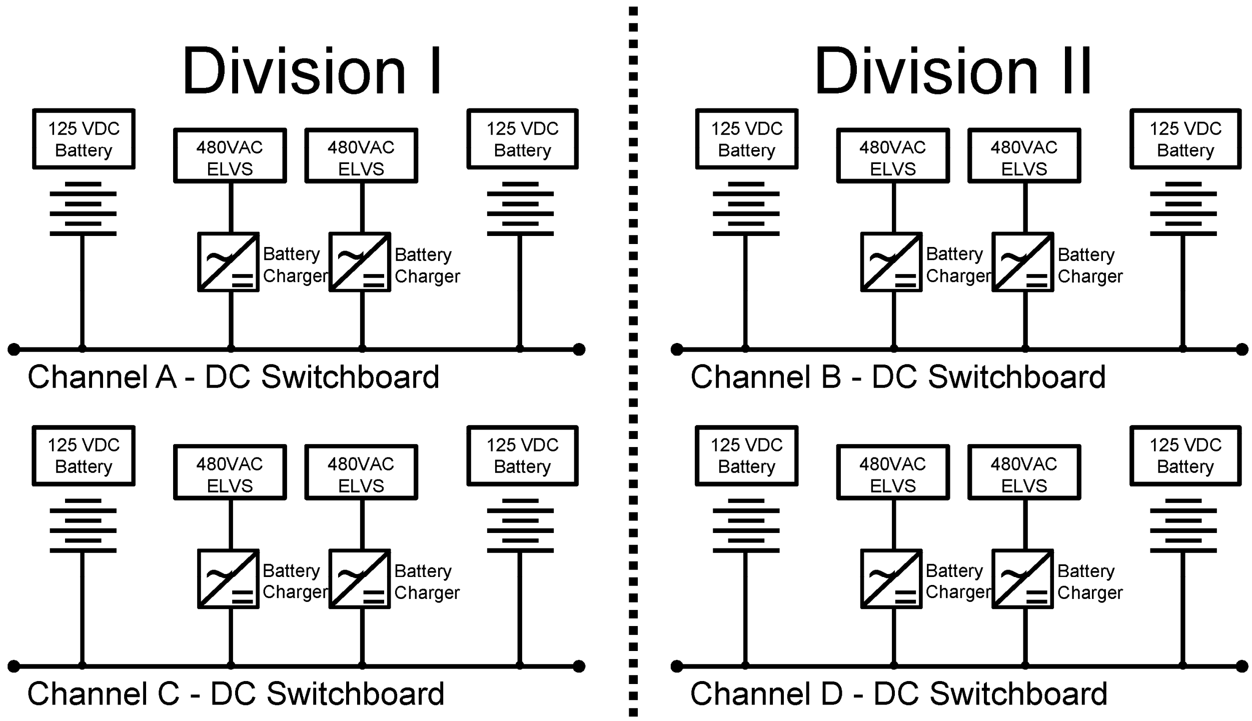

Figure 5-2 Highly reliable direct current power system-module-specific simplified overview

The EDSS design provides the same redundancy in battery chargers as that for the EDSS batteries. Specifically, each EDSS-C division and all four of the EDSS-MS channels have redundant battery chargers. Each battery charger is capable of supplying electrical power to its associated loads while simultaneously recharging its associated batteries from their design minimum charge state to 95 percent of full charge within 24 hours.

During normal operation, the EDSS battery chargers, which are powered from the ELVS, supply all EDSS system loads and maintain the batteries charged. Upon a loss of AC power, the EDSS batteries supply power to the system loads automatically and without interruption. The availability of AC power to the EDSS-MS and EDSS-C battery chargers is monitored by the MPS and the PPS, respectively. When a loss of AC power to the battery chargers is sensed, the EDSS batteries are placed into loss of AC power operation, which consists of two operating modes: emergency core cooling system (ECCS) hold mode or PAM-only mode.

The ECCS hold mode is an EDSS-MS mode of operation in which power continues to be supplied to hold the ECCS valves in the closed position until a valid ECCS actuation signal is received or for 24 hours from the initiation of ECCS hold mode operation. The MPS contains 24-hour timers used to terminate ECCS hold mode if an actuation has not occurred. This mode of operation includes providing power to the instrumentation and control equipment required to maintain the ECCS valves closed and is exclusive to the four EDSS-MS channels.

The PAM-only mode is an EDSS mode of operation in which power is supplied only to instrumentation and controls equipment used to track PAM variables. This mode of operation is exclusive to both divisions of EDSS-C and one channel from each division of EDSS-MS, Channels B and C. The batteries that support PAM-only mode operation are sized to ensure PAM is available for a minimum of 72 hours, but continue to provide power beyond 72 hours if their capacity is not exhausted.

5.6.2 Equipment Qualification

The EDSS-MS and EDSS-C structures, systems, and components are qualified to 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 EDSS-MS components are housed in battery rooms and switchgear rooms located on the 75-ft elevation and 86-ft elevation of the RXB, respectively. Similarly, all major EDSS-C components are housed in battery rooms and switchgear rooms located on the 50-ft elevation of the CRB. Conditions within these rooms are limited to mild environments and EDSS components housed within the rooms are qualified to Institute of Electrical and Electronics Engineers (IEEE) Std. 323-2003 (Reference 13.1.22), as endorsed with enhancements and exceptions in RG 1.209 (Reference 13.1.11).

The EDSS cabling located in a harsh environment in the RXB is qualified to IEEE Std. 323-1974 (Reference 13.1.23), as endorsed with modification by RG 1.89 (Reference 13.1.12).

5.6.3 System Response to an Extended Loss of Alternating Current Power

Per initial event assumptions number 3, number 7, and number 9 of Section 4.2.3, the EDSS batteries and the associated distribution system are assumed to survive the BDBEE and remain fully available during the ELAP.

The loss of the ELVS at the initiation of the event results in the transfer of the EDSS loads to the batteries. The MPS detects the loss of AC power to the EDSS battery chargers and places all four channels of EDSS-MS into ECCS hold mode operation. At the same time, the PPS places EDSS-C in the PAM-only mode of operation. After 24 hours, ECCS hold mode operation is terminated and EDSS-MS Channels B and C join EDSS-C in PAM-only mode operation.

While PAM-only mode operation is ensured for a minimum of 72 hours, the presence of 100 percent redundant batteries in all channels of EDSS-MS and both divisions of EDSS-C means that actual availability of power for PAM likely extends beyond 72 hours without operator action.

5.7 Module Protection System

5.7.1 System Design

The MPS is the safety-related monitoring and logic protection system responsible for evaluating parameters associated with the NPMs and commanding execution of safety-related functions when required. Each NPM has its own dedicated (i.e., not shared with other NPMs) MPS. The variables monitored for each module by MPS include[8]:

- pressurizer level
- RPV water level
- RCS pressure
- containment water level
- containment pressure
- containment isolation valve (CIV) position
- ECCS valve position
- core inlet and exit temperature
- neutron flux
- reactor trip breaker (RTB) position

The MPS provides information regarding these parameters to the control room operators through the SDIS and PCS.

As previously discussed, each NPM has four separate and independent EDSS-MS channels. Each of these channels provides electrical power to a corresponding separation group within MPS (e.g., EDSS-MS Channel A supplies power to MPS Separation Group A). The EDSS-MS also supplies power for the two divisions of the reactor trip system (RTS) and the two divisions of the engineered safety features actuation system (ESFAS).

The RTS and ESFAS are subsystems of each NPM's MPS, and command execution of safety-related functions such as:

- reactor trip
- containment isolation
- decay heat removal system (DHRS) actuation
- ECCS actuation
- de-energization of the pressurizer heaters
- isolation of demineralized water

Each NPM's safety-related reactor trip circuitry is part of its Class 1E MPS, which is powered from EDSS-MS. Included in each MPS are the safety-related circuit breakers

8 This list does not represent a complete list of variables monitored by MPS.

feeding the control rod drive mechanism (CRDM) coils. These circuit breakers are referred to as the RTBs.

During normal power operations, the RTS provides power to the RTBs to maintain the breakers closed. This allows the EDNS to supply power to the CRDM coils which, 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 EDNS 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, and containment isolation[9]. Initiation of these ESFs is the function of the MPS subsystem referred to as the ESFAS.

During normal power operation, the safety-related CIV required to be open for plant operation are maintained in this position by energizing their valve actuator solenoids. The power used to energize these solenoids is provided by EDSS-MS and is routed through the MPS. When a containment isolation signal is generated, the ESFAS de-energizes the valve actuator solenoids and allows 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 EDSS-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 EDSS-MS through the MPS and removal of this power from the solenoids upon an ECCS actuation signal allows the ECCS valves to open.

Each NPM contains two pressurizer heater bundles, each with a proportional heater group and a backup heater group used for pressure control. Electric power for operation of the pressurizer heaters is delivered from the ELVS through Class 1E breakers in the MPS. When a low pressurizer level is detected or the DHRS is actuated, the ESFAS de-energizes and opens the pressurizer heater breakers to ensure ELVS power is interrupted.

[9] This list does not represent a complete list of ESFs initiated by the ESFAS.

The chemical and volume control system (CVCS) contains safety-related demineralized water isolation valves. Similar to the CIVs, the demineralized water isolation valves require solenoids to remain energized to maintain the valves open. The power for these solenoids is provided from EDSS through the MPS. When required by plant conditions, the MPS interrupts power to the solenoids to close the valves.

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 EDSS.

5.7.2 Equipment Qualification

The safety-related MPS equipment and cabling are housed in 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. The RXB and the CRB (below the 120-ft elevation) are Seismic Category I, reinforced concrete structures.

The MPS is a Seismic Category I system, and is seismically qualified in accordance with IEEE Std. 344-2004, as endorsed and modified by RG 1.100.

The MPS equipment required to operate during and after a design basis accident is environmentally qualified in accordance with IEEE Std. 323-2003 (Reference 13.1.22) as endorsed by RG 1.209 (Reference 13.1.11) for mild environments and IEEE Std. 323-1974 (Reference 13.1.23) as endorsed by RG 1.89 (Reference 13.1.12) for harsh environments. Rack-mounted MPS equipment is located in an environmentally controlled area. However, the equipment is designed to accommodate abnormal conditions due to the loss of normal HVAC in the area for a minimum of 72 hours.

5.7.3 System Response to an Extended Loss of Alternating Current Power

Per baseline coping capability assumption number 1 of Section 4.2, the MPS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

For this event, the EDSS and the MPS remain energized. Per initial event assumption number 1 of Section 4.2.3, the ELVS de-energizes at event initiation and remains de-energized for the duration. The MPS monitors the voltage of the ELVS buses that provide power to the Channel B and Channel C EDSS battery chargers as a method of detecting a loss of all AC power. With the loss of voltage on the ELVS buses and following a short time delay (i.e., less than one minute), the MPS initiates a reactor trip, DHRS actuation, and containment isolation; and starts 24-hour timers for the ECCS hold mode of operation for EDSS-MS[10]. The effects of the 24-hour timers are discussed in Sections 5.6.1 and 5.13.3.

5.8 Plant Protection System

5.8.1 System Design

The PPS monitors and controls systems that are common to all reactor modules and are not specific to an individual NPM. The variables monitored and equipment actuated by the PPS have an augmented level of quality. The variables monitored by the PPS include[11]:

- the ELVS voltages for buses that supply EDSS-C battery chargers
- reactor pool (RP) and refueling pool (RFP) level
- SFP level

The PPS provides information regarding these parameters to the control room operators through the SDIS, the PCS, or both.

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

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

[10] Because the loss of AC power at event initiation results in a loss of feedwater to the steam generators, the MPS may generate a reactor trip and DHRS actuation prior to the loss of AC power signal time delay expires.

[11] This list does not represent a complete list of variables monitored by PPS.

5.8.2 Equipment Qualification

The PPS is classified as Seismic Category I and is seismically qualified in accordance with IEEE Std 344-2004, as endorsed and modified by Regulatory Guide 1.100. The PPS equipment is housed in the Seismic Category I portion of the CRB.

The PPS equipment is located in a mild environment and is qualified in accordance with the requirements of IEEE Std 323-2003 (Reference 13.1.22), as endorsed and modified by Regulatory Guide 1.209 (Reference 13.1.11).

5.8.3 System Response to an Extended Loss of Alternating Current Power

Per baseline coping capability assumption number 1 of Section 4.2, the PPS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

For this event, EDSS-C and the PPS remain energized. Per initial event assumption number 1 of Section 4.2.3, 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 EDSS-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.

5.9 Safety Display and Indication System

5.9.1 System Design

The SDIS is a nonsafety-related system that provides the design function of accident monitoring. The primary purpose of the SDIS is to display accurate, complete, and timely information provided by the MPS and the PPS regarding:

- parameter values
- logic status
- equipment status
- actuation device status

The information displayed is provided to the SDIS from communications containing information from each separation group and each division of the ESFAS, RTS, and PPS. This information contains the data necessary for the operators to ensure the NPMs are in a safe condition following an event.

Electrical power to the SDIS is provided from two separate and independent divisions of EDSS-C, the same electrical power source as for the PPS. In the MCR, the SDIS provides two divisions of monitors for each NPM and the PPS, with both divisions of MPS and PPS data displayed on each division of the monitors. The variables include[12]:

- neutron flux
- core inlet and 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

5.9.2 Equipment Qualification

The SDIS equipment is housed in the concrete, Seismic Category I portions of the CRB. The SDIS is qualified to Seismic Category I requirements as required by IEEE Std. 497-2002 for accident monitoring instrumentation.

SDIS hubs are located at the 50-ft level of the CRB in the PPS rooms. The display interface modules and display panels are located at the 76 ft - 6 in. level of the CRB in the MCR. The SDIS is qualified in accordance with IEEE Std. 323-2003 (Reference 13.1.22), to ensure system operation for a minimum of 72 hours following the loss of HVAC.

5.9.3 System Response to an Extended Loss of Alternating Current Power

Per baseline coping capability assumption number 1 of Section 4.2, the SDIS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

For this event, EDSS-C and the SDIS remain energized for a minimum of 72 hours. The SDIS continues to display the data necessary for the control room operators to ensure the actuation of the systems required to maintain core cooling and containment integrity. Also displayed are the parameters necessary for the control room operators to verify the success of the strategy for maintaining core cooling, containment integrity, and spent fuel cooling.

[12] This list does not represent a complete list of PAM variables displayed by the SDIS.

5.10 Containment System

5.10.1 System Design

The containment system (CNTS) is part of the NPM and is the containment for the RCS. The CNTS components include multiple support structures, the CNV, the CIVs, and CNTS instruments.

The CNV is a metal containment pressure vessel forming a barrier to prevent release of radioactivity and radiological contaminants. The RCS, the control rod drive system, select DHRS components, 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, which allows the CNTS design to provide the function of containment heat removal. The CNV is safety-related and its significant design parameters are listed in Table 5-1.

Table 5-1 Significant containment vessel parameters

Parameter	Value
Internal design pressure	1,000 psia
External design pressure	Atmospheric pressure plus UHS static head pressure
Design temperature	550 degrees F
Internal operating pressure	0.1 psia
External operating pressure	Atmospheric pressure plus UHS static head pressure
Operating temperature	100 degrees F

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

For lines that penetrate a CNV boundary and are either part of the reactor coolant pressure boundary or are connected directly to the containment atmosphere, two in-series safety-related PSCIVs are provided. The two PSCIVs for a given line are located in the same valve body, which is welded directly to the CNV penetration to minimize the distance the valves are from the CNV.

One SSCIV is provided per line[13] for the main steam lines, main steam bypass 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.

[13] The feedwater lines also include a safety-related check valve that is provided for DHRS inventory preservation rather than containment isolation.

Each CIV has a nitrogen-filled accumulator 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 nitrogen gas pressure to open the valve. The power for these solenoids is provided by EDSS through the MPS.

When the ESFAS generates a containment isolation signal, power is removed from the CIV actuator solenoids. This aligns the high-pressure hydraulic fluid holding the CIVs open to a vent path and the stored energy in the compressed nitrogen gas 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 an NPM has been 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 has been sufficiently cooled, the CNV is partially filled with borated water. This 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 are opened. This allows passive communication between containment and the RPV, both of which are filled with borated water to a level near the pressurizer baffle plate. When the RCS has been sufficiently cooled, all CIVs are closed. This establishes a passively safe condition and eliminates the need for any control connections to the CNV.

5.10.2 Equipment Qualification

The CNTS, including the CIVs, is designed and constructed to Seismic Category I requirements and is located in the Seismic Category I RXB, which provides protection from non-seismic natural phenomena, such as tornados, storms, and floods.

The CIVs are designed to withstand primary system internal pressure and temperature of 2,100 psia and 650 degrees F.

The CNV components and penetrations (piping, electrical, and instrumentation and controls) are designed for harsh environment conditions. The CIVs are part of the NuScale Power Plant Environmental Qualification Program. Under this program the valve design is demonstrated to be capable of performing its safety function under the prescribed environmental conditions.

5.10.3 System Response to an Extended Loss of Alternating Current Power

Per baseline coping capability assumption number 1 of Section 4.2, the CNTS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

As discussed in Section 5.7.1, 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 CIVs to close.

5.11 Ultimate Heat Sink System

5.11.1 System Design

The NuScale Power Plant safety-related UHS is composed of a large pool complex where the NPMs 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. Together, these pools contain more than 6.5 million gallons[14] of water for use as the UHS. The UHS system also includes the assured makeup line and the level instrumentation associated with the pools.

The NPMs are located in the RP during power operations and are moved to the RFP for refueling and maintenance operations. The operating NPMs are partially immersed in the RP, and as shown in Figure 5-3, the DHRS passive condensers are submerged in the RP.

[14] This volume represents total of the pool volumes minus the volume displaced by the 12 NPMs in the RP.

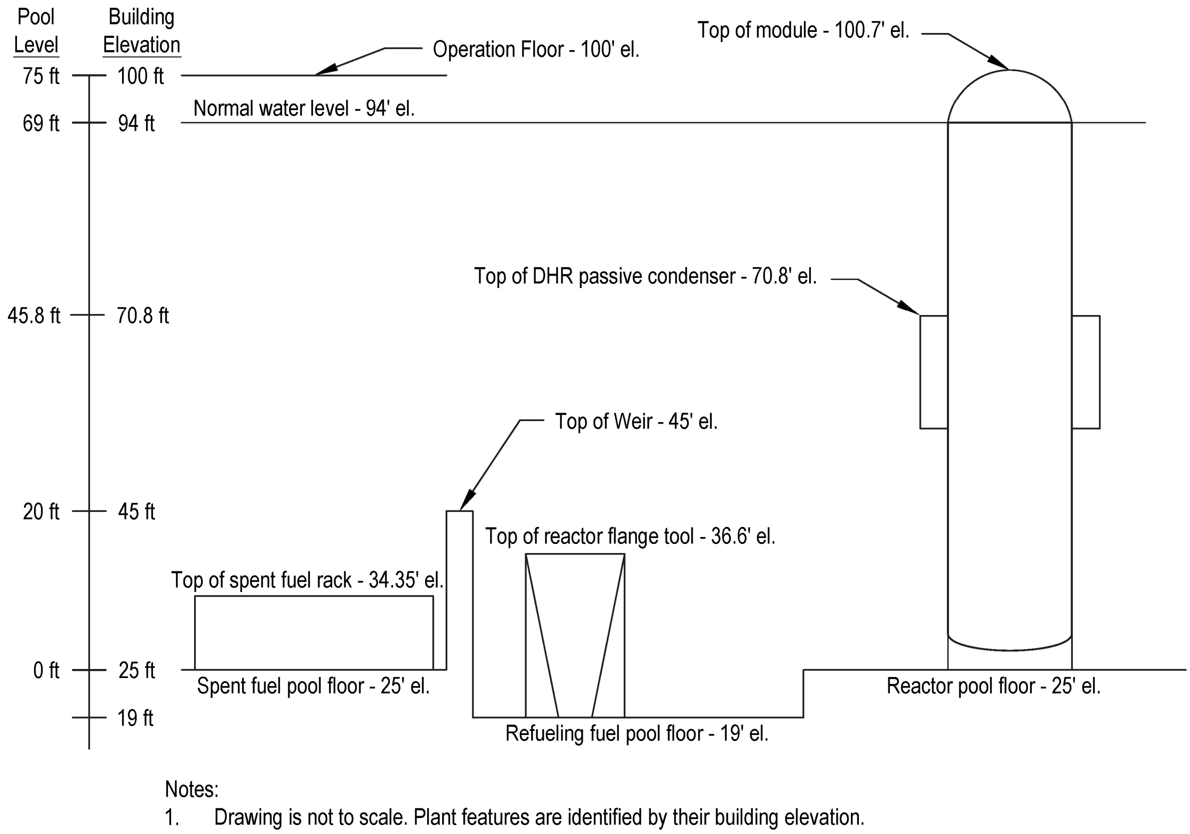

Figure 5-3 Ultimate heat sink level and Reactor Building elevation guide

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 in the UHS is maintained at 94 ft during normal operations through an interface with the spent fuel pool cooling system (SFPCS) and temperature is maintained at an average of 100 degrees F by the combined operation of the reactor pool cooling system (RPCS) and the SFPCS.

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 all three pool levels. Power is provided to the detectors from EDSS-C through the PPS and pool level is displayed in the MCR on the SDIS displays. Each level instrument includes a backup battery power supply independent of the site distribution network and local readout capability located in an operator-accessible area away from the pool area.

The UHS system design includes an assured makeup line to the SFP. The line is furnished with standard fire protection connectors external to the RXB that facilitate hookup of emergency water sources. The assured makeup line is designed to allow water to be gravity fed to the SFP at a minimum rate of 100 gpm. Adding the water directly to the SFP ensures that the stored fuel remains covered, and when level is above the weir, serves to add inventory to the other pools of the UHS.

5.11.2 Equipment Qualification

The UHS pool walls and pool liner are designed to Seismic Category I requirements and are completely contained within the Seismic Category I RXB.

The 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 13.1.14).

The assured makeup line is designed to Seismic Category I requirements 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).

5.11.3 System Response to an Extended Loss of AC Power

Initial event assumption number 5 in Section 4.2.3 states:

> *Normal access to the ultimate heat sink is lost, but the water inventory in the UHS remains available and robust piping connecting the UHS to plant systems remains intact. The motive force for UHS flow, i.e., service water or circulating water pumps, is assumed to be lost with no prospect for recovery.*

The LUHS condition is assumed to occur due to the BDBEE and ELAP. Because the NuScale Power Plant design does not require electrical power or pumps for the UHS to perform the functions required for decay heat removal from NPMs and spent fuel in the SFP, and given that the water inventory in the UHS remains available per the boundary conditions established in NEI 12-06 (Reference 13.1.16), the potential for this LUHS event is not plausible for the NuScale plant design.

In Attachment 3, Requirements for Mitigation Strategies for Beyond-Design-Basis External Events at COL Holder Reactor Sites (Vogtle Units 3 and 4), of NRC Order EA-12-049 (Reference 13.1.4), the NRC recognized that including a passive, safety-related UHS that does not rely on access to any external water source in the AP1000 design made the LUHS not plausible. The NRC altered the wording to "loss of normal access to the normal heat sink" in the Order to address this possibility. The proposed rule 10 CFR 50.155 and the associated draft guidance (Reference 13.1.3) also uses this phrase when referring to plants with passive UHS design.

The UHS is the only plant system heat sink credited for coping with an ELAP.

Regarding the LUHS as described in NEI 12-06, per baseline coping capability assumption number 1 of Section 4.2, the UHS system is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

The ELAP results in the transfer of heat from the DHRS and ECCS to the UHS as described in Sections 5.12 and 5.13, respectively. This effect, combined with the loss of normal heat removal from the UHS, results in heating the pool water. For this event, the UHS reaches boiling after {{ }}2(a),(c), at which point UHS level begins to lower due to evaporation and boil off. Without the implementation of a FLEX strategy, UHS level decreases as shown in Table 5-2.

Table 5-2 Ultimate heat sink heat up and boil off – 12 NuScale Power Modules initially operating

Location	Pool Level	Cumulative Time
Pool heat up to boiling	{{	
Pool boil off to top of DHRS passive condenser		
Pool boil off to top of weir in SFP weir wall		
Pool boil off to top of spent fuel in SFP		}}2(a),(c)

The analysis that determined the levels and cumulative time in Table 5-2 assumed that all of the water vapor from evaporation and boiloff exits the Reactor Building and does not re-condense into the UHS. Given the change in UHS volume during the ELAP, the average boil off rate can be approximated at various times throughout the event. As the decay heat rate of the previously operating NPMs lowers, so too does the UHS boil off rate, which, as seen in Table 5-3, is below the minimum gravity feed capacity (100 gpm) of the UHS assured makeup line.

Table 5-3 Approximate ultimate heat sink boil off rate during an extended loss of alternating current power

Time After Event Initiation (days)	UHS Volume Remaining (ft^3)	Change in UHS Volume (gallons)	Change in Time (minutes)	Average Boil Off Rate During Period (gpm)
5.57	{{			
15.53				
28.97				
39.09				
49.95				}}2(a),(c)

5.12 Decay Heat Removal System

5.12.1 System Design

Each NPM has a safety-related DHRS designed to passively remove decay heat in order to establish safe shutdown conditions within {{ }}2(a),(c) of a reactor trip without onsite or offsite power available. The DHRS consists of two independent and redundant trains, and each train has the capability to provide sufficient heat removal to satisfy the safety function.

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 on top of the NPM between the 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 5.7.1, the DHRS is normally maintained in standby with all four DHRS actuation valves held closed by maintaining their associated actuation solenoids energized. This 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 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.

5.12.2 Equipment Qualification

All components of the safety-related DHRS are located within the Seismic Category I RXB, which provides 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.

5.12.3 System Response to an Extended Loss of AC Power

Per baseline coping capability assumption number 1 of Section 4.2, the DHRS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

Depending on the operating condition of a given NPM when the loss of all AC power occurs, the DHRS actuates as a result of the conditions in the NPM reaching an initiation setpoint or as a result of the MPS detecting the loss of AC power as described in Section 5.7.3.

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.

5.13 Emergency Core Cooling System

5.13.1 System Design

Each NPM has a safety-related ECCS designed to provide adequate core cooling by maintaining the core covered with water during all design basis events in which the system is actuated. The NuScale Power Plant ECCS is unique in that it does not include or require a source of water for injection. The system functions by releasing coolant to the CNV to be cooled by the UHS and returning the coolant to the RPV to remove heat from the core. This is done without the use of any 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 five main valves and their associated hydraulic lines and actuator assemblies. The main valves are welded to nozzles on the RPV and are located inside the CNV. There are three upper valves and two lower valves, the reactor vent valves (RVVs) and the reactor recirculation valves, respectively. The actuator assemblies are welded directly to CNV nozzles in the vicinity of their associated main valve and are submerged in the RP.

Each main valve actuator assembly includes two pilot valves: the trip valve and the reset valve[15]. 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 EDSS through the MPS.

Each ECCS main valve includes an inadvertent actuation block (IAB) feature designed to reduce the frequency of inadvertent operation (opening) of the main valve during power operations. The IAB is located in the path from the main valve control chamber to the trip and reset pilot valves. The IAB consists of a block valve with a spring-loaded disc that functions to block venting of the main valve control chamber when RPV pressure is above a predetermined pressure threshold. When differential pressure across the block valve lowers to below the pressure threshold, the spring retracts the block valve to open the control chamber vent path.

During normal power module operation, all five main valves are closed with their trip valve actuator solenoids energized and closed. When the ECCS is actuated, the ESFAS removes ECCS actuator solenoid power and the trip valves are opened by spring force. With the RCS at normal operating pressure, the IAB prevents the ECCS main valves from opening. After RCS to CNV differential pressure has lowered due to other factors (e.g., DHRS operation or a loss-of-coolant accident), the IAB is released and all five ECCS main valves immediately open.

With all five 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 reactor recirculation valve 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[16], 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 reactor recirculation valves 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.

---

15 One of the three RVVs has an additional trip valve and actuator solenoid to allow either MPS division to trip and open the valve.

16 While ECCS never achieves true steady-state operation, coolant levels in the CNV and RPV do achieve relative stability.

5.13.2 Equipment Qualification

The ECCS valves, hydraulic lines, and actuators are part of the reactor coolant pressure boundary and all components are Seismic Category I. All components of the ECCS are located within the Seismic Category I RXB.

The ECCS main valves are designed to withstand internal pressure of 2,100 psia, and external pressure as low as 0 psia and as high as 1,000 psia. The design temperature of the main valves is 650 degrees F.

The pressure-retaining portions of the ECCS hydraulic actuator systems are designed to withstand internal pressure of 2,100 psia. The design temperature for pressure-retaining portions of the actuators is 650 degrees F.

The ECCS pilot valves are subject to the RP conditions externally and reactor pressure internally. The solenoid operators and position indication are provided with a bolted protective cover designed for submergence to ensure a dry environment for the electronics. The design temperature for the electronics is 250 degrees F. The external design pressure is 50 psia.

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.

5.13.3 System Response to an Extended Loss of Alternating Current Power

Per baseline coping capability assumption number 1 of Section 4.2, the ECCS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

As described in Section 5.7.3, 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 were to require an ECCS actuation within this 24-hour period, the ECCS valves would open as designed, but because this event does not include a loss-of-coolant accident, the ECCS valves remain closed during this period.

After 24 hours, the 24-hour timers expire and de-energize the ESFAS, which de-energizes the ECCS trip valve solenoids. Because the DHRS has been in operation for the previous 24 hours, RCS pressure is below the IAB pressure and the ECCS main valves open to establish system operation.

5.14 Chemical and Volume Control System

5.14.1 System Design

The CVCS is an NPM-specific system used to maintain the required coolant inventory in the RCS during all modes of operation [17] through use of makeup and letdown components[18]. The CVCS makeup pumps are supplied with demineralized water from the demineralized water system, borated water from the boron addition system, or a proportioned mixture of the two through the utilization of a combining valve. If an addition to the RCS is required either for inventory management or reactivity adjustment, the CVCS makeup pumps are used. Reactor coolant inventory is reduced by opening the isolation valves and control valve in the letdown line to discharge fluid to the liquid radwaste system. Because they penetrate the reactor pressure boundary, both the CVCS makeup flow path and the letdown flow path contain two CIVs in series.

The only safety-related function associated with the CVCS is to limit the potential amount and rate of a reactivity increase due to an inadvertent boron dilution event. The demineralized water supply to the CVCS makeup pumps is provided with two series, safety-related isolation valves that isolate on high subcritical multiplication, low RCS flow, or any RTS actuation. These valves require pressurized instrument air to open and fail closed on a loss of air. The instrument air is admitted to the valve operator through a solenoid-operated valve when its solenoid is energized. The electric power to energize the solenoids is supplied by EDSS through the MPS. When required by plant conditions, the MPS removes power from the valve actuator solenoids that remove air pressure and closes the demineralized water isolation valves.

5.14.2 Equipment Qualification

The demineralized water isolation valves are Seismic Category I components and are located within the Seismic Category I RXB.

5.14.3 System Response to an Extended Loss of Alternating Current Power

Per baseline coping capability assumption number 1 of Section 4.2, the demineralized water isolation valves in the CVCS are assumed to survive the BDBEE and remain fully

[17] In this context, “modes of operation” is not referring to Technical Specification Modes, but rather conditions when the NPM is at power, in hot shutdown, or safe shutdown. The CVCS is intentionally disabled during Technical Specification Modes 4 and 5.

[18] The CVCS performs additional functions, such as pressurizer spray flow for RCS pressure control, but these functions and their associated components are not related to the safety-related function, and are not discussed here.

available during the period when the system functions necessary for FLEX strategy implementation are required.

Although the CVCS makeup pumps are de-energized at event initiation due to the loss of the ELVS, the demineralized water isolation valves still function to prevent a condition in which an unacceptable inventory of demineralized water could be introduced into the RCS in the time period before the CVCS containment isolation valves close when the containment isolation occurs as described in Section 5.10.3.

5.15 Reactor Pool Cooling System

5.15.1 System Design

The RPCS consists of three trains, each containing one pump and one heat exchanger, to transfer heat from the RP to the SCWS to maintain the RP at or below 100 degrees F. The RPCS pumps are powered by the ELVS.

5.15.2 Equipment Qualification

The RPCS pumps and heat exchangers are Seismic Category III components.

5.15.3 System Response to an Extended Loss of Alternating Current Power

The loss of the ELVS at initiation of the ELAP makes the RPCS unavailable for the duration of the event.

5.16 Spent Fuel Pool Cooling System

5.16.1 System Design

The SFPCS consists of two trains, each containing one pump and one heat exchanger, to transfer heat from the SFP to the SCWS to maintain the SFP at or below 100 degrees F and to add inventory when necessary. The SFPCS pumps are powered by the ELVS.

5.16.2 Equipment Qualification

The SFPCS pumps and heat exchangers are Seismic Category III components.

5.16.3 System Response to an Extended Loss of Alternating Current Power

The loss of the ELVS at initiation of the ELAP makes the SFPCS unavailable for the duration of the event.

5.17 Reactor Building Heating, Ventilation, and Air Conditioning System

5.17.1 System Design

The Reactor Building 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. These spaces include the following[19]:

- RP area
- SFP and RFP area
- battery, battery charger, and input/output (I/O) rooms

During normal operation, power for operation is supplied by the ELVS. If the ELVS is unavailable, backup power from the BDGs can be used to power the AHUs associated with the EDSS and MPS equipment rooms (battery, battery charger, and I/O rooms), with cooling provided by air-cooled direct expansion cooling coils.

In the event of an SBO, (i.e., no power is available for the exhaust systems) 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 allows a passive HEPA-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.

5.17.2 Equipment Qualification

The majority of the Reactor Building HVAC system components, including the AHUs, are designed to Seismic Category III. The SSC of the RBVS whose structural failure could affect the operability of safety-related SSC are designed as Seismic Category II.

5.17.3 System Response to an Extended Loss of Alternating Current Power

The loss of the ELVS at initiation of the ELAP 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.

[19] This list does not represent a complete list of spaces serviced by the RBVS.

5.18 Control Room Heating, Ventilation, and Air Conditioning System

5.18.1 System Design

The CRVS consists of the AHUs, filter units, flow dampers, cooling coils, heaters, duct work, and fans necessary to maintain a suitable environment in the entire CRB for equipment operation and worker habitability.

The CRVS is designed to maintain the areas served by the system at a positive pressure with respect to adjacent areas during normal plant conditions and during postulated accidents when power is available from offsite or the BPSS. The system is designed with the capability to remove radioactive contamination from the incoming outside air by charcoal filtration during a postulated accident. If radiation levels are above the design limit of the charcoal filters, or if power is not available, the CRVS provides isolation of the CRE from the surrounding areas and outside environment by isolation dampers. The dampers have spring return actuators that act to close the dampers and isolate the CRE when power is removed. The CRVS components used to isolate the CRE receive power from and are controlled by the PPS.

The CRVS components used to provide environment controls such as cooling and heating are powered by the ELVS. The CRVS components that are relied on during a postulated radiological event to filter outside air and pressurize the control room and technical support center areas are provided with backup power from the backup power supply system BDGs. These components include the outside air charcoal filtration unit, the outside air bubble tight isolation dampers, the main supply AHUs, and associated components.

5.18.2 Equipment Qualification

CRVS SSC whose structural failure could affect the operability of safety-related SSC are designed as Seismic Category II. All other CRVS equipment, except the CRE isolation dampers, is classified as Seismic Category III.

The CRE isolation dampers are designed as Seismic Category I since they support establishing and maintaining the CRE. The fire and smoke dampers serving the control room are also designed as Seismic Category I since they provide protection of SSC necessary to attain or maintain safe shutdown from the adverse effects of fires.

5.18.3 System Response to an Extended Loss of Alternating Current Power

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

Per baseline coping capability assumption number 1 of Section 4.2, the CRE isolation dampers in the CRVS are assumed to survive the BDBEE and remain fully available for isolation of the CRE.

As described in Section 5.8.3, the PPS monitors for and detects the loss of AC power at event initiation. As a result, the PPS removes power from the CRE isolation dampers causing them to close and isolate the CRE.

5.19 Control Room Habitability System

5.19.1 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 high pressure air compressor (HPAC), 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 in order to provide reliable, clean breathing air for up to 20 plant personnel in the CRE, including 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 three separate rooms on the 50 ft - 0 in. elevation of the CRB.

The CRHS high pressure air compressor charges the air bottles to a working pressure of 3600 psig. The HPAC is sized to recharge 25 percent of the installed inventory in 24 hours and is located on the 63 ft - 3 in. elevation of the CRB.

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 EDSS-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 EDSS-C through the PPS.

The CRHS is automatically actuated by high radiation levels in the CRVS ducting or a loss of AC power. 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 places the CRHS in service and allows air to flow from the air bottles into the CRE.

5.19.2 Equipment Qualification

The CRHS components are located below grade on the Seismic Category I portions of the 50 ft - 0 in. elevation and 63 ft - 3 in. elevation of the CRB.

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

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

5.19.3 System Response to an Extended Loss of Alternating Current Power

Per baseline coping capability assumption number 1 of Section 4.2, the CRHS is assumed to survive the BDBEE and remain fully available during the period when the system functions necessary for FLEX strategy implementation are required.

As described in Section 5.8.3, 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 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.

5.20 Reactor Building Crane System

5.20.1 System Design

The RBC system provides the structural support and mobility needed to lift and move loads in the RXB, including the NPMs, to support normal operations, maintenance, receipt of new equipment, and refueling activities. The RBC system includes the following:

- RBC
- auxiliary hoists
- module lifting adapter (MLA)
- wet hoist
- below-the-hook lifting devices necessary to support and move loads
- instrumentation and controls necessary to support and move loads

The RBC includes a bridge, trolley, and hoist. The RBC rails allow the bridge to travel over the RP, RFP, and SFP[20] areas of the UHS, as well as the drydock area. The RBC is an 850 ton-capacity crane designed to remain in place, but not operate, during an SSE.

If seismic activity occurs, an electrical earthquake seismic switch shuts off all power to the RBC, the auxiliary hoists, the MLA, and the wet hoist. As a result, the trolley, bridge, and hoist stop and their associated brakes set to stop motion in all axes. The RBC is designed with redundant holding brakes. If one set of brakes fails to engage during an event in which the holding brakes are intended to be engaged, 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 an NPM in transition continues to be suspended by the RBC during and after a seismic event.

The RBC system 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 power. All 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

5.20.2 Equipment Qualification

The RBC, including the auxiliary hoists, MLA, and wet hoist, is designed as a single-failure-proof crane in accordance with the requirements of NUREG-0554 (Reference 13.1.13) and ASME NOG-1 (Reference 13.1.24) for Type I cranes. The MLA is designed as a single-failure-proof lifting device in accordance with the requirements of ANSI N14.6 (Reference 13.1.25).

The RBC and MLA are Seismic Category I equipment. The RBC system is located within the Seismic Category I RXB.

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

The brakes used to maintain the RBC load suspended are designed to withstand the environmental conditions that exist in the RXB during an ELAP.

5.20.3 System Response to an Extended Loss of AC Power

Per baseline coping capability assumption number 1 of Section 4.2, the RBC system is assumed to survive the BDBEE and remain capable of maintaining a design capacity load suspended for the duration of the ELAP.

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.

5.21 Communications System

The communications system will include at least one on-site and one off-site communications system capable of remaining functional during an ELAP 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 COL applicant.

6.0 Safety Functions during an Extended Loss of Alternating Current Power

An analysis of the NuScale Power Plant integrated response to an ELAP allows the baseline coping capability to be determined, which is then used for the determination of a FLEX strategy. As discussed in Section 3.3, NEI 12-06 (Reference 13.1.16) states an analysis for determining the specific sequence of events during an ELAP is not necessary for shutdown conditions in part "due to the small fraction of the operating cycle that is spent in an outage condition."

While the operating cycle for an NPM is comparable with existing designs, a NuScale Power Plant, because it includes 12 NPMs, spends a greater percentage of time with an NPM undergoing refueling. As such, in addition to analyzing the integrated plant response and key safety functions for an ELAP with all NPMs initially operating, the effects of an ELAP on the key safety functions for an NPM transitioning to refueling and for an NPM being refueled are specifically analyzed.

6.1 Integrated Plant Response

The following description of the overall plant response is based on the descriptions of the individual system responses provided in Section 5.0 and is intended to facilitate the discussions regarding the key safety functions in this section. The response for a single NPM is provided, but represents the response of all NPMs operating at 100 percent power when the BDBEE and ELAP occur.

Per the Section 4.2.3 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 not provided from station batteries by the inverters. For the NuScale Power Plant design, the initiating event results in the loss of all power, except for that power provided by the EDSS and the EDNS.

At event initiation, the MPS and the PPS detect the loss of AC power. The MPS starts the time 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 rapidly rise.

Approximately one minute into the ELAP, the reactor is tripped, the DHRS is actuated, all CIVs are closed, and the ECCS hold mode 24-hour timers have started.

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 NPM 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 below the ECCS inadvertent actuation block setpoint. At T=24 hours, the ECCS hold mode timers expire and the ESFAS de-energizes. This results in all five ECCS main valves opening and establishes passive decay heat removal via the ECCS.

The opening of the ECCS valves causes an immediate 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 ELAP have been completed or are passively occurring. These conditions are established without operator action and their maintenance requires neither AC power, nor DC power, nor operator action.

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 NPMs. Without operator action, pool temperature continues to rise and after more than {{ }}2(a),(c), the pool begins to boil. While boiling, the UHS continues to provide adequate cooling to the spent fuel in the SFP and removes sufficient heat from the NPMs to maintain containment and core cooling.

6.2 Core Cooling

Adequate core cooling is provided by plant safety-related systems for more than {{ }}2(a),(c) following initiation of an ELAP. The systems begin to operate passively without operator action and require no electrical power for initial or continued operation.

Specifically, the DHRS provides passive decay heat removal for the first 24 hours and the ECCS provides cooling for the remainder of the ELAP. Both the DHRS and the ECCS are supported by the passive functionality of the UHS to perform their safety-related functions. Because sufficient UHS inventory is present, adequate core cooling is ensured for more than {{ }}2(a),(c).

6.2.1 Reactor Coolant System Inventory

The design of the NPM 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 an ELAP.

During normal operation, RCS inventory is sufficient for ECCS operation and plant cooldown. During an ELAP, this inventory is preserved by the containment isolation that occurs within the first minute of the event.

However, as discussed in Section 4.2.5 of this report, NEI 12-06 (Reference 13.1.16) deterministically prescribes that RCS inventory losses during an ELAP are significant enough to require a strategy for inventory addition. Specifically regarding RCS inventory addition, the following two statements are made in NEI 12-06:

"Procedurally-directed actions can significantly extend the time to core uncovery in PWRs[21]. However, RCS makeup capability is assumed to be required at some point in the extended loss of ac power condition for inventory and reactivity control."

and

"Extended coping without RCS makeup is not possible without minimal RCS leakage."

Five sources of expected PWR and BWR coolant inventory loss are listed in NEI 12-06, Section 3.2.1.5 (Reference 13.1.16). Because the NuScale Power Plant design has unique aspects that differ from traditional PWR and BWR designs, each of the five listed sources was evaluated for applicability. The following list includes the NEI 12-06 expected leakage source followed by an explanation of why the listed source does not require an inventory addition strategy for the NuScale Power Plant design.

1. Normal system leakage

   NEI 12-06 does not provide a definition for what constitutes "normal system leakage," so all potential leakage paths that do not occur as the result of a failure (i.e., not "abnormal system leakage") were considered.

   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. As described in Sections 5.10 and 5.14, the CVCS is automatically isolated by the closure of safety-related CIVs and eliminates this potential leakage pathway outside of containment within the first minute of the event.

   Per technical specifications, an NPM may operate at full power with a certain allowable amount of primary-to-secondary leakage in each SG. As described in Sections 5.10 and 5.12 of this report, the MSIVs and FWIVs close within the first minute of the event to isolate this potential leakage path.

   All other potential leakage paths that could be considered "normal system leakage" would result in leakage into the CNV. This is important because, as described in Section 5.13 of this report, the CNV is part of the normal ECCS flow path and function. Thus, any leakage from these sources would be preserved, as designed, for ECCS operation and would not result in any need for inventory addition.

[21] Pressurized water reactor (PWR)

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

   As described in Sections 5.10 and 5.14, the CVCS is automatically isolated by the closure of safety-related CIVs and eliminates this potential leakage pathway outside of containment within the first minute of the event.

3. Losses due to reactor coolant pump seal leakage (rate is dependent on the RCP[22] seal design)

   The NuScale Power Plant design does not include RCPs or any similar component for which seal leakage would reduce inventory.

4. Losses due to BWR recirculation pump seal leakage

   The NuScale Power Plant design does not include recirculation pumps or any similar component for which seal leakage would reduce inventory.

5. BWR inventory loss due to operation of steam-driven systems, SRV[23] cycling, and RPV depressurization.

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

   The SRV cycling in a BWR is similar to reactor safety valve cycling for an NPM. However, when a reactor safety valve cycles, the RCS inventory exiting through the valve is condensed and collected in the CNV. This design preserves the RCS inventory for ECCS operation and does not result in the need for inventory addition.

   The NuScale Power Plant does not depressurize by reducing RCS inventory. Actuation of ECCS valves reduces RCS pressure, but is not a reduction in RCS inventory.

As described above, the five potential sources of leakage identified in NEI 12-06 are not directly applicable to the NuScale design. However, containment leakage was evaluated as a possible path for the loss of RCS inventory in an ELAP scenario, because the RCS partially transfers to the CNV following ECCS actuation. Using extremely conservative assumptions (e.g., worst case CNTS design leakage, the CNV remains pressurized to 1000 psia, and all lost inventory is deducted from lower riser level), the average change in liquid level is 0.41 in. per day. After the ECCS valves are opened and RPV level stabilizes, there is more than 9 ft of liquid water above the top of the core. With a

[22] Reactor coolant pump (RCP)

[23] Safety relief valve (SRV)

leakage rate of 0.41 inches per day and 9 ft of inventory above the core, core coverage is ensured for more than 260 days without RCS inventory addition.

The above discussion demonstrates that the NuScale Power Plant design is capable of extended coping without RCS inventory addition. The functions necessary to ensure adequate RCS inventory is maintained occur without operator action and the conditions are passively maintained by safety-related systems. The NuScale design eliminates the need for an inventory addition FLEX strategy to extend coping capabilities.

6.2.2 Reactivity Control

To establish and maintain safe shutdown conditions during an ELAP, the NPM is made subcritical and then maintained subcritical during the cooldown that follows.

Per boundary condition number 3 in Section 4.1.3, when the reactor trip occurs in the ELAP event, all control rods fully insert. This achieves initial subcriticality; however, depending on the time in core life, for some currently licensed designs the control rods alone may not provide sufficient negative reactivity to compensate for the positive reactivity added as the RCS cools.

To account for the cooldown reactivity addition, the NuScale Power Plant design includes a unique core design limit: the hot full power critical boron concentration is such that at cold zero power with all rods inserted, k_{eff} is less than 1.0. This allows the NPM to transition from operating conditions to shutdown and cooled down without the need for boron addition for an indefinite period.

To ensure the required boron concentration is maintained, the safety-related isolation of the demineralized water supply to the CVCS makeup pumps function is provided by the MPS, as described in Section 5.14. Although the demineralized water system pumps and the CVCS makeup pumps stop operating at the initiation of the ELAP due to the loss of ELVS, the MPS continues to monitor for plant conditions that would require actuation of this function. Additionally, all CVCS paths to and from the CNV are isolated within the first minute of the event when the containment isolation occurs.

During the ELAP event, all NPMs are made subcritical and remain subcritical without boron addition. This occurs without operator action and is maintained passively by the presence of the control rods and the boron in the RCS. This eliminates the need for a boron addition FLEX strategy to extend coping capabilities.

6.2.3 Decay Heat Removal

As detailed in Section 5.12, the safety-related DHRS actuates without operator action within the first minute of the ELAP 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 NPM. With both trains of the DHRS in service and passively transferring RCS heat to the UHS, safe shutdown conditions are established. The DHRS continues to passively reduce RCS temperature and perform the core cooling function until ECCS actuates at the 24-hour mark.

As described in Section 5.13, the ECCS valves have an IAB feature that prevents the valves from opening when the differential pressure between the RPV and CNV is greater than {{ }}2(a),(c), and the block releases at or above {{ }}2(a),(c) to allow valve operation. With the DHRS in operation, RPV pressure is greatly reduced and approaches {{ }}2(a),(c) into the ELAP. As a result, when the MPS timers expire after 24 hours, the IAB is defeated and the ECCS actuates. The natural circulation flow path established by the actuation of the ECCS causes a majority 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 require adequate UHS inventory for the performance of their core cooling safety function. As described in Section 5.11, the NPMs are partially immersed and the DHRS passive condensers are submerged in the UHS. With both systems transferring RCS heat from all 12 NPMs to the UHS and the addition of decay heat from the spent fuel in the SFP, the UHS temperature rises and begins to boil, but not before more than {{ }}2(a),(c).

As seen on Figure 6-1, it is at this point that UHS level begins to lower and no sooner than {{ }}2(a),(c) into the ELAP, the level lowers to the top of the DHRS passive condensers. This period of submergence ensures the DHRS can provide adequate core cooling far beyond the first 24 hours of the ELAP event.

Analysis of the long-term cooling capability of the ECCS demonstrates that safe shutdown conditions are maintained by ECCS operation with a UHS level of {{ }}2(a),(c). Per Figure 6-1, this UHS level is reached after more than {{ }}2(a),(c) into the ELAP. This means that without any operator actions to add inventory to the UHS, adequate core cooling is provided for no less than {{ }}2(a),(c) following the initiation of an ELAP.

[24] The lowest UHS level analyzed is {{ }}2(a),(c), which does not mean that the ECCS cannot provide adequate core cooling at lower levels, but instead that the analysis only demonstrates success to this UHS level.

{{

}}2(a),(c)

Figure 6-1 Ultimate heat sink level during an extended loss of alternating current power – 12 NuScale Power Modules initially operating

During the ELAP event, all NPMs are provided adequate core cooling to establish and maintain safe shutdown conditions for more than {{ }}2(a),(c). This occurs without operator action and is maintained by the passive operation of the DHRS and the ECCS. The redundancy and diversity provided within the DHRS and the independent operation of the two systems eliminates the need for additional FLEX equipment to extend coping capabilities.

6.2.4 Core Cooling Parameters

The key safety function of core cooling is established and maintained for greater than {{ }}2(a),(c) without operator action. The only plant parameter used to ensure the function is maintained is SFP level indication, which also represents UHS level. However, the parameters listed in Table 6-1 are available to assure the control room operators that the safety-related systems have performed as designed.

Per baseline coping capability assumption number 1 of Section 4.2, the instrumentation associated with each parameter is assumed to survive the BDBEE and remain fully available for a duration beyond the time necessary for the associated FLEX strategy function to be established and monitored.

Table 6-1 Core cooling key parameters

Function	Parameters for Assuring the Function is Established	Parameters for Assuring the Function is Maintained
RCS inventory control	RPV water level	None(1)
	Containment isolation valve positions	
Reactivity control	Neutron flux	None(1)
	Core inlet temperature	
	Core exit temperature	
	RTB status	
	CVCS containment isolation valve positions	
DHRS decay heat removal	Core exit temperature	SFP level(2)
	DHRS valve positions	
	MSIV positions	
	MSIV bypass isolation valve positions	
	FWIV positions	
	SFP level(2)	
ECCS decay heat removal	ECCS valve positions	SFP level(2)
	Containment water level	
	RPV water level	
	Core exit temperature	
	SFP level(2)	

(1) By design, once these functions are established, they are maintained indefinitely with no operator actions or power sources required.

(2) Spent fuel pool level provides indication of UHS level when UHS level is above the SFP weir.

6.3 Containment

The containment function is provided and maintained by plant safety-related systems for more than {{ }}2(a),(c) following initiation of an ELAP. The systems begin to operate passively without operator action and require no electrical power for initial or continued operation.

As previously discussed, the CNV, in addition to providing the containment function typical of existing designs, 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 an ELAP, the large volume of the UHS passively ensures adequate containment cooling is provided for more than {{ }}$^{2(a),(c)}$.

The CNV is isolated by closing the safety-related CIVs that occurs without operator action and requires no electrical power.

6.3.1 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 RP portion of the UHS.

During the ELAP, mass and energy are released into the CNV when the ECCS valves open. This 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.

As described in Section 6.2.3, analysis of the long-term cooling capability of the ECCS demonstrates that safe shutdown conditions are maintained by ECCS operation with a UHS level of {{ }}$^{2(a),(c)}$. Because of the CNV role in the operation of ECCS, this same analysis demonstrates that containment integrity with respect to temperature and pressure is maintained for no less than {{ }}$^{2(a),(c)}$ without operator action.

Because the containment function is provided passively by the CIVs and the CNV, and temperature and pressure are passively controlled by heat removal through the UHS in a manner that ensures containment integrity, no FLEX equipment is required to extend coping capabilities for this function for more than {{ }}$^{2(a),(c)}$.

6.3.2 Containment Parameters

The key safety function of containment is established and maintained for greater than {{ }}$^{2(a),(c)}$ without operator action. The only plant parameter used to ensure the function is maintained is SFP level indication, which also represents UHS level. However, the parameters listed in Table 6-2 are available to assure the control room operators that the safety-related systems have performed as designed.

Per baseline coping capability assumption number 1 of Section 4.2, the instrumentation associated with each parameter is assumed to survive the BDBEE and remain fully available for a duration beyond the time necessary for the associated FLEX strategy function to be established and monitored.

Table 6-2 Containment key parameters

Function	Parameters for Assuring the Function is Established	Parameters for Assuring the Function is Maintained
Containment isolation	CIV positions	None(1)
Containment heat removal	Wide range containment pressure	SFP level(2)
	SFP level(2)	

(1) By design, once these functions are established, they are maintained indefinitely.
(2) Spent fuel pool level provides indication of UHS level when UHS level is above the SFP weir.

6.4 Spent Fuel Cooling

Spent fuel in the SFP is passively cooled by the UHS for more than {{ }}2(a),(c) following initiation of an ELAP. At no time are operator actions or electrical power required for this to occur.

6.4.1 Spent Fuel Pool Level

As described in Section 5.11, 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 an ELAP unless 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 NPMs due to the ELAP and the decay heat addition from spent fuel in the SFP, UHS temperature rises and the pool begins to boil after more than {{ }}2(a),(c). At this point, UHS level begins to decrease, and without inventory addition, reaches the top of the SFP weir wall no sooner than {{ }}2(a),(c) into the event (see Table 5-2 and Figure 6-1).

With the SFP now divorced from the other pools in the UHS, level reduction continues based on spent fuel decay heat. Assuming no inventory is added to the SFP, level reaches the top of the spent fuel in the SFP after more than {{ }}2(a),(c) from the initiation of the ELAP. During this entire period, adequate cooling of the spent fuel in the SFP is ensured through submergence.

6.4.2 Spent Fuel Pool Cooling Parameters

The key safety function of SFP cooling is established and maintained for greater than {{ }}2(a),(c) without operator action. The only plant parameter used to ensure the function is established and maintained is SFP level indication.

Per baseline coping capability assumption number 1 of Section 4.2, the SFP level instruments are assumed to survive the BDBEE and remain fully available for a duration beyond the time necessary for the associated FLEX strategy function to be established and monitored.

Table 6-3 Spent fuel pool parameter

Function	Parameter for Assuring the Function is Established	Parameter for Assuring the Function is Maintained
Spent fuel cooling	SFP level	SFP level

6.5 Transition Mode (MODE 4)

As described in Section 5.10.1, an NPM 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 to the pressurizer baffle plate level, the NPM is cooled down below 200 degrees F, 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 NPM is then lifted {{ }}[2(a),(c)] above the RP floor using the RBC, and transported to the CNV flange tool in the RFP. At the CNV flange tool, the NPM is lifted a maximum of {{ }}[2(a),(c)] above the RP floor to clear the top of the tool and allow the NPM to be positioned for placement in the tool.

6.5.1 Core Cooling

Although the NPM 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 an NPM in transition occur when the NPM is lifted to the maximum lift height. This is because the maximum lift height represents the minimum CNV immersion in the UHS.

If an ELAP occurs during the brief period of time when an NPM in transition is lifted to the maximum height required for placement in the CNV flange tool, the UHS provides sufficient decay heat removal to maintain safe shutdown conditions during the approximately {{ }}[2(a),(c)] needed for pool heat up and boil off to reduce UHS level to {{ }}[2(a),(c)].

6.5.2 Reactor Building Crane Capacity

During lifts of an NPM by the RBC, the pool water in the UHS provides buoyancy that reduces the weight required to be lifted. For an NPM lift to the maximum height required for placement in the CNV flange tool, the RBC does not exceed the rated lift capacity with UHS level at or above {{ }}[2(a),(c)] However, a crane of the RBC design is permitted to conduct lifts that are up to 125 percent of the rated capacity for a planned engineering lift, which equates to 1062.5 tons for the RBC.

As seen on Figure 6-2, an NPM submergence of {{ }}2(a),(c) corresponds to just below 1000 tons. With an NPM lifted to the maximum lift height, {{ }}2(a),(c) of submergence occurs at a UHS pool level of {{ }}2(a),(c).

{{

}}2(a),(c)

Figure 6-2 NuScale Power Module submergence and buoyancy effect

During an ELAP, if inventory is not added to the UHS, pool level would not lower to {{ }}2(a),(c). Therefore, during an ELAP event with an NPM lifted to the maximum height for placement in the CNV flange tool, the reduction in buoyancy that occurs due to lowering UHS level does not result in the load on the RBC exceeding 125 percent of the rated capacity within {{ }}2(a),(c). Additionally, the cumulative time any of the NPMs spend lifted to the maximum lift height represents a small fraction of the operating cycle.

6.6 Refueling Mode (MODE 5)

During refueling operations, the NPM 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 through 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 through the weir.

During an ELAP with one NPM refueling, less heat is added to the UHS than if all 12 NPMs were initially operating. Therefore, a comparatively longer period of time is required for the UHS to begin to boil and pool level is reduced at a lower rate. With the fuel submerged, adequate cooling is ensured. Per Table 6-4, UHS pool level would not lower to the top of fuel traveling through the weir in the SFP weir wall before {{ }}2(a),(c) and level would not reach the fuel in the refueling NPM's reactor core before {{ }}2(a),(c).

Table 6-4 Ultimate heat sink heat up and boil off – 11 NuScale Power Modules initially operating, 1 NuScale Power Module refueling

Location	Pool Level	Cumulative Time
Pool heat up to boiling	{{	
Pool boil off to 10 ft. above spent fuel traveling above weir in SFP weir wall		
Pool boil off to top of spent fuel traveling above weir in SFP weir wall		
Pool boil off to top of weir in SFP weir wall		
Pool boil off to top of spent fuel in RFP		}}2(a),(c)

6.7 Passive Building Cooling

As described in Sections 5.17 and 5.18, normal RXB and CRB cooling are lost at the initiation of an ELAP. The NuScale Power Plant design relies on passive cooling during the event to maintain area temperatures within allowable limits for equipment operation, as necessary.

During an ELAP, the key safety functions are established and maintained by the automatic responses of safety-related equipment. These responses occur in the 24 hours following initiation of the event, with the ECCS valves opening at the 24-hour mark. Although no operator action is required, indications are available in the MCR to provide assurance that the systems have responded as designed.

6.7.1 Instrumentation and Indication

The MPS for each NPM functions to actuate safety-related equipment to establish and maintain the key safety functions. Although not required for the MPS to perform its safety-related functions, EDSS-MS does supply power for the NPM instrumentation and associated indications provided to the MCR. A group of rooms on the 75-ft and 86-ft elevations of the RXB contain the EDSS-MS batteries and battery chargers, and the MPS I/O cabinets for each NPM.

Analysis of the passive cooling that occurs during an ELAP shows that for all 12 NPMs the environmental conditions of the EDSS-MS and MPS rooms in the RXB remain below {{ }}2(a),(c) relative humidity for the 24-hour period in which the MPS actuations occur, and continue to remain below these values for more than 96 hours. See Appendix A, Table A-1 for more detail.

The PPS does not perform any functions required to establish or maintain the key safety functions, but the system does provide indications to the MCR during the event. The

SDIS displays these indications, as well as those provided by the MPS, and both the PPS and the SDIS are powered by the EDSS-C.

Analysis of the passive cooling that occurs during an ELAP shows that the environmental conditions of the EDSS-C, PPS, and SDIS rooms in the CRB remain below {{ }}2(a),(c) relative humidity for the 24-hour period in which the MPS actuations occur, and continue to remain below these values for more than 96 hours. See Appendix A, Table A-2 for more detail.

6.7.2 Control Room Habitability

As described in Sections 5.18 and 5.19, the CRVS stops performing HVAC functions at the initiation of the event and isolates the CRE, while the CRHS actuates to pressurize the MCR. As a result, the environmental conditions in the MCR are determined by passive cooling and the flow of breathing air from the CRHS.

Analysis of the environmental conditions in the MCR during an ELAP shows that the wet bulb globe temperature remains below {{ }}2(a),(c) for the 24-hour period in which the MPS actuations occur, and continues to remain below {{ }}2(a),(c) for more than 96 hours. See Appendix A, Table A-3 for more detail.

6.8 Baseline Coping Capability

Based on the analysis of the key safety functions coping capability provided by installed plant equipment, a baseline coping capability has been established. The baseline capability is summarized in Table 6-5.

Table 6-5 FLEX baseline capability summary

Safety Function			Method	Baseline Capability
Core cooling	Decay heat removal (installed NPMs)		• DHRS with UHS • ECCS with UHS	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))
Core cooling	Monitoring	Initial	• Core exit temperature • DHRS valve positions • MSIV positions • MSIV bypass isolation valve positions • FWIV positions • SFP level(1) • ECCS valve positions • Containment water level • RPV water level	Powered instruments (used for initial verification)
Core cooling	Monitoring	Continuous	• SFP level(1)	SFP level instruments (indefinite)
Core cooling	Decay heat removal (transition NPM)		• Conduction to the UHS	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))
Core cooling	Monitoring		• SFP level(1)	SFP level instruments (indefinite)
Core cooling	Decay heat removal (refueling NPM)		• Direct cooling by UHS submergence	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))
Core cooling	Monitoring		• SFP level(1)	SFP level instruments (indefinite)
Core cooling	RCS inventory		• Containment isolation	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))
Core cooling	Monitoring		• RPV water level • CIV positions	Powered instruments (used for initial verification)
Core cooling	Reactivity control		• Containment isolation	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))

Safety Function			Method	Baseline Capability
	Monitoring		• Neutron flux • Core inlet temperature • Core exit temperature • RTB status • CVCS containment isolation valve positions	Powered instruments (used for initial verification)
Containment	Containment heat removal		Conduction to the UHS	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))
Containment	Monitoring	Initial	• Wide range containment pressure • SFP level(1)	Powered instruments (used for initial verification)
Containment	Monitoring	Continuous	• SFP level(1)	SFP level instruments (indefinite)
Containment	Containment		• Containment isolation	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))
Containment	Monitoring		• CIV positions	Powered instruments (used for initial verification)
SFP cooling	Spent fuel cooling		• Direct cooling by SFP submergence	Use of safety-related plant equipment for initial coping (beyond {{ }}2(a),(c))
SFP cooling	Monitoring		• SFP level(1)	SFP level instruments (indefinite)

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

7.0 Diverse and Flexible Coping Strategies

7.1 Phase 1

Analysis of the NuScale Power Plant design demonstrates a baseline coping capability of more than {{ }}2(a),(c) for maintaining the three key safety functions with installed plant equipment and without operator action.

The core cooling function is automatically established and passively maintained by safety-related equipment as follows:

- The DHRS actuation valves open to establish natural circulation flow and commence the transfer of RCS heat to the fluid contained in the passive condenser loops.
- The DHRS passive condensers are submerged in the UHS and transfer their heat to the UHS.
- The CIVs close to maintain RCS inventory.
- The ECCS valves open to establish natural circulation flow of reactor coolant between the RPV and the CNV. The CNV is partially immersed in the UHS and transfers heat to the UHS.

The containment function is automatically established and passively maintained by safety-related equipment as follows:

- The CIVs close to establish containment of the RCS.
- Containment temperature and pressure control are provided by partial immersion of the CNV in the UHS.

The SFP cooling function is maintained by submergence of the spent fuel in the UHS.

7.2 Phase 2

Phase 2, as defined in NEI 12-06, involves augmenting plant equipment or transitioning to on-site FLEX equipment and consumables to maintain or restore key functions. This phase is intended to ensure safety functions are maintained until off-site resources can be used to obtain additional capability and redundancy. The NuScale Power Plant design does not require equipment beyond the installed safety-related plant equipment used to establish and maintain the key safety functions in Phase 1, and the operation of that equipment does not need to be augmented to extend the coping time to 72 hours. As a result, the NuScale Power Plant FLEX strategy does not include the traditional Phase 2 strategy.

7.3 Phase 3

The baseline coping capability for the NuScale Power Plant design extends beyond {{ }}2(a),(c). This capability is based on a minimum UHS pool level of {{ }}2(a),(c). As such, the Phase 3 FLEX strategy is to monitor UHS pool level and add inventory to the UHS from the SFP assured makeup line as necessary to maintain UHS pool level at or above {{ }}[2(a),(c)].

8.0 Configuration Management

The proposed rulemaking for 10 CFR 50.155 (Reference 13.1.3) includes requirements for change control of the FLEX strategies. NEI 12-06 (Reference 13.1.16) provides guidance for the change (configuration) control of the FLEX strategies. The NEI 12-06 guidance states that the FLEX strategies and basis for these strategies will be maintained in an overall program document.

Changes to the FLEX strategies are to be documented in the FLEX program document. NEI 12-06 also states that changes to the FLEX strategies may be made without NRC pre-approval provided the revised FLEX strategy meets the following:

i. the provisions of this guideline (NEI 12-06)
ii. the change to the strategies and guidance implement an alternative or exception approved by the NRC, provided that the bases of the NRC approval are applicable to the licensee's facility, or
iii. an evaluation demonstrates that the provisions of Order EA-12-049 continue to be met

In addition, an engineering basis is documented that ensures that the change in FLEX strategy continues to ensure the key safety functions (core and SFP cooling, containment function) are met. Documentation of all changes shall be maintained as long as the FLEX strategies are required.

A COL applicant will develop a FLEX program document using the guidance in NEI 12-06 (Reference 13.1.16).

9.0 Diverse and Flexible Coping Strategies Equipment

The proposed rulemaking for 10 CFR 50.155 (Reference 13.1.3) includes requirements for maintenance of FLEX equipment. NEI 12-06 (Reference 13.1.16) provides the guidance for maintenance, testing, and out-of-service time of FLEX equipment.

The installed credited FLEX equipment is safety-related and maintenance, testing, and out-of-service times are conducted in accordance with the existing plant processes and programs. The NuScale Power Plant coping capability is greater than {{ }}$^{2(a),(c)}$; therefore, no onsite FLEX equipment is required and no program to address FLEX equipment maintenance, testing, and out of service time is required to be developed by the COL applicant.

10.0 Procedures

The NTTF Recommendation 8.1 was related to integrating applicable plant procedures and NTTF Recommendation 4 resulted in the requirement to develop FLEX strategies and implementing procedures. These recommendations are incorporated in the proposed rulemaking for 10 CFR 50.155 (Reference 13.1.3). NEI 12-06 contains guidance for developing the FLEX strategies and implementing procedures.

The procedures or guidelines developed to reflect the FLEX strategies using the guidance of NEI 12-06 (Reference 13.1.16) are referred to as FLEX Support Guidelines (FSGs). Each licensee will also have EOPs per technical specifications. In addition, 10 CFR 50.54(hh)(2)[25] requires licensees to develop and implement strategies intended to maintain or restore key safety function capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire. These strategies are normally called extensive damage mitigation guidelines (EDMG).

The proposed rulemaking (Reference 13.1.3) will require that the EDMG and FSGs be integrated with the EOPs. NEI 14-01 (Reference 13.1.17), not presently endorsed, provides guidance for the integration of these procedures.

A COL applicant will develop FSGs using the guidance in NEI 12-06.

A COL applicant will integrate FSGs and EDMG with the EOPs using the guidance of NEI 14-01, or future endorsed guidance.

[25] 10 CFR 50.54(hh)(2) may be moved to new regulation 10 CFR 50.155, presently in rulemaking process (Federal Register notice Vol. 80, No. 219 pages 70610-701647)

11.0 Personnel

Numerous NTTF recommendations address staffing and training requirements as shown in Table 2-1 of this report. These recommendations are included in the proposed rulemaking for 10 CFR 50.155 (Reference 13.1.3). Additionally, NEI 12-06 and NEI 12-01 (Reference 13.1.14) contain guidance for staffing and training for personnel during a BDBEE. A COL applicant that references the NuScale Power Plant design certification will develop, implement, and maintain the training and qualification of personnel that perform activities in accordance with FLEX support guidelines, severe accident mitigation guidelines, and extensive damage mitigation guidelines. The training and qualification on these activities will be developed using the systems approach to training as defined in 10 CFR 55.4 except for elements already covered under other NRC regulations.

A COL applicant that references the NuScale Power Plant design certification will perform an analysis that demonstrates the Emergency Response Organization staff has ability to implement the strategies of the emergency operating procedures, severe accident mitigation guidelines, FLEX support guidelines, and extensive damage mitigation guidelines. The analysis will be performed with the off-site response organization access to on-site being impeded. The event shall be a loss of all on-site and off-site alternating current power and loss of normal access to the ultimate heat sink.

A COL applicant that references the NuScale Power Plant design certification will develop a supporting Emergency Response Organization structure with defined roles and responsibilities to implement the strategies of the emergency operating procedures, severe accident mitigation guidelines, FLEX support guidelines, and extensive damage mitigation guidelines.

A COL applicant that references the NuScale Power Plant design certification will develop drills or exercises that demonstrate the ability to transition to one or more of the strategies and guidelines of the emergency operating procedures, FLEX support guidelines, extensive damage mitigation guidelines, and severe accident mitigation guidelines using only the station communication equipment designed to be available following an extended loss of alternating current including effects of the loss of the local communications infrastructure.

12.0 Conclusion

As discussed in this report the NuScale Power Plant design has a Phase 1 coping time of greater than {{ }}[2(a),(c)] to maintain the key safety functions of core cooling and containment integrity. The Phase 1 coping time for the spent fuel cooling is greater than {{ }}[2(a),(c)]. These Phase 1 coping times are maintained without electrical power and no operator actions. The only FLEX strategy needed to maintain the key safety functions is to provide makeup to the UHS, through the UHS assured makeup line, to maintain the UHS water level greater than {{ }}[2(a),(c)].

The COL applicant is required to develop and implement a FLEX Program that will include but not limited to a program document, applicable procedures or guidelines, staffing, and training requirements.

13.0 References

13.1.1 U.S. Nuclear Regulatory Commission, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," Commission Paper SECY-11-0093, July 12, 2011, Agencywide Document Access and Management System (ADAMS) Accession No. ML111861807.

13.1.2 U.S. Nuclear Regulatory Commission, "Recommended Actions to be Taken Without Delay from the Near-Term Task Force Report," Commission Paper SECY-11-0124, September 9, 2011, Agencywide Document Access and Management System (ADAMS) Accession No. ML11245A158.

13.1.3 U.S. Nuclear Regulatory Commission, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," Commission Paper SECY-11-0137, October 3, 2011, Agencywide Document Access and Management System (ADAMS) Accession No. ML11272A111.

13.1.4 U.S. Nuclear Regulatory Commission, "Mitigation of Beyond-Design-Basis Events; Proposed Rule," Federal Register, Vol. 80, No. 219, November 13, 2015, pp. 70610-701647.

13.1.5 U.S. Nuclear Regulatory Commission, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" EA-12-049, March 12, 2012, Agencywide Document Access and Management System (ADAMS) Accession No. ML12056A045.

13.1.6 U.S. Nuclear Regulatory Commission, "Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" JLD-ISG-2012-01, Revision 1, January 22, 2016, Agencywide Document Access and Management System (ADAMS) Accession No. ML15357A163.

13.1.7 U.S. Nuclear Regulatory Commission, "Closure of Fukushima Tier 3 Recommendations Related to Containment Vents, Hydrogen Control, and Enhanced Instrumentation, Commission Paper SECY-16-0041, March 31, 2016, Agencywide Document Access and Management System (ADAMS) Accession No. ML16049A079.

13.1.8 U.S. Nuclear Regulatory Commission, “Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation” EA-12-051, March 12, 2012, Agencywide Document Access and Management System (ADAMS) Accession No. ML12054A679.

13.1.9 U.S. Nuclear Regulatory Commission, “Compliance with Order EA-12-051, Reliable Spent Fuel Pool Instrumentation”, JLD-ISG-2012-03, Revision 0, August 29, 2012, Agencywide Document Access and Management System (ADAMS) Accession No. ML12221A339.

13.1.10 U.S. Nuclear Regulatory Commission, “Consolidation of Japan Lessons Learned Near Term Task Force Recommendations 4 and 7 Regulatory Activities,” COMSECY-13-0002, March 4, 2013, Agencywide Document Access and Management System (ADAMS) Accession No. ML13063A548.

13.1.11 U.S. Nuclear Regulatory Commission, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants,” Regulatory Guide 1.209, March 2007.

13.1.12 U.S. Nuclear Regulatory Commission, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,” Regulatory Guide 1.89, Rev. 1, June 1984.

13.1.13 U.S. Nuclear Regulatory Commission, “Single-Failure-Proof Cranes for Nuclear Power Plants”, NUREG-0554, May 1979.

13.1.14 Nuclear Energy Institute, “Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities,” NEI 12-01, Revision 0, May 2012, Agencywide Document Access and Management System (ADAMS) Accession No. ML12125A412.

13.1.15 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.

13.1.16 Nuclear Energy Institute, “Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,” NEI 12-06, Revision 2, December 2015.

13.1.17 Nuclear Energy Institute, “Enhancements to Emergency Response Capabilities for Beyond Design Basis Events and Severe Accidents,” NEI 13-06, Revision 0, September 2014, Agencywide Document Access and Management System (ADAMS) Accession No. ML14269A230.

13.1.18 Nuclear Energy Institute, “Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents,” NEI 14-01, Revision 0, September 2014, Agencywide Document Access and Management System (ADAMS) Accession No. ML14269A236.

13.1.19 U.S. Nuclear Regulatory Commission, “Flexible Mitigation Strategies for Beyond-Design-Basis Events,” Draft RG DG-1301 (Proposed New RG 1.226), November 2015.

13.1.20 U.S. Nuclear Regulatory Commission, “Wide-Range Spent Fuel Pool Level Instrumentation,” Draft RG DG-1317 (Proposed New RG 1.227), November 2015.

13.1.21 U.S. Nuclear Regulatory Commission, “Integrated Response Capabilities For Beyond-Design-Basis Events,” Draft RG DG-1319 (Proposed New RG 1.228), November 2015.

13.1.22 Institute of Electrical and Electronics Engineers, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” IEEE Std. 323-2003, Piscataway, NJ.

13.1.23 Institute of Electrical and Electronics Engineers, “IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations,” IEEE Std. 323-1974, Piscataway, NJ.

13.1.24 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.

13.1.25 American National Standards Institute, “American National Standards (for Radioactive Materials), Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500kg) or More,” ANSI N14.6-1993, New York, NY.

Appendix A. Passive Cooling - Environmental Conditions during an Extended Loss of Alternating Current Power

The purpose of this appendix is to provide the expected environmental conditions during an ELAP for the following rooms:

- EDSS-MS
- MPS I/O
- Battery Chargers
- EDSS-C
- PPS and SDIS
- MCR

A.1 Highly Reliable Direct Current Power System-Module Specific and Module Protection System Rooms in the Reactor Building

Table A-1 demonstrates that the environmental conditions of the EDSS-MS and MPS rooms associated with NPM 01[1] remain below {{ }}2(a)(c) relative humidity for the 24-hour period in which the MPS actuations occur and continue to remain below these values for more than 96 hours.

Table A-1 Highly reliable direct current power system and module protection system input/output room environmental conditions for NuScale Power Module 01

{{

}}2(a),(c)

[1] The 12 NPMs are identified numerically, 01 through 12. The equipment room numbers are specific to the RXB, and do not identify the associated NPM.

The environmental conditions for the rooms containing the same equipment for the other 11 NPMs are similar to those seen in Table A-1, and in all cases remain below {{ }}2(a),(c).

A.2 Highly Reliable Direct Current Power System -Common, Plant Protection System, and Safety Display and Indication System Rooms in the Control Building

Table A-2 demonstrates that the environmental conditions of CRB rooms for the EDSS-C, PPS, and SDIS remain below {{ }}2(a),(c) relative humidity for the 24-hour period in which the MPS actuations occur and continue to remain below these values for more than 96 hours.

Table A-2 Highly reliable direct current power system, plant protection system, and safety display and indication system Control Building room environmental conditions

{{

}}2(a),(c)

(1) Values have been rounded up for conservatism.
(2) This room air temperature spikes to {{ }}2(a),(c) at the beginning of the event due to a PCS cabinet, but cools when the cabinet de-energizes.

A.3 Control Room

Analysis of the environmental conditions in the control room during an ELAP shows that the wet bulb globe temperature remains below {{ }}2(a),(c) for the 24-hour period in which the MPS actuations occur, and continues to remain below {{ }}2(a),(c) for more than 96 hours.

Table A-3 Control room environmental conditions during an extended loss of alternating current power

{{

}}2(a),(c)

Appendix B. Evaluation of the NEI 12-06 Approach to Pressurized Water Reactor Functions

The purpose of this appendix is to provide an evaluation of NuScale's approach to the strategies prescribed by NEI 12-06 for the following key safety functions:

- core cooling
- containment
- spent fuel cooling

B.1 Approach to Pressurized Water Reactor Functions

Appendix D, Tables D-1, D-2, and D-3 of NEI 12-06 provide a detailed summary of performance attributes for each of the three key safety functions. The evaluation of the recommended methods, including the purpose and performance attributes for each method, is detailed in Tables B-1, B-2, and B-3. The evaluation includes applicability determinations for each method and performance attribute, as well as verification that the intended purpose of each applicable recommendation is met either through the design of the NuScale Power Plant or through the FLEX strategy.

Table B-1 Evaluation

NEI 12-06 Recommendation	NuScale Power Plant Design
Core Cooling	
<u>**Safety Function**</u>**: Reactor Core Cooling and Heat Removal (steam generators available)**	
<u>Method</u>: Auxiliary Feedwater/emergency feedwater	Not applicable. The NuScale Power Plant design does not include an equivalent system. This method provides core cooling by maintaining SG water level while bleeding steam, which is not part of the NuScale Power Plant FLEX strategy.
Purpose: Provide SG makeup sufficient to maintain or restore SG level with plant equipment and power supplies to the greatest extent possible to provide core cooling.	The primary purpose of this method, to provide core cooling, is inherent to the NuScale Power Plant design. Passive decay heat removal without electrical power or operator action is provided for more than {{ }}2(a),(c) by the DHRS and ECCS.

NEI 12-06 Recommendation	NuScale Power Plant Design
Performance Attributes: • Extend installed coping capability through procedural enhancements (e.g., load shedding), provision of FLEX battery chargers and other power supplies. • Objective is to provide extended baseline coping capability with plant equipment. • Procedures/guidance to include local manual initiation of AC-independent auxiliary feedwater/emergency feedwater pumps.	Evaluation of Performance Attributes: • This attribute is inherent to the NuScale Power Plant design. The electrical power required for operation of the SFP level indicators is included in the design of the instruments. Extending installed coping capability is not necessary, thus FLEX battery chargers and other power supplies are also not necessary. • This attribute objective is met inherently by the NuScale Power Plant design. The DHRS and ECCS with UHS core cooling baseline coping capability extends beyond {{ }}[2(a),(c)]. • Not applicable. The intent of this attribute is to allow manual operation of systems that have lost power necessary for remote operation in support of the core cooling function. The DHRS and ECCS actuate without operator action, are AC-independent, and fail safe.
Method Summary: The specific method recommended is not applicable to the NuScale Power Plant design, but the intended purpose is met. All performance attributes are either met or not applicable.	
Method: Depressurize SG for Makeup with FLEX Injection Source	Not applicable. This method provides backup to the initial method (auxiliary feedwater/emergency feedwater) of maintaining SG water level while bleeding steam, which is not part of the NuScale Power Plant FLEX strategy.

NEI 12-06 Recommendation	NuScale Power Plant Design
Purpose: Provide SG makeup sufficient to maintain or restore SG level with diverse and flexible capability.	The NEI strategy for core cooling, including this method, is intended to provide diversity and flexibility. The ability to feed the SGs with either of two pumps provides flexibility, while the ability to connect the FLEX pump to SG injection points in separate divisions or trains provides diversity. This same flexibility and diversity is inherent in the DHRS design through the two independent passive condenser loops. Either passive condenser can perform the function of decay heat removal, and they are connected to separate SG divisions or trains. Additionally, the ECCS provides passive core cooling in a manner entirely independent of the DHRS to provide diversity and flexibility.
Performance Attributes: • Primary and alternate injection points are required to establish capability to inject through separate divisions or trains, i.e., should not have both connections in one division or train. • Makeup paths supply required SGs • SG makeup rate should exceed decay heat levels at time of planned deployment in order to support restoring SG water level, e.g., 200 gpm.	Evaluation of Performance Attributes: • This attribute is not applicable; however the intent of the attribute is met inherently by the NuScale Power Plant design. The DHRS includes fully independent divisions within the system’s design, and although both divisions actuate during an ELAP, each division is capable of sufficient decay heat removal to establish safe shutdown conditions. As a result of this design, the DHRS functionally includes a “primary and alternate.” • Not applicable. The DHRS trains are closed loops. Inventory addition is not needed for decay heat removal. The ECCS uses the RCS inventory present at the time of the containment isolation to provide decay heat removal. Inventory addition is not necessary. • Not applicable. DHRS and ECCS inventory addition is not necessary. (UHS inventory addition, which also ensures core cooling is maintained, is discussed in the spent fuel cooling section of this table.)

NEI 12-06 Recommendation	NuScale Power Plant Design
• Analysis should demonstrate that the guidance and equipment for combined SG depressurization and makeup capability supports continued core cooling.	• Analysis demonstrates that the DHRS/ECCS with UHS core cooling baseline coping capability extends beyond {{ }}[2(a),(c)].
Method Summary: The specific method recommended is not applicable to the NuScale Power Plant design, but the intended purpose is met. All performance attributes are either met or not applicable.	
Method: Sustained Source of Water[2]	The RCS inventory present at the initiation of an ELAP is sufficient for passive decay heat removal for more than {{ }}[2(a),(c)]. The UHS inventory present at the initiation of an ELAP is sufficient to support passive decay heat removal for more than {{ }}[2(a),(c)].
Purpose: Water is a critical resource in sustaining coping capability.	The inventory of water present in the RCS and the UHS at the initiation of an ELAP is sufficient to sustain core cooling coping capability for more than {{ }}[2(a),(c)].
Performance Attribute: • Water source sufficient to supply water indefinitely.	Evaluation of Performance Attribute: • The intent of this performance attribute is met through the NuScale Power Plant design that includes a baseline coping capability beyond 50 days. The FLEX strategy requires UHS level to be maintained at or above 45 feet and permits the plant operators to determine the most appropriate source for the given conditions, while providing an ample period of time to make that determination.
Method Summary: The NuScale Power Plant design includes a sustained source of water for the core cooling function for more than {{ }}[2(a),(c)], but not indefinitely. However, the duration of the baseline coping capability provides sufficient time for the plant operators to ensure an indefinite supply of water exists. This satisfies the intended purpose of this recommended method and its associated performance attribute.	

[2] Although this method is discussed in Table D-1 of NEI 12-06, which is specific to core cooling, the “Baseline Capability” statement also addresses SFP heat removal. An evaluation of SFP water inventory for the NuScale Power Plant design is included in Table B-3 of this assessment.

NEI 12-06 Recommendation	NuScale Power Plant Design
Safety Function: RCS Inventory Control and Long-Term Subcriticality	
Method: Low Leak RCP Seals and/or borated high pressure RCS makeup required	The method of including low leakage RCP seals is not applicable to the NuScale Power Plant design. The method of borated high pressure RCS makeup for inventory control is not needed for the NuScale Power Plant, because the RCS was designed with minimal leakage. The method of borated high pressure RCS makeup for reactivity control is not needed for the NuScale Power Plant, because the combination of inserted control rods and normal operating RCS boron concentration ensures the core is maintained subcritical during RCS cooldown.
Purpose: Extended coping without RCS makeup is not possible without minimal RCS leakage. Plants must evaluate use of low leak RCP seals and/or providing a high pressure RCS makeup pump.	Purpose: The NuScale Power Plant design includes operating with boron concentrations that maintain the core subcritical with all rods inserted during RCS cooldown. Automatic containment isolation isolates potential external leakage paths, and any RCS leakage into containment is preserved for ECCS operation by design.

NEI 12-06 Recommendation	NuScale Power Plant Design
Performance Attributes:	Evaluation of Performance Attributes:
• Makeup capability to maintain core cooling.	• Sufficient RCS inventory is maintained for more than {{ }}2(a),(c) by containment isolation. Makeup capability is not necessary to maintain core cooling.
• Sufficient letdown to support required makeup and ensure subcriticality.	• With adequate boron present in the RCS at event initiation to maintain subcriticality during cooldown, and sufficient inventory maintained for core cooling, makeup is not necessary. Because makeup capability is not necessary, RCS letdown is not necessary during an ELAP.
• In order to address the requirement for diversity, if re-powering of installed charging pumps is used for this function, then either (a) multiple power connection points should be provided to the charging pump, or (b) provide a single power supply connection point for the charging pump and a single connection point for a FLEX makeup pump.	• Not applicable.
Method Summary: The specific method recommended, which includes two distinct elements, is not applicable (RCP seal leakage) or not necessary for the NuScale Power Plant (borated RCS makeup), but the intended purposes are met through the design. All performance attributes are either met or not applicable.	

NEI 12-06 Recommendation	NuScale Power Plant Design
<u>Safety Function</u>: Core Cooling and Heat Removal (Modes 5 and 6 with steam generators not available)[3]	
<u>Method</u>: All Plants Provide Means to Provide Borated RCS Makeup[4]	A method of borated RCS makeup is not needed for an NPM in transition, because the RPV and CNV are flooded to the level of the pressurizer baffle plate with borated water, and the CNV is isolated. A method of borated RCS makeup is not needed for an NPM in refueling because the reactor core is in direct contact with the borated UHS. A source or borated water is not needed, because borated RCS makeup is not required.
Purpose: Long-term sustained coping will require RCS makeup and boration	An NPM in transition is borated to the refueling concentration, the RPV and CNV are flooded to the level of the pressurizer baffle plate, and the CNV is isolated. Only direct contact between the CNV and UHS is needed. During refueling, the NPM is disassembled, and the reactor core is directly exposed to the borated UHS. Cooling is provided by submergence.
Performance Attributes: • Diverse injection points or methods are required to establish capability to inject through separate divisions/trains, i.e., should not have both connections in one division or train.	Evaluation of Performance Attributes: • Not applicable.

[3] This safety function represents shutdown and refueling modes, which are addressed by two different methods for the NuScale design.

[4] This method requires a source of borated water in addition to a means of RCS makeup.

NEI 12-06 Recommendation	NuScale Power Plant Design
• Connection to RCS for makeup should be capable of flow rates sufficient for simultaneous core heat removal and boron flushing (combined makeup flow exceeding 300* gpm). • On-site FLEX pump for RCS makeup. This can be the SG makeup pump since both will not be required at same time.	• Not applicable. • Not applicable.
• Source of borated water required. Could be an on-site tank, or could be provided by off-site resources.	• This attribute is met inherently by the NuScale Power Plant design. The borated water supply necessary for long-term sustained coping is present in the RPV and CNV for an NPM in transition, and in the UHS for an NPM in refueling.
Method Summary: The recommended method is not necessary for the NuScale Power Plant, but the intended purpose, to ensure an adequate source of borated water exists for core cooling, is met inherently through the design. All performance attributes are met or not applicable.	
Safety Function: Key Reactor Parameters	
Method: SG Level	This method is not applicable for the NuScale Power Plant, because SG water level is not controlled for core cooling during an ELAP.
Purpose: Necessary to control heat removal.	The NEI recommended parameter of SG Level is included because operators are actively controlling SG injection to maintain core cooling. For the NuScale Power Plant, the DHRS and ECCS passively remove decay heat without operator action, and this function is ensured when UHS level is at or above {{ }}[2(a),(c)].
Method: SG Pressure	This method is not applicable for the NuScale Power Plant, because SG injection is not part of the FLEX strategy.

NEI 12-06 Recommendation	NuScale Power Plant Design
Purpose: Necessary to transition to FLEX pump.	The NEI recommended parameter of SG Pressure is included to allow the operators to take the action of transitioning to the FLEX pump. For the NuScale Power Plant, the core cooling function transfers from the DHRS to the ECCS without operator action.
Method: RCS Pressure	This method is not applicable for the NuScale Power Plant, because RCS makeup is not needed.
Purpose: Necessary to assure depressurization to gain access to inventory for RCS makeup in safety injection accumulators.	The NEI recommended parameter of RCS pressure is included because inventory addition and boron addition will be required for coping in traditional PWRs. The parameter will also be used by operators to actively control the RCS cooldown rate to ensure subcooling is maintained. For the NuScale Power Plant, neither inventory addition nor boron addition are necessary for more than {{ }}[2(a),(c)].
Method: RCS Temperature	This method is not applicable for the NuScale Power Plant, because cooldown occurs passively.
Purpose: Necessary to monitor subcooling.	The NEI recommended parameter of RCS Temperature is included to allow the operators to control RCS cooldown and ensure subcooling is maintained. For the NuScale Power Plant, the RCS cooldown occurs passively, and does not need to be controlled by the operators to ensure subcooling is maintained.

NEI 12-06 Recommendation	NuScale Power Plant Design
Performance Attributes: • Identify instruments to be relied upon, including control room and field instruments. • Depending on strategy employed, additional parameters may be required.	Evaluation of Performance Attributes: • This attribute is met. The UHS system includes four level instruments that meet the recommendations of NEI 12-02. Level indication is available from the PPS through the SDIS displays in the control room, and locally in the field. • This attribute is met. Because the core cooling safety function is established and maintained without operator action for more than {{ }}[2(a),(c)], no additional parameters are required. SFP and UHS level indication are used to prompt the addition of inventory to the SFP, as necessary.
<u>Methods Summary</u>: The specific methods recommended are not applicable for the NuScale Power Plant because of the differences in the recommended NEI 12-06 and the NuScale Power Plant FLEX strategies. However, the performance attributes are met through the identification of the parameters and associated instruments relied upon for the NuScale Power Plant FLEX strategy.	

Table B-2 Evaluation of the NEI 12-06 approach to pressurized water reactor containment functions

NEI 12-06 Recommendation	NuScale Power Plant Design
Containment	
Safety Function: Containment Function	
Method: Containment Spray	Not applicable. The NuScale Power Plant design does not include an equivalent system or function.
Purpose: In the long-term containment pressure may rise due to leakage from RCS adding heat to containment. Containment spray can help manage containment pressure.	The purpose of this method, to provide containment pressure control, is inherent to the NuScale Power Plant design. Passive CNV heat removal without electrical power or operator action is provided for more than {{ }}$^{2(a),(c)}$ by partial immersion in the UHS.
Performance Attribute: • Due to the long-term nature of this function, the connection does not need to be a permanent modification. However, if a temporary connection, e.g., via valve bonnet, then this should be pre-identified.	Evaluation of Performance Attribute: • This attribute is not applicable.
Methods Summary: The specific method recommended is not applicable for the NuScale Power Plant, but the intended purpose, to maintain containment pressure and temperature within design limits during an ELAP, is inherent to the design. The performance attribute is not applicable.	
Safety Function: Containment Integrity (Ice Condenser Containments Only)	
Method: Hydrogen Igniters	Not applicable. The NuScale Power Plant design does not include hydrogen igniters, and analysis demonstrates that core damage does not occur during an ELAP as described and bounded by NEI 12-06.
Purpose: Maintain containment function post-core damage.	Not applicable.

NEI 12-06 Recommendation	NuScale Power Plant Design
Performance Attributes:	Evaluation of Performance Attributes:
• Diverse power connection points are required to establish capability through separate divisions/trains, i.e., should not have both connections in one division/train. • Procedures/guidance to prioritize deployment strategies.	• This attribute is not applicable. • This attribute is not applicable.
Methods Summary: The recommended method is not applicable for the NuScale Power Plant, because core damage does not occur during an ELAP as described and bounded by NEI 12-06. The performance attributes are not applicable.	
Safety Function: Key Containment Parameters	
Method: Containment Pressure	This method is not applicable to the NuScale Power Plant, because containment pressure and temperature control are passively provided.
Purpose: Monitor long-term pressure buildup in containment.	The NEI recommended parameter of Containment Pressure is to prompt the operators to take the action of spraying containment to reduce pressure. For the NuScale Power Plant, following ECCS actuation CNV pressure lowers as heat is passively transferred to the UHS. This function is ensured when UHS level is at or above {{ }}[2(a),(c)].
Performance Attribute: • Identify instruments to be relied upon, including control room and field instruments.	Evaluation of Performance Attribute: • This attribute is met. The UHS system includes four level instruments that meet the requirements of NEI 12-02. Level indication is available from the PPS through the SDIS displays in the control room, and locally in the field.

Table B-3 Evaluation of the NEI 12-06 approach to pressurized water reactor spent fuel pool cooling functions

NEI 12-06 Recommendation	NuScale Power Plant Design
Safety Functions: Spent Fuel Cooling	
Method: Makeup with FLEX Injection Source - Makeup via hoses on refuel floor.	This method is not needed for the NuScale Power Plant design because the reactor and RFPs passively provide makeup to the SFP for more than {{ }}[2(a),(c)] following initiation of an ELAP.
Purpose: Exceed SFP boil-off to support long-term cooling of spent fuel with sufficient makeup.	The purpose of this method, to ensure sufficient SFP inventory exists to cool the spent fuel, is inherent to the NuScale Power Plant design.
Performance Attribute: • Minimum makeup rate must be capable of exceeding boil-off rate for the boundary conditions described in Section 4.2.6.	Evaluation of Performance Attribute: • Although this attribute is explicitly not met, the makeup rate of water to the SFP from the other pools of the UHS is sufficient to ensure spent fuel cooling is provided by submergence for more than {{ }}[2(a),(c)] without operator action to add inventory. This design element meets the intended purpose of the attribute.
Methods Summary: The specific method recommended is not needed for the NuScale Power Plant, but the intended purpose, to ensure sufficient SFP inventory exists to cool the spent fuel during an ELAP, is inherent to the design. The intended purpose of the performance attribute is met.	
Method: Makeup with FLEX Injection Source - Makeup via connection to SFP cooling piping or other alternate location.	This method is specifically provided for in the NuScale Power Plant design and included in the FLEX strategy.
Purpose: Exceed SFP boil-off and provide a means to supply SFP makeup without accessing the refueling floor.	This is the purpose of the UHS assured makeup line to the SFP, which includes a connection external to the RXB, and permits inventory addition without accessing the SFP area.

NEI 12-06 Recommendation	NuScale Power Plant Design
Performance Attribute:	Evaluation of Performance Attribute:
• Minimum makeup rate must be capable of exceeding boil-off rate for the boundary conditions described in Section 4.2.6.	• This attribute is met by the assured makeup line, which is capable of gravity feeding water at a minimum rate of 100 gpm.
Method Summary: This method is specifically provided for in the NuScale Power Plant design and included in the FLEX strategy. The associated performance attribute is met.	
Method: Makeup with FLEX Injection Source - Vent pathway for steam & condensate from SFP.	This method is not applicable to the NuScale Power Plant design, because access to the SFP area is not necessary, and the equipment in the pool area needed for coping is designed for the steam and condensate environment.
Purpose: Steam from boiling pool can condense and cause access and equipment problems in other parts of plant.	Not applicable. Access to the SFP area is not necessary, and the equipment in the pool area needed for coping is designed for the steam and condensate environment.
Performance Attribute: • Plant-specific strategy should be considered as needed.	Evaluation of Performance Attribute: • This performance attribute has been considered. Although not needed or credited for coping, the RBVS SFP ductwork provides a passive HEPA-filtered exhaust path by automatic system realignment on a loss of power.
Method Summary: This method is not applicable to the NuScale Power Plant design, because access to the SFP area is not necessary, and the equipment in the pool area needed for coping is designed for the steam and condensate environment. The performance attribute has been considered, as suggested.	
Safety Function: SFP Parameters	
Method: SFP Level	This method is specifically provided for in the NuScale Power Plant design and included in the FLEX strategy.
Purpose: Confirm SFP level is adequate to provide cooling.	This is the purpose of the SFP level instruments.

NEI 12-06 Recommendation	NuScale Power Plant Design
Performance Attribute:	Evaluation of Performance Attribute:
• Wide-range spent fuel pool level instruments.	• The attribute is met. The UHS system includes four level instruments that meet the guidance of NEI 12-02. Level indication is available from the PPS through the SDI system displays in the control room, and locally in the field.
Method Summary: This method is specifically provided for in the NuScale Power Plant design and included in the FLEX strategy. The associated performance attribute is met.	