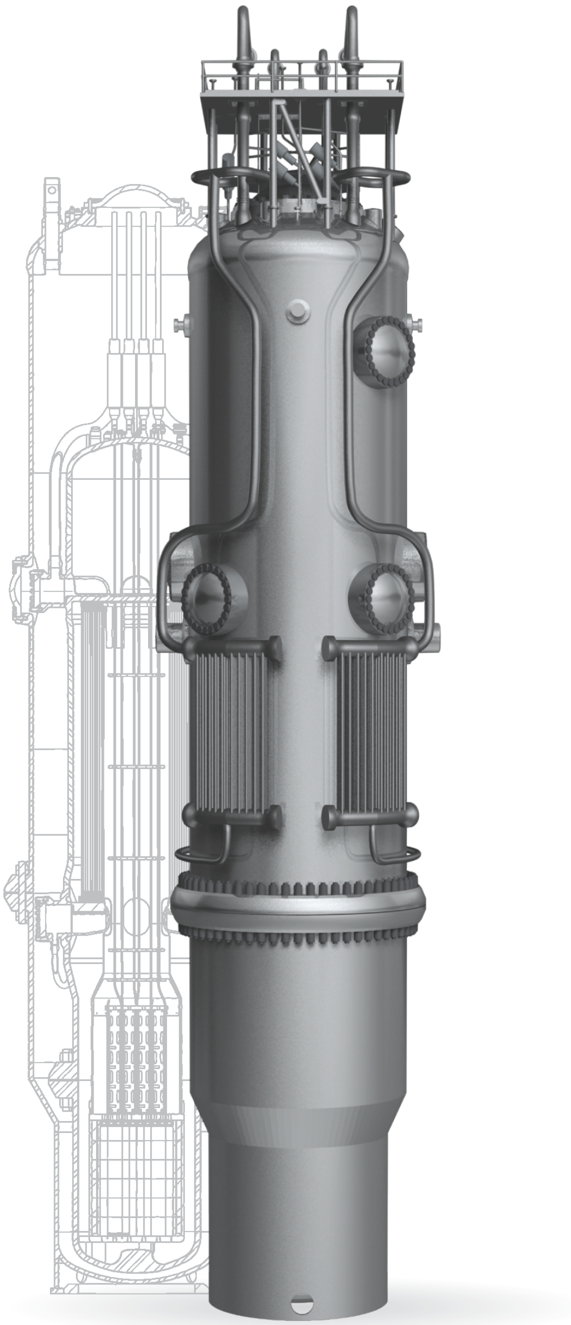

NuScale Standard Plant
Design Certification Application

Chapter Twenty
Mitigation of Beyond-Design-Basis Events

PART 2 - TIER 2

Revision 0
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TABLE OF CONTENTS

CHAPTER 20 MITIGATION OF BEYOND-DESIGN-BASIS EVENTS 20.1-1

20.1 Mitigating Strategies for Beyond Design-Basis External Events 20.1-1

20.1.1 Determining Applicable Extreme External Hazards 20.1-2

20.1.2 Extended Loss of AC Power and Loss of Ultimate Heat Sink Design Assessment 20.1-3

20.1.3 Mitigating Strategies for an Extended Loss of AC Power Event 20.1-5

20.1.4 Spent Fuel Pool and Reactor Pool Level Instrumentation 20.1-7

20.1.5 References 20.1-9

20.2 Loss of Large Areas of the Plant due to Explosions and Fires 20.2-1

20.2.1 Phase 1 - Enhanced Fire Fighting Capabilities 20.2-1

20.2.2 Phase 2 - Measures to Mitigate Damage to Fuel in the Spent Fuel Pool 20.2-2

20.2.3 Phase 3 - Measures to Mitigate Damage to Fuel in the Reactor Vessel and to Minimize Radiological Release 20.2-2

20.2.4 References 20.2-2

20.3 Integration with Emergency Procedures 20.3-1

20.4 Enhanced Emergency Response Capabilities for Beyond-Design-Basis Events 20.4-1

20.4.1 Enhanced Emergency Plan Staffing 20.4-1

20.4.2 Enhanced Emergency Plan Communications 20.4-1

20.4.3 Enhanced Emergency Plan Training, Drills, and Exercises 20.4-1

20.4.4 Multi-Unit Multi-Source Dose Assessment Capability 20.4-2

LIST OF TABLES

Table 20.1-1: Core Cooling Parameters 20.1-11
Table 20.1-2: Containment Parameters 20.1-12
Table 20.1-3: Spent Fuel Pool Parameters 20.1-13

CHAPTER 20 MITIGATION OF BEYOND-DESIGN-BASIS EVENTS

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 Near-Term Task Force (NTTF). The NTTF conducted a systematic and methodical review of the NRC regulations and processes to determine if the agency should make safety improvements in light of the events in Japan. As a result of this review, the NRC issued SECY-11-0093 (Reference 20.1-1). SECY-11-0124 (Reference 20.1-2), and SECY-11-0137 (Reference 20.1-3) were issued to establish the NRC staff's recommendations and prioritization of the recommendations.

As a result of NRC's involvement with stakeholders and nuclear industry representatives, the following guidance documents were published:

- NEI 12-01, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities" (Reference 20.1-4)
- 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 20.1-5)
- NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (Reference 20.1-6)
- NEI 13-06, "Enhancements to Emergency Response Capabilities for Beyond Design Basis Events and Severe Accidents" (Reference 20.1-7)
- NEI 14-01, "Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents" (Reference 20.1-8)

Also as part of the NTTF recommendations, rulemaking is in progress (Reference 20.1-9)

The purpose of this chapter is to describe the NuScale Power Plant response to beyond design basis external events (BDBEE). Although the proposed regulation is in rulemaking, as published in the Federal Register on November 13, 2015 (Reference 20.1-9), it is used as guidance for this chapter. The latest revisions of the applicable guidance, as of the writing of this chapter, are used in review for this chapter. Some guidance has not been endorsed by the NRC.

Section 20.1 addresses the diverse and flexible coping strategies (FLEX) strategies, Section 20.2 addresses loss of large area of the plant due to fire or explosion, Section 20.3 addresses procedure integration, and Section 20.4 addresses emergency response.

20.1 Mitigating Strategies for Beyond Design-Basis External Events

This section discusses the mitigating strategies that address an extended loss of alternating current power (ELAP) and loss of normal access to the ultimate heat sink (LUHS) resulting from a BDBEE.

NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, Revision 2 (Reference 20.1-6) is the NRC endorsed guidance for developing diverse and flexible coping strategies (FLEX).

20.1.1 Determining Applicable Extreme External Hazards

FLEX equipment credited in the mitigation strategies is required to be available following a BDBEE. The extreme external hazards required to be considered for a BDBEE are seismic, flooding, high winds (including applicable missiles), snow, ice, and extreme (cold and high) temperatures. Descriptions of the external hazards design criteria are provided in the following sections.

20.1.1.1 Seismic

The seismic design criteria are identified in Section 3.7.1.

COL Item 20.1-1: A COL applicant that references the NuScale Power Plant design certification will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific seismic hazard.

20.1.1.2 External Flooding

The external flood design criteria are identified in Section 3.4.2.

COL Item 20.1-2: A COL applicant that references the NuScale Power Plant design certification will determine if a flood hazard is applicable at the site location. If a flood hazard is applicable, then the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific flood (including wave action) hazard.

20.1.1.3 High Winds / Missile Protection

The high winds (hurricane and tornado) and applicable missile design criteria are identified in Section 3.3 and Section 3.5.

COL Item 20.1-3: A COL applicant that references the NuScale Power Plant design certification will determine if high wind and applicable missile hazards are applicable at the site location. If high wind and applicable missile hazards are applicable, then the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific high wind and applicable missile hazards.

20.1.1.4 Snow, Ice, and Extreme Cold

The snow and ice design criteria are identified in Section 3.8.4. The minimum (i.e., extreme cold) design temperature is identified in Table 2.0-1.

COL Item 20.1-4: A COL applicant that references the NuScale Power Plant design certification will determine if snow, ice and extreme cold temperature hazards are applicable at the site location. If snow, ice and extreme cold hazards are applicable, the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific snow, ice or extreme cold temperature hazard.

20.1.1.5 High Temperatures

The maximum (i.e. high temperature) design temperature is identified in Table 2.0-1

COL Item 20.1-5: A COL applicant that references the NuScale Power Plant design certification will determine if extreme high temperature hazard is applicable at the site location. If extreme high temperature hazard is applicable, the COL applicant will ensure equipment and structures credited for FLEX strategies are designed to be available following a site-specific extreme high temperature hazard.

20.1.2 Extended Loss of AC Power and Loss of Ultimate Heat Sink Design Assessment

This section discusses the inherent coping capability of the NuScale Power Plant design to maintain the key safety functions following an ELAP and an LUHS event. The key safety functions are maintaining core cooling, containment and spent fuel pool cooling.

20.1.2.1 Definitions

An ELAP event is defined as a loss of all alternating current (AC) electric power to the essential and nonessential switchgear buses except for those fed by qualified DC batteries through inverters.

NEI 12-06 (Reference 20.1-6) defines an LUHS as the loss of motive force for UHS flow, i.e., service water or circulating water pumps, with no prospect for recovery. The LUHS event assumes the water inventory in the UHS remains available following the event, and the piping connecting the UHS to plant systems, which are qualified to survive the applicable external hazards, remains intact.

NEI 12-06 defines the following three phases for developing FLEX strategies (Reference 20.1-6):

- Phase 1 (initial): cope relying on plant equipment
- Phase 2 (transition): augment or transition from plant equipment to on-site FLEX equipment and consumables to maintain or restore key functions
- Phase 3 (final): obtain additional capability and redundancy from off-site equipment until power, water, and coolant injection systems are restored or commissioned

20.1.2.2 Applicable Systems, Structures, and Components

The UHS is a large pool of water consisting of the combined water volumes of the reactor pool (RP), refueling pool (RFP) and spent fuel pool (SFP), as described in Section 9.2.5. Each NuScale Power Module (NPM) is partially immersed in the UHS. Since the NPMs are partially immersed and in direct contact with the UHS, there is no need for UHS piping or motive force such as service water or circulating water pumps to provide cooling functions following BDBEE. Therefore, the loss of the UHS, as defined in NEI 12-06 (Reference 20.1-6), is not plausible. No other heat sink is credited for maintaining the key safety functions.

The NPM is a self-contained nuclear steam supply system composed of a reactor core, a pressurizer, and two steam generators integrated within the reactor pressure vessel (RPV) and housed in a compact steel containment vessel (CNV). The plant design relies on passive systems for core cooling during a loss of AC power or DC power.

The reactor coolant system (RCS) design does not require inventory makeup following an ELAP event, but instead relies on maintaining the coolant inventory contained in the RPV and CNV, as described below.

The containment isolation valves (CIVs), which isolate the CNV, fail-safe to their closed position using stored energy. A discussion of the design and function of the CIVs is described in Section 6.2.4.

The decay heat removal system (DHRS) actuation valves fail-safe to the open position, and the main steam isolation valves (MSIVs) and feedwater isolation valves (FWIVs) fail-safe to the closed position to place the DHRS passive condensers in service. These valves fail-safe using stored energy and do not require any operator actions or electric power to perform this function. Once a DHRS passive condenser is in service, a closed natural circulation loop is established transferring core decay heat and sensible heat to the UHS. DHRS is further described in Section 5.4.3.

Emergency Core Cooling System (ECCS) valves are also designed fail-safe to their open position. The ECCS valves fail-safe using the differential pressure between the RPV and CNV and do not require any operator actions or electric power to perform this function. The function of the ECCS is further described in Section 6.3. With the ECCS valves open and the CIVs closed, a closed natural circulation loop is established in which decay heat is transferred from the core to the UHS. The containment heat removal capability of the CNV is described in Section 6.2.2, and the natural circulation process after ECCS initiation is described in Section 6.3. Once natural circulation is established, core cooling and containment integrity are assured by maintaining sufficient inventory in the UHS. Sufficient inventory to maintain core cooling and containment integrity is available in the UHS for greater than 30 days (Reference 20.1-10) without the need for operator actions.

Instrumentation and its associated indications remain available as needed to confirm proper CIV positions and verify that the natural circulation passive cooling is established. Once the valve positions are verified to be in the proper position and key parameters indicate that passive natural circulation is occurring, the only key parameters that require monitoring for the duration of the event are the UHS and SFP level when below the weir.

The UHS and SFP are monitored using four reliable independent level instruments that are designed to the augmented quality requirements specified in NEI 12-02 (Reference 20.1-5). Two level instruments are positioned in the SFP area, one level instrument in the RFP area, and one level instrument is positioned in the reactor pool area. All four instruments monitor the UHS level until the UHS level decreases below the SFP weir when the RFP and reactor pool inventory is separate from the inventory in the SFP.

A robust makeup line with an external connection point for providing inventory to the SFP is available to support SFP makeup following a BDBEE. The makeup to the SFP will overflow into the UHS when level reaches the weir. The makeup line is sized to provide at least 100 gpm of gravity-fed makeup inventory to the SFP. The 100 gpm makeup is greater than the UHS boil off rate. The SFP makeup line is discussed in Section 9.2.5.

During hot shutdown, safe shutdown, transition, and refueling conditions the NPM remains partially immersed in the UHS. The DHRS and ECCS are available or are in service to support core cooling.

Prior to entering the transition mode, the CNV is flooded and the ECCS valves are opened. The NPM remains partially immersed in the UHS while being transported to the RFP. In the RFP the upper module, lower module, and RPV are disassembled while immersed in the UHS.

The reactor core is submerged in water either via the CNV being flooded or directly in the UHS, while in refueling, through the duration of the event. Therefore, passive core cooling remains available in shutdown and refueling conditions.

The SFP will begin to boil no earlier than five days after the start of the ELAP event. The UHS is sized such that sufficient inventory is available to provide spent fuel cooling for greater than 30 days (Section 9.2.5).

The two level instruments located in the SFP meet the guidance in accordance with NEI 12-02 (Reference 20.1-5). See Section 20.1.4 for more detail on the SFP level instruments.

20.1.3 Mitigating Strategies for an Extended Loss of AC Power Event

A summary of the three phases as defined above for NuScale's FLEX strategies is provided in the following sections. The key parameters that are monitored for the FLEX strategies are summarized in Table 20.1-1, Table 20.1-2 and Table 20.1-3.

20.1.3.1 Phase 1

As described in Section 20.1.2, the key safety functions are maintained for greater than 30 days with installed plant equipment. No operator actions or supplemental equipment are necessary to perform these functions.

Core Cooling

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

- Decay heat removal system (DHRS) actuation valves open to establish natural circulation flow and commence the transfer of reactor coolant system (RCS) heat to the fluid contained in the passive condenser loops.
- The DHRS passive condensers are submerged in the ultimate heat sink (UHS), and transfer their heat to the UHS.

- The containment isolation valves close to maintain RCS inventory.
- Emergency core cooling system (ECCS) valves open to establish natural circulation flow of reactor coolant between the reactor pressure vessel and the CNV. The CNV is partially immersed in the UHS, and transfers heat passively to the UHS.

Maintain Containment

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

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

Spent Fuel Pool Cooling

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

20.1.3.2 Phase 2

A FLEX strategy for a transition phase is not needed for the NuScale Power Plant design. The coping time utilizing installed plant equipment is greater than 72 hours. The initial phase is of sufficient duration to transition directly to the final phase.

20.1.3.3 Phase 3

The baseline coping capability utilizing installed plant equipment is greater than 30 days, and therefore, immediate actions after 72 hours are not necessary. Sufficient time is available for detailed planning and procurement of offsite equipment, if necessary to maintain the key safety functions. Due to this extended baseline coping capability, the Phase 3 FLEX strategy is to monitor UHS pool level, utilizing the level instruments described in Section 20.1.4, and add inventory to the UHS via the SFP assured makeup line, if necessary.

COL Item 20.1-6: A COL applicant that references the NuScale Power Plant design certification will develop and implement the strategies and guidance for makeup to the UHS after an ELAP event utilizing supplemental FLEX equipment.

COL Item 20.1-7: A COL applicant that references the NuScale Power Plant design certification will develop a training and qualification program using the systems approach to training process. The training will ensure personnel will be able to perform activities in accordance with the FLEX strategies and guidelines.

20.1.4 Spent Fuel Pool and Reactor Pool Level Instrumentation

20.1.4.1 Design Bases

The design of the four (4) UHS level instruments meet the guidance of NEI 12-02. The design basis functions of the pool level instrumentation are to provide plant personnel with a reliable wide-range water level indication of the UHS level until the UHS level decreases below the SFP weir when the RFP and reactor pool inventory is separate from the inventory in the SFP and reactor pool relative to the following water levels:

- Level 1 - level that is adequate to support operation of the normal pool cooling systems,
- Level 2 - level that is adequate to provide substantial radiation shielding for a person standing on the operating deck,
- Level 3 - level where stored fuel remains covered and actions to implement make-up water should no longer be deferred.

During refueling, the NPM is disassembled in the RFP area to allow transferring of new and spent fuel to and from the reactor core. When an NPM is disassembled, the water level in the RFP area will be monitored to ensure the fuel in the reactor is covered during an ELAP event.

The UHS level instruments are designed to withstand external hazards; such as seismic, flooding, high winds (including applicable missiles) extreme temperatures, and snow and ice; without loss of capability to perform their monitoring function.

The UHS instruments are designed to withstand the effects of and to be compatible with the environmental conditions associated with the expected conditions in the Reactor Building during normal operations and an ELAP event.

The UHS level instruments and their power supplies are physically and electrically separated and independent.

20.1.4.2 Description

Two (2) wide-range water level instruments are provided for both the spent fuel pool and the reactor pool, for a total of four (4) instruments. The wide-range instruments encompass the elevations from the top of the fuel storage racks to near the operating deck. The pool water level instrumentation is consistent with the guidance of NEI 12-02, Revision 1.

Instruments

Two (2) permanent wide-range instruments monitor the level of the SFP. Two permanent wide-range instruments are also installed to monitor the UHS water level, one in the RP area and one in the RFP area. The instruments transmit signals to the main control room. All four of these instruments are capable of monitoring Levels 1 and 2. The two (2) SFP level instruments are capable of monitoring Level 3, when the RP water level is below the weir wall elevation. The two (2) level instruments in the RP and

RFP areas are capable of monitoring the level of the water above the fuel in the reactor core when the NPM is disassembled in the RFP during refueling.

Arrangement

The two (2) SFP level instruments are arranged in opposite corners of the SFP. The two (2) UHS level instruments are located in the one (1) in the RFP area and the other instrument is located in the RP area. The separation of these instruments is adequate to provide protection from missiles that may be generated within the reactor building from affecting all instruments. Protection from external missiles is provided by the Reactor Building structure.

The instrument cables are also separated to provide protection against a single missile damaging both trains.

Mounting

The four (4) UHS instruments are seismically mounted such that the instruments will maintain their design configuration during and following an SSE (Seismic Category I).

Qualification

The four (4) UHS instruments and associated cabling are environmentally qualified to operate following a BDBEE in the following environmental conditions:

- SSE seismic event (Seismic Category I)
- Concentrated borated water environment,
- Temperature of approximately 212 degrees Fahrenheit and 100 percent relative humidity,
- Boiling water or steam environment,
- Radiological conditions existing from a normal refueling with a freshly discharged fuel batch that remains covered with SFP water (Level 3).

Independence

The four (4) UHS level instruments are both physically and electrically independent.

Power Supplies

The power to the four (4) UHS level instruments is supplied by the highly reliable DC power system (EDSS) with interface through the plant protection system (PPS). Power to the redundant level instruments is from separate bus sources such that the loss of one supply will not result in a loss of power supply function to both divisions of UHS level instrumentation. Additionally, a replaceable battery that is isolated from faults on the normal power supply provides an alternate source of power independent from the plant AC and DC power systems.

Accuracy

The instrument channels are designed to maintain the minimum accuracy following a power interruption or change in power source without recalibration.

Testing

The permanently installed UHS level instruments are designed such that testing and calibration can be performed in-situ.

Display

The four (4) UHS level instruments transmit signals to the main control room and the remote shutdown panel, and are immediately available to the operators following an event. The instrument signals also initiate high or low level alarms, both locally and in the main control room.

Programs

COL Item 20.1-8: A COL applicant that references the NuScale Power Plant design certification will develop procedures, training and qualification program for operations, maintenance, testing, and calibration of UHS level instrumentation to ensure the level instruments will be available when needed and personnel are knowledgeable in interpreting the information as addressed in NEI 12-02.

Safety Evaluation

The four (4) UHS level instruments are designed to withstand and be protected from natural phenomena such as earthquakes, tornados, hurricanes, floods, tsunamis and seiches without loss of function.

These instruments are also designed to accommodate or be protected from the effects of the postulated environmental conditions, including missiles, pipe whipping, and jet impingement.

The instruments and associated cabling is protected by both physical and electrical separation such that a failure in one channel will leave the other channel functional.

20.1.5 References

20.1-1 U.S. Nuclear Regulatory Commission, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan," SECY-11-0093, July 12, 2011.

20.1-2 U.S. Nuclear Regulatory Commission, "Recommended Actions to be Taken Without Delay from the Near-Term Task Force Report," SECY-11-0124, September 9, 2011.

20.1-3 U.S. Nuclear Regulatory Commission, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," SECY-11-0137, October 3, 2011.

20.1-4 Nuclear Energy Institute, "Guideline for Assessing Beyond Design Basis Accident Response Staffing and Communications Capabilities," NEI 12-01, Revision 0, May 2012.

20.1-5 NEI 12-02, Industry Guidance for Compliance with NRC Order EA-12-051, "To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," Revision 1.

20.1-6 Nuclear Energy Institute, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," NEI 12-06, Revision 2, December 2015.

20.1-7 Nuclear Energy Institute, "Enhancements to Emergency Response Capabilities for Beyond Design Basis Events and Severe Accidents," NEI 13-06, Revision 0, September 2014.

20.1-8 Nuclear Energy Institute, "Emergency Response Procedures and Guidelines for Beyond Design Basis Events and Severe Accidents," NEI 14-01, Revision 0, September 2014.

20.1-9 U.S. Nuclear Regulatory Commission, FRN Vol. 80, No. 219 pages 70610-701647, November 13, 2015.

20.1-10 NuScale Power LLC, Mitigation Strategies for Extended Loss of AC Power Event, TR-0816-50797, Revision 0, November 2016.

Table 20.1-1: Core Cooling Parameters

Function	Parameters for Assuring Function is Established	Parameters for Assuring 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	
	Reactor trip breaker status	
	CVCS containment isolation valve positions	
DHRS decay heat removal	Core exit temperature	Spent fuel pool level(2)
	DHRS valve positions	
	MSIV positions	
	MSIV bypass isolation valve positions	
	FWIV positions	
	Spent fuel pool level(2)	
ECCS decay heat removal	ECCS valve positions	Spent fuel pool level(2)
	Containment water level	
	RPV water level	
	Core exit temperature	
	Spent fuel pool 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.

Table 20.1-2: Containment Parameters

Function	Parameters for Assuring Function is Established	Parameters for Assuring Function is Maintained
Containment isolation	Containment isolation valve positions	None(1)
Containment heat removal	Wide range containment pressure	Spent fuel pool level(2)
	Spent fuel pool 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.

Table 20.1-3: Spent Fuel Pool Parameters

Function	Parameter for Assuring Function is Established	Parameter for Assuring Function is Maintained
Spent Fuel Pool Cooling	Spent fuel pool level	Spent fuel pool level

20.2 Loss of Large Areas of the Plant due to Explosions and Fires

NRC regulation 10 CFR 50.54(hh)(2)[1] requires licensees to develop and implement guidance and strategies to maintain or restore core cooling, containment, and spent fuel cooling under the circumstance associated with the loss of large areas of the plant due to explosion or fire (LOLA). The strategies that are required to be addressed are: (i) fire fighting; (ii) operations to mitigate fuel damage; and (iii) actions to minimize the radiological release.

Technical Report TR-0816-50796 (Reference 20.2-1) documents an assessment evaluating the NuScale Power Plant response to a LOLA event using the guidance in Nuclear Energy Institute (NEI) 06-12 (Reference 20.2-2). The report defines LOLA criteria and identifies the design features that meet those criteria and expected combined license (COL) applicant requirements.

The analysis was performed using the three phases recommended in NEI 06-12 (Reference 20.2-2): Phase 1 - Enhanced Fire Fighting Capabilities; Phase 2 - Measures to Mitigate Damage to Fuel in the Spent Fuel Pool; and Phase 3 - Measures to Mitigate Damage to Fuel in the Reactor Vessel and to Minimize Radiological Release.

This section describes the results of the assessment with no Security Related Information.

20.2.1 Phase 1 - Enhanced Fire Fighting Capabilities

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

The fire protection system includes an underground yard fire main loop. Hydrants are provided on the yard fire main loop in accordance with the National Fire Protection Association at intervals up to 250 feet and located on all four sides of the Reactor Building. The lateral to each hydrant is controlled by an isolation valve. There are several connections in the yard main that can support supplying the yard main using a portable diesel-driven pump and several valves that can isolate damaged section(s) when required. The fire protection system is described in more detail in Section 9.5.1.

COL Item 20.2-1: A COL applicant that references the NuScale Power design certification will develop enhanced firefighting capabilities by implementing the guidance in NRC guidance document "Developing Mitigating Strategies/Guidance for Nuclear Power Plants to Respond to Loss of Large Areas of the Plant in Accordance with B.5.b of the February 25, 2002, Order" dated February 25, 2005 (Reference 20.2-3). The enhanced firefighting capabilities should address the expectation elements listed in Section 4.1.3 of the Technical Report TR-0816-50796 (Reference 20.2-1).

1. 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

20.2.2 Phase 2 - Measures to Mitigate Damage to Fuel in the Spent Fuel Pool

No additional spent fuel cooling strategies are required, in accordance with the guidance in NEI 06-12 (Reference 20.2-2).

20.2.3 Phase 3 - Measures to Mitigate Damage to Fuel in the Reactor Vessel and to Minimize Radiological Release

The generic pressurized water reactor key safety functions identified in NEI 06-12 (Reference 20.2-2) were developed based on a traditional pressurized water reactor plant design. These key safety functions are applicable to the NuScale Power Plant and no new key safety functions are identified. The key safety functions that must be evaluated for a LOLA event are:

- reactor coolant system (RCS) inventory control
- RCS heat removal
- containment isolation
- containment integrity
- release mitigation

A primary and alternate means for RCS inventory control, RCS heat removal, containment isolation, and containment integrity are maintained for the NuScale Power Plant with installed plant capabilities and no additional mitigating strategies are necessary. An additional means or strategy of minimizing a potential radiological release from a LOLA event is needed to augment the NuScale Power Plant installed plant capabilities.

COL Item 20.2-2: A COL applicant that references the NuScale Power design certification will provide a means for water spray scrubbing using fog nozzles and the availability of water sources, and address runoff water containment issues (sandbags, portable dikes, etc.) as an attenuation measure for mitigating radiation releases outside containment.

20.2.4 References

20.2-1 NuScale Power, LLC, "Loss of Large Areas Due to Explosions and Fires Assessment," TR-0816-50796 (Security Related Information).

20.2-2 Nuclear Energy Institute, "B.5.b Phase 2 & 3 Submittal Guideline," NEI 06-12, Revision 3, September 2009.

20.2-3 U.S. Nuclear Regulatory Commission, “Developing Mitigating Strategies/ Guidance for Nuclear Power Plants to Respond to Loss of Large Areas of the Plant in Accordance with B.5.b of the February 25, 2002, Order,” February 25, 2005.

20.3 Integration with Emergency Procedures

COL Item 20.3-1: A COL applicant that references the NuScale Power Plant design certification will ensure that the severe accident management guidelines, FLEX support guidelines, and extensive damage mitigation guidelines are integrated with the emergency operating procedures consistent with Recommendation 8.1 of SECY-11-0093, "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan."

20.4 Enhanced Emergency Response Capabilities for Beyond-Design-Basis Events

This section describes the provisions implemented to enhance the emergency response capabilities as they relate to emergency response staffing, communication, training, drills, exercises, and multi-source dose assessment capabilities for beyond-design-basis events.

A description of the emergency plan is provided in Section 13.3.

The beyond-design-basis event (BDBE) emergency response enhancements of Section 20.4.1, Section 20.4.2, and Section 20.4.3 are not required to be part of the plant's emergency plan, and therefore do not require the change control provisions of 10 CFR 50.54(q).

20.4.1 Enhanced Emergency Plan Staffing

COL Item 20.4-1: A COL applicant that references the NuScale Power Plant design certification will perform an analysis that demonstrates the Emergency Response Organization staff has the 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.

COL Item 20.4-2: 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.

20.4.2 Enhanced Emergency Plan Communications

The installed plant communication capabilities are described in Section 9.5.2.

COL Item 20.4-3: A COL applicant that references the NuScale Power Plant design certification will develop and describe at least one onsite and one offsite communications system capable of remaining functional during an extended loss of alternating current power including the effects of the loss of the local communications infrastructure.

20.4.3 Enhanced Emergency Plan Training, Drills, and Exercises

COL Item 20.4-4: 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.

COL Item 20.4-5: 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.

20.4.4 Multi-Unit Multi-Source Dose Assessment Capability

COL Item 20.4-6: A COL applicant that references the NuScale Power Plant design certification will develop and describe the means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials to the environment including releases from all reactor core and spent fuel pool sources.