May 31, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Technical Report "NuScale Generic Technical Guidelines," TR-1117-57216, Revision 1.

REFERENCE: **1.** Letter from NuScale Power, LLC to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Technical Report "NuScale Generic Technical Guidelines," TR-1117-57216 ," dated November 30, 2017 (ML17334B822)

NuScale Power, LLC (NuScale) has submitted Revision 0 of the "NuScale Generic Technical Guidelines" (Reference 1). The purpose of this letter is to provide NuScale Technical Report "NuScale Generic Technical Guidelines," TR-1117-57216, Revision 1. This technical report will be incorporated by reference in a future revision of the NuScale Final Safety Analysis Report.

Enclosure 1 contains the proprietary version of "NuScale Generic Technical Guidelines," TR-1117-57216, Revision 1. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version of "NuScale Generic Technical Guidelines," TR-1117-57216, Revision 1.

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

If you have any questions, please feel free to contact Nadja Joergensen at 541-452-7338 or at njoergensen@nuscalepower.com.

Sincerely,

Thomas A. Bergman
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Distribution: Samuel Lee, NRC, OWFN-8H12
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Prosanta Chowdhury, NRC, OWFN-7E11

Enclosure 1: "NuScale Generic Technical Guidelines," TR-1117-57216-P, Revision 1, proprietary version
Enclosure 2: "NuScale Generic Technical Guidelines," TR-1117-57216-NP, Revision 1, nonproprietary version
Enclosure 3: Affidavit of Thomas A. Bergman, AF-0519-65790

Enclosure 1:

“NuScale Generic Technical Guidelines,” TR-1117-57216-P, Revision 1, proprietary version

Enclosure 2:

"NuScale Generic Technical Guidelines," TR-1117-57216-NP, Revision 1, nonproprietary version

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Licensing Technical Report

NuScale Generic Technical Guidelines

May 2019

Revision 1

Docket: 52-048

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Department of Energy Acknowledgement and Disclaimer

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Abstract

This document describes the generic guidance provided to emergency procedure writers to develop procedures used by plant operators to ensure plant safety during transient and accident conditions. This guidance covers the content provided in existing industry procedure classes including: emergency operating procedures, severe accident management guidelines, extended loss of AC power and loss of large area guidance.

Executive Summary

TMI Action Plan Item I.C.1, a post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the emergency operating procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with Clarification of TMI Action Plan Requirements, NUREG-0737, and Supplement 1, Requirements for Emergency Response Capability, NUREG-0737, that also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7 is an acceptance criterion in Operating and Emergency Operating Procedures, SRP Chapter 13.5.2.1, NUREG-0800. Design-specific generic technical guidelines (GTGs) will be used by combined license applicants to develop their plant-specific technical guidelines, from which their EOPs will be developed. The GTGs are described in Guidelines for the Preparation of Emergency Operating Procedures, NUREG-0899. The GTGs are documents that identify the equipment or systems to be operated and list the steps necessary to mitigate the consequences of transients and accidents and restore safety functions. The GTGs represent the translation of engineering data derived from transient and accident analyses and probabilistic risk assessment (PRA) into information presented in such a way that it can be used to write EOPs.

The NuScale Power, LLC plant design is both simple and passive with limited manual actions required to ensure nuclear safety. There are no operator actions credited in the evaluation of NuScale design-basis accidents. All of the analyzed operator actions are used to mitigate beyond-design-basis events.

This document describes the NuScale generic technical guidance, including:

1. the process used to develop GTGs.
2. the assumptions used during development.
3. a description of the supportive analysis.
4. a description of the validation process.

1.0 Introduction

1.1 Purpose

This document describes the generic guidance provided to the emergency procedure writers to develop procedures used by plant operators to ensure plant safety during transient and accident conditions. This guidance covers the content provided in existing industry operating procedure classes including: emergency operating procedures, severe accident management guidelines, extended loss of AC power and loss of large area guidance.

1.2 Scope

This document describes the actions assumed within analysis of accident conditions that occur outside the design basis. The NuScale transient and accident analyses conducted for design-basis events (DBEs) do not identify any required operator actions. The generic technical guidelines (GTGs) contain those actions that have been credited by probabilistic risk assessment (PRA) and described in the final safety analysis report (FSAR) to respond to beyond-design events such as severe accidents, extended loss of AC power, and loss of large area. Additionally, some actions have been included solely based on task analysis performed as part of the human factors engineering program. This document is intended to provide guidance and basis for the emergency operating procedures but does not describe actions to the level of detail expected for a final operating procedure.

1.3 Abbreviations

Table 1-1 Abbreviations

Term	Definition
AAPS	auxiliary AC power source
ATWS	anticipated transient without scram
BDG	backup diesel generator
CET	core exit thermocouple
CFDS	containment flood and drain system
CHLA	candidate high-level actions
CIS	containment isolation signal
CIV	containment isolation valve
CNV	containment vessel
CVCI	chemical volume control isolation
CVCS	chemical and volume control system
DBE	design-basis event
DHRS	decay heat removal system
DWS	demineralized water system
ECCS	emergency core cooling system
EDSS	highly reliable DC power system
ELAP	extended loss of AC power

Term	Definition
ELVS	electrical low voltage system
EOP	emergency operating procedure
ESFAS	engineered safety features actuation system
FSAR	final safety analysis report
GTG	generic technical guideline
HSI	human-system interface
I&C	instrumentation and controls
IHA	important human action
ISV	integrated system validation
LOCA	loss of coolant accident
LOCA-IC	loss of coolant accident inside containment
LOCA-OC	loss of coolant accident outside containment (containment bypassed)
LOLA	loss of large area
MCC	motor control center
MCS	module control system
MPS	module protection system
PCS	plant control system
PRA	probabilistic risk assessment
PZR	pressurizer
RCS	reactor coolant system
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RTS	reactor trip system
RVV	reactor vent valve
SDI	safety display and indication
SGTF	steam generator tube failure
SME	subject matter expert
TAF	top of active fuel
TMI	Three Mile Island
UHS	ultimate heat sink
UTB	under the bioshield
WR	wide range

Table 1-2 Definitions

Term	**Definition**
defense-in-depth function	A function defined by the plant design that verifies and restores nonsafety related equipment to mitigate beyond-design-basis events, provides additional margin to safety system actuation or maintains parameters within normal operating ranges. The defense-in-depth functions are: (1) electrical, (2) reactor coolant system (RCS) makeup, (3) secondary heat removal, and (4) post trip actions.
design basis accident	A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures and components necessary to ensure public health and safety.
design basis event	Postulated events used in the design to establish the acceptable performance requirements for the structures, systems, and components.
safety function	A function defined by the plant design that maintains the unit in a safe shutdown condition and provides mitigating actions for accidents that cause core damage or release of radioactivity. The NuScale design has three safety functions defined: (1) maintain containment integrity, (2) reactivity control, and (3) remove fuel assembly heat.
maintain containment integrity	Design features used to prevent fission products from escaping the containment boundary. This function is referred to as “containment integrity” for ease of communication.
reactivity control	Design features used to control reactivity to maintain core and RCS integrity in order to prevent the release of radioactive material to the environment.
remove fuel assembly heat	Design features used to remove heat from the fuel assemblies via passive convection and conduction. This function maintains fuel integrity and prevents the release of radioactive material to the environment. This function is referred to as “core heat removal” for ease of communication.
Operations subject matter expert	A person who has completed the NuScale human factors engineering (HFE)/Operations initial company training program, has previous licensed operating nuclear plant experience, and has performed task analysis or NuScale system reviews so they are familiar with the NuScale plant design.
secondary heat sink	Use of the steam generators to accomplish core heat removal. Each of the two steam generators acts as a secondary heat sink to facilitate core heat removal. Heat removal can be accomplished by supplying feedwater to the steam generator where it is converted to steam and subsequently transferred to the circulating water system via the main condenser, or to the decay heat removal system (DHRS) where core heat is transferred to the reactor pool.

2.0 Background

The generic technical guidance is described in the Standard Review Plan, NUREG-0800, Chapter 13.5.2.1. The Standard Review Plan is the parent document providing guidance for the review of submittal documents and states that a procedure's generation package should be submitted. The procedure's generation package describes the applicants program for developing the emergency operating procedures as well as the content. Vendor-supplied generic technical guidance is developed to be used by applicants or licensees to develop emergency operating procedures.

The Three Mile Island accident in 1979 provided the impetus for renewed guidance and clarity on the development of emergency operating procedures. The nuclear steam supply system vendors at the time developed generic guidance that was then given to the plant owners to develop specific plant technical guidance. NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures, provides guidance that the U.S. Nuclear Regulatory Commission uses in evaluating whether an applicant or licensee meets the requirements for Emergency Operating Procedures of Title 10 Code of Federal Regulations, Part 50.34(b)(6)(ii).

The NuScale GTGs have been developed by referring to the content and format that legacy vendor guidelines have used. Traditionally, generic guidance has been arranged in four volumes: (1) the actual guidelines, (2) an annotated version of the guidelines with reference to a basis document, (3) the guidelines basis, and (4) an implementation guide. NuScale is presenting the GTGs by consolidating the content that would traditionally have been contained in Volumes 1 through 3 into a single volume. An implementation guide is not included with this current guidance, but may be developed later when a combined operating license applicant is identified.

Additionally, four events are typically described in past vendor-submitted generic technical guidance: (1) loss of coolant events, (2) steam generator tube rupture events, (3) loss of feed events, and (4) inadequate core cooling events. The NuScale generic technical guidance supports a symptom-based approach and is, therefore, not event specific.

3.0 Methodology

Multiple inputs are evaluated to determine the appropriate mitigating strategies and actions based on plant design. The following topics are reviewed to include any assumed operator actions:

1. The plant design bases response to DBEs as described in FSAR Chapter 15
2. The instrumentation and controls (I&C) failure defense-in-depth analysis as described in FSAR Chapter 7
3. The operator actions assumed in the beyond-design-basis PRA as described in FSAR Chapter 19
4. The operator actions assumed in beyond-design-basis event evaluation as described in FSAR Chapter 20
5. Multi-unit design considerations as described in FSAR Chapter 21
6. Plant technical specifications as described in Part 4 of the NuScale design certification application (DCA)
7. System requirements and limitations as defined in system description documents
8. HFE task analyses results as described in FSAR Chapter 18.4 and its associated reference

Single operator errors of both omission and commission were considered and analyzed in the transient and accident analysis of FSAR Chapter 15. The NuScale design limits operator error consequences to be less severe than the worst-case component single failure. The NuScale plant is a passive design that requires no operator action for 72 hours for any design basis event. Automated actions place and maintain the unit in a safe-state for at least 72 hours after a DBE even with assumed failures. Operator actions directed by procedure make the consequences of an event less severe, but failure to take one of these actions cannot make the consequences worse than the bounding FSAR Chapter 15 analysis.

Safety analysis of DBEs are performed from event initiation until a safe, stabilized condition is reached. A safe, stabilized condition is reached when the initiating event is mitigated, the FSAR Chapter 15 acceptance criteria are met, and system parameters are trending in the favorable direction. No operator action is required to reach or maintain a safe, stabilized condition for any DBE.

Instrumentation and controls diversity and defense-in-depth analysis documented in FSAR Chapter 7 did not result in any required operator actions.

Multiple operator errors or errors that result in common mode failures are beyond-design-basis and analyzed in the PRA.

The PRA assessment documented in FSAR Chapter 19 identified seven human actions to prevent core damage in beyond-design-basis accident sequences. These actions are listed in Section 4.4 of this report.

Additional evaluations for beyond design-basis events, performed and documented in FSAR Chapter 20, identified two human actions that would mitigate core damage once it has occurred or minimize the radioactive release. These actions are also listed in Section 4.4 of this report.

Other actions have been included within the generic technical guidelines that do not originate from analysis but promote consistent operator performance. For instance, the PRA analysis only credits manual initiation of emergency core cooling system (ECCS) or containment isolation signal (CIS) when those systems fail to actuate properly. Any failed engineered safety features actuation system (ESFAS) actuation is manually actuated to provide prudent and consistent guidance for operators.

4.0 Procedure Development

4.1 Symptom-Based Procedures

The guidelines have been developed using a symptom-based approach. Symptom based procedures are used to allow the operator to respond directly to the indications presented as part of the accident progression. Symptom-based procedures do not require the operator to diagnose the accident in progress. Symptom-based procedures allow the operator to respond to an event without knowledge of the initiating event or equipment status. These procedures also allow the operator to respond to unanticipated events, because they evaluate key parameters and direct actions to maintain them within the prescribed limits rather than responding in a predetermined sequence based on a diagnosed accident.

Legacy generic guidelines have included event-based descriptions. These events were based on the transient and accident analysis events and associated operator actions described in those designs' FSAR Chapter 15. Because the NuScale design has no FSAR Chapter 15 manual actions credited, the symptom-based approach allows for mitigating strategies to be effective with multiple failures regardless of the combination.

4.2 Critical Safety Functions

The evaluation of symptoms is grouped into critical safety functions for the NuScale plant design. Evaluation of the NuScale design, in addition to performing a comparison with traditional light water reactor safety functions, was used to determine the appropriate NuScale safety functions. These functions are accomplished by maintaining the following, listed in order of priority:

- containment integrity
- reactivity
- core heat removal

Additional safety functions are not needed due to the simplicity and reliance on passive systems in the NuScale design. For example, current operating pressurized-water reactors (PWRs) typically have a critical safety function of maintaining a secondary heat sink. Heat sink maintenance exists in other PWR designs because its loss can directly lead to core damage. Timely assessment and recovery or mitigation is critical to preventing core damage and, therefore, a separate critical safety function is warranted. For the NuScale design a loss of secondary heat sink (i.e., heat removal through the steam generators), by itself, does not result in core damage. Mitigation of a complete loss of secondary heat sink has been analyzed as part of PRA and, as such, is a best estimate analysis. This analysis demonstrates that loss of secondary heat sink is mitigated passively with the reactor safety valves transferring coolant to containment therefore establishing a conduction heat removal path through containment. The ECCS actuates as RCS level lowers and containment level rises to an actuation setpoint. The ECCS is also fully capable of removing decay heat in all required operating conditions.

RCS integrity is not a stand alone safety function as it is in other designs. The primary safety system that mitigates leaks of the RCS is the ECCS. When ECCS actuates, an intentional breach of the RPV integrity occurs which establishes a natural circulation heat removal path outside of the RCS but remaining within the containment. RCS integrity is therefore monitored indirectly through the core heat removal safety function.

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4.2.1 Containment Integrity

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4.2.2 Reactivity

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4.2.3 Core Heat Removal

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4.3 Structure and Use

The GTGs and associated basis are contained within Section 5.0 of this report. {{

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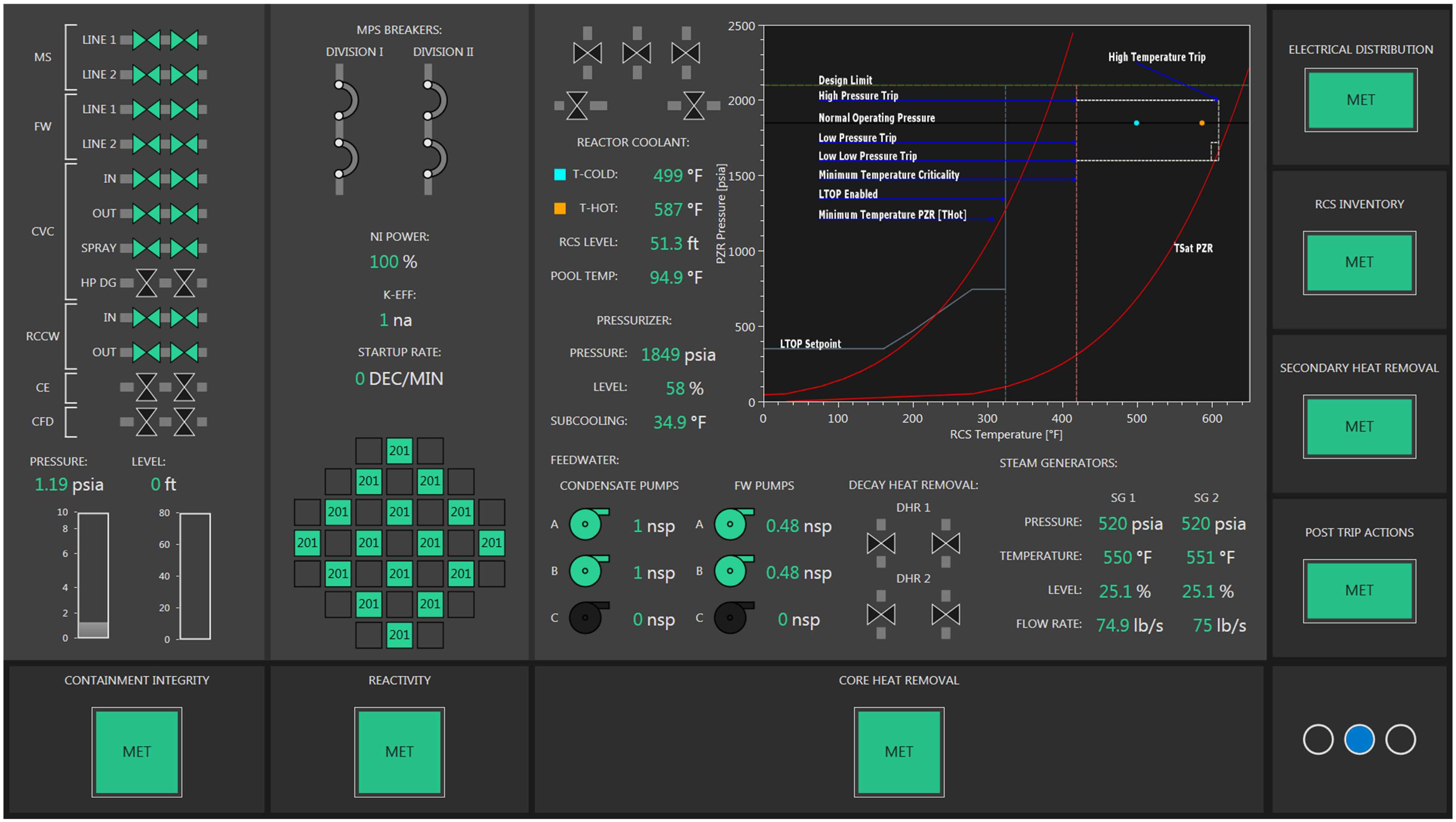

Figure 4-1 Example safety function status display on the HSI

4.4 Manual Actions

There are limited operator actions required in the NuScale design. All of the analyzed operator actions are used to mitigate beyond-design-basis events. There are no operator actions credited in the evaluation of NuScale DBEs. Automated actions place and maintain the unit in a safe state for at least 72 hours after a DBE even with assumed failures. Operator errors of both omission and commission were analyzed. Operator actions directed by guidance are intended to mitigate the consequences of the event. Failure to take action cannot make the consequences more severe than the bounding FSAR Chapter 15 analysis.

The I&C analysis of diversity and defense-in-depth identified no manual operator actions for a DBE.

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The PRA analysis examined beyond-design-basis events that lead to core damage and identified six human actions, including two that are important human actions (IHAs) that are highlighted in Table 4-1. Two actions to mitigate large radiological releases are listed separately in Table 4-2.

Table 4-1 PRA credited operator actions to mitigate core damage

Name	Description	Context
CFDS-HFE-0001C-FOP-N	Unisolate and initiate containment flood and drain system (CFDS) injection (IHA)	Used for LOCA-OC, SGTF, and general transients
CVCS-HFE-0001C-FOP-N	Unisolate and initiate CVCS injection (IHA)	Used for LOCA-IC, LOCA-OC (letdown), general transients and secondary steam line break, upon failure of ECCS, and SGTF
CVCS-HFE-0002C-FOP-N	Locally unisolate and initiate CVCS injection	Local unisolation due to lack of control from a partial loss of DC power
ECCS-HFE-0001C-FTO-N	Actuate ECCS following failure of MPS to automatically isolate	Backup action to MPS auto function failure
EHVS-HFE-0001C-FTS-N	Operator starts and loads combustion turbine generator	Backup local action to control room initiation failure during loss of offsite power
ELVS-HFE-0001C-FTS-N	Operator starts and loads backup diesel generator	Backup local action to control room initiation failure during loss of offsite power

The important human actions of injecting water into the NuScale Power Module are performed to mitigate core damage when taken as assumed by supporting PRA analysis. The actions identified for prevention of core damage are also taken to arrest the progression of core damage once it has begun and to retain the core within the RPV. (Reference 7.2.1, Section 19.2.5).

Table 4-2 PRA credited operator actions to mitigate large radiological release

Name	Description	Context
CFDS-HFE-0001C-FOP-N	Unisolate and initiate CFDS injection (IHA)	Used for LOCA-OC, SGTF, and general transients
CNTS-HFE-0001C-FTP-N	Actuate CIS following failure of MPS to automatically isolate	Backup action to MPS auto function failure

Additional evaluations for beyond-design-basis events, performed and documented in FSAR Chapter 20, identified the following two actions not covered in other programs that are meant to prevent core damage during an extended loss of AC power (ELAP), or to mitigate core damage and minimize the spread of radioactive release during a loss of large area (LOLA) event.

Table 4-3 Operator actions described in FSAR for Chapter 20 beyond-design-basis events

Name	Description	Context
LOLA phase 2/ELAP	Add inventory to the UHS through the spent fuel pool assured makeup line	Long-term extended loss of AC power action (> 30 days)
LOLA phase 3	Mitigate damage to fuel in the reactor vessel and radiological release – evaluate safety functions, provide a means for water spray scrubbing using fog nozzles and available water sources. Address runoff water containment issues (sandbags, dikes, etc.)	Long-term action loss of large area to support a reduced emergency planning zone

Firefighting activities are not included within the generic technical guidelines and are addressed through other procedures. This is consistent with current industry practice where separate fire response plans and procedures are maintained specific to each site.

In addition to the actions specifically required by analysis, manual actions have been included when deemed appropriate by subject matter expert experience or to maintain consistency among the mitigating strategies. For example, the PRA only assumes manual actuation of one of seven ESFAS functions. For consistent performance and standardization for operators, when ANY ESFAS function does not actuate as designed, the manual action is taken to actuate that function from the control room. This provides the operator with a consistent response to any ESFAS failure and had demonstrated relatively quick response times during validation.

Manual actions are prioritized based on both the time allowed to perform the action and the location where the action is taken. The allowed times are based on PRA analysis of operator actions to mitigate core damage or large radiological releases. The following examples demonstrate how this prioritization is taken into account:

- Actions in the main control room are prioritized over actions taken locally in the plant.
- Actions with the shortest analyzed time for success will be prioritized first.

Table 4-4 provides a comprehensive list of all actions determined to be included in the generic technical guidance from sources described in Tables 4-1, 4-2, 4-3, and from subject matter expert task analysis. These are grouped by critical safety function.

Table 4-4 Operator actions to support critical safety functions

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4.5 Entry Conditions

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Figure 4-2 Example unit overview display

4.6 Exit Conditions

The generic technical guidance is intended to inform the writing of the emergency operating procedures to enable operators to mitigate accident conditions. The GTGs are exited and normal operating procedures commence when the event has been mitigated to the extent that parameters allow return to normal operation.

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4.7 Implementation Strategy

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4.8 Setpoint Selection

The listed setpoints have been derived from safety analysis (Table 7.1-4 of Reference 7.2.1), calculations, or best estimate. These setpoints may be adjusted to reflect design changes due to final selection of instrumentation, accuracy, and allowing appropriate time for the operator to respond. The values have been included within these guidelines to provide a reference and it is anticipated that the basis for the setpoints will remain constant.

Instrumentation requirements have been provided based on NuScale requirements, regulatory requirements, or vendor recommendations, but must be refined once the actual instrumentation is selected. Emergency procedures developed from these GTGs will need to reference the plant specific equipment values, ranges, and accuracies.

5.0 Generic Technical Guidelines and Basis

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5.1 Containment Integrity Safety Function

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5.1.1 Containment Atmosphere

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5.1.2 Containment Bypass

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Figure 5-1 Comparison of riser level vs. average RCS temperature for bounding cases

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5.1.3 Containment Isolation Signal Actuation

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5.1.4 Chemical and Volume Control System Isolation Actuation

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5.1.5 Primary-to-Secondary Leakage

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5.2 Reactivity Safety Function

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5.2.1 Reactor Trip System Actuation

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5.2.2 Dilution Isolation

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5.2.3 Reactivity Verification

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Figure 5-2 ATWS response under different conditions.

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5.3 Core Heat Removal Safety Function

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5.3.1 Ultimate Heat Sink Inventory

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5.3.2 Fuel Clad Protection

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5.3.3 Emergency Core Cooling System (ECCS) & Low Temperature Overpressure (LTOP) Actuation

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5.3.4 Decay Heat Removal Actuation

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5.3.5 Pressurizer Heater Trip Actuation

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5.4 Defense-in-Depth Actions

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5.4.1 Electrical

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5.4.2 Other Defense-in-Depth Actions

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6.0 Validation

This revision of the GTGs is issued with the understanding that the selected setpoints are based on analysis and engineering judgement. Instrument selection is not complete and final instrument selection may impact the listed setpoints as tolerances and accuracy are considered.

Validation is performed to determine that the actions specified within Section 5.0 of this report can be performed by trained operators to manage emergency conditions described within the FSAR.

Human Factors Engineering (HFE) verification and validation activities are included as part of Chapter 18 of the design certification submittal to the NRC. Through this HFE program, an Integrated System Validation (ISV) was conducted from 7/23/2018 to 9/6/2018. ISV is the process by which an integrated system design (i.e., hardware, software, and personnel elements) is evaluated using performance-based tests to determine whether it acceptably supports safe operation of the plant. During ISV testing, much of the GTG methodology was tested.

Individuals that participated in ISV were selected to approximate the expected crew characteristics that would operate a NuScale facility. The individuals selected had varied backgrounds, from recent college graduates to long term licensed plant operators. The individuals selected had not participated in the development of the HSI design in order to provide an unbiased assessment of the generic technical guidance in a wide range of normal, abnormal, and emergency conditions. These individuals were trained to have sufficient knowledge of the NuScale plant design, plant controls, conduct of operations and the emergency procedures, which had been written based off of the GTG guidance.

Specific GTG validation activities were performed for those sections of the GTG that were not covered through ISV testing. Each logic flow path and procedure terminus was tested using the ISV trained individuals. A combination of simulated exercises and table-top walkthroughs were performed for these items. Table-top validation was performed when validating actions taken either outside of the control room or for very long term actions that could not reasonably be checked by simulation.

Comments and corrective items associated with ISV/GTG validation have been incorporated into Revision 1 of this document.

Information on the simulator scenarios used and feedback received during ISV is available in the HFE Verification and Validation Results Summary (Reference 7.2.11). Validation check sheets for table-top sessions are maintained as saved files in the NuScale corporate server(s).

7.0 References

7.1 Source Documents

7.1.1 U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, May 1980.

7.2 Referenced Documents

7.2.1 NuScale Power Standard Plant Design Certification Application, Rev. 0.

7.2.2 U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

7.2.3 U.S. Nuclear Regulatory Commission, "Requirements for Emergency Response Capability," NUREG-0737, Supplement No. 1, December 1982.

7.2.4 U.S. Nuclear Regulatory Commission, "Standard Review Plan, Operating and Emergency Operating Procedures," NUREG-0800, Rev. 2, Chapter 13, Section 13.5.2.1.

7.2.5 U.S. Nuclear Regulatory Commission, "Guidelines for the Preparation of Emergency Operating Procedures," NUREG-0899, August 1982.

7.2.6 "Combustible Gas Control," TR-0716-50424, Rev. 0.

7.2.7 Electric Power Research Institute, "Severe Accident Management Guidance Technical Basis Report Volume 1: Candidate High-Level Actions and Their Effects," EPRI #1025295, EPRI, Palo Alto, CA, 2012.

7.2.8 Electric Power Research Institute, "Severe Accident Management Guidance Technical Basis Report Volume 2: The Physics of Accident Progression," EPRI #1025295, EPRI, Palo Alto, CA, 2012.

7.2.9 Electric Power Research Institute, "Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines—Revision 4," EPRI #1022832, EPRI, Palo Alto, CA, 2011.

7.2.10 Electric Power Research Institute, "Hydrogen Detection in Nuclear Power Plants: Comparison of Potential, Existing, and Innovative Technologies," EPRI #3002002107, EPRI, Palo Alto, CA, 2013.

7.2.11 "Human Factors Engineering Verification and Validation Results Summary Report," RP-1018-61289, Rev. 0.

7.2.12 U.S. Nuclear Regulatory Commission, "Methods for Review and Evaluation of Emergency Procedure Guidelines Volume I: Methodologies", NUREG/CR-3177, March 1983.

8.0 Appendix A – Event Progression Comparison

This appendix is intended to illustrate the differences in how a loss of coolant accident, steam generator tube rupture, loss of feedwater, and inadequate core cooling are mitigated between existing PWR technology and the NuScale design. The list of actions is provided at a high level for illustrative purposes only. These tables compare events between the two designs for those events within the scope of design-basis events.

Table 8-1 Loss of coolant accident

Typical PWR Mitigating Strategy/Actions	NuScale Mitigating Strategy/Actions
Maximize safety injection flow into the RCS	There is no injection phase
Manually isolate the source of RCS leakage	Containment Isolation occurs automatically
Manually control RCS pressure to maintain subcooling	RCS pressure control is not required. Passive cooling systems are designed so that subcooling is not necessary
Manually control RCS inventory to maintain subcooling	No additional makeup is necessary as long as the RCS leak is being captured in containment
Manually perform a controlled cooldown using the steam generators	DHRS passively performs cooldown within design cooldown rates
Manually secure RCS recirculation pumps	No recirculation pumps exist
Manually align shutdown cooling system if RCS leakage is small or has been isolated	ECCS occurs automatically to provide additional passive cooling
Transition from injection to containment sump recirculation	There is no injection phase and recirculation occurs naturally using the ECCS
Manually alternate hot and cold leg recirculation	There is no injection phase
Manual actions required	No manual actions required

Table 8-2 Steam generator tube failure

Typical PWR Mitigating Strategy/Actions	NuScale Mitigating Strategy/Actions
Manually confirm the affected steam generator	Not required as both are isolated
Manually perform a controlled cooldown using the steam generators	DHRS passively performs cooldown within design cooldown rates
Manually isolate the most affected steam generator	DHRS occurs automatically and isolates both steam generators
Manually control RCS pressure to maintain subcooling and match steam generator pressure	RCS pressure quickly matches DHRS pressure and effectively stops the RCS leak in the affected steam generator
Manually control RCS inventory to maintain subcooling	DHRS passively performs cooldown within design cooldown rates
Manually secure RCS recirculation pumps as needed	No recirculation pumps exist
Manually align shutdown cooling system if RCS leakage is small or has been isolated	ECCS occurs automatically to provide additional passive cooling if necessary
Manual actions required	No manual actions required

Table 8-3 Loss of feedwater

Typical PWR Mitigating Strategy/Actions	NuScale Mitigating Strategy/Actions
Manually minimize heat input to the RCS by securing RCS recirculation pumps	No recirculation pumps exist
Manually conserve steam generator inventory by isolating steam generator blowdowns	Decay heat removal occurs automatically that isolates the inventory in both steam generators
Manually regain a source of feedwater to at least one steam generator	Decay heat removal occurs automatically and isolates both steam generators and retains it in the DHRS
Manually control RCS pressure to maintain subcooling	RCS pressure control is not required. Passive cooling systems are designed so that subcooling is not jeopardized
Manually control RCS inventory to maintain subcooling	No additional RCS inventory is required
If feed cannot be restored, manually align injection to the RCS and openpower operated relief valves for once through core cooling	ECCS occurs automatically to provide additional passive cooling if necessary
Manual actions required	No manual actions required

Table 8-4 Inadequate core cooling/loss of forced circulation

Typical PWR Mitigating Strategy/Actions	NuScale Mitigating Strategy/Actions
Verify natural circulation established	No action, natural circulation is the normal cooling mechanism
Manually perform a controlled cooldown using the steam generators	DHRS passively performs cooldown within design cooldown rates
Manually control RCS pressure to maintain subcooling	RCS pressure control is not required. Passive cooling systems are designed so that subcooling is not jeopardized
Manually control RCS inventory to maintain subcooling	No additional RCS inventory is required
Manually restore electrical power to de-energized AC buses	No AC power is required to obtain safe shutdown conditions
Manually start RCS recirculation pumps as needed	No recirculation pumps exist
Manually align shutdown cooling system	ECCS occurs automatically to provide additional passive cooling if necessary
Manual actions required	No manual actions required

9.0 Appendix B – NuScale Design Severe Accident Management Guideline Development

As part of the development of the severe accident management guidelines, EPRI “Severe Accident Management Guidance Technical Basis Report, Volume 1: Candidate High-Level Actions and Their Effects” (Reference 7.2.7) was reviewed and the 20 candidate high-level actions (CHLAs) identified in the report were categorized by their applicability to the NuScale design. Given the unique design features of the NuScale Power Module, the applicability review was not limited to PWR actions only.

Table 9-1 EPRI Candidate high-level action comparison to NuScale design

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}}2(a),(c)

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10.0 Appendix C – Instrumentation Utilized in the General Technical Guidelines

The following instrumentation is used as inputs to the RTS. A valid reactor trip signal is processed when two of four of the input channel parameters exceed the analytical limit setpoint. A reactor trip is the initiator for entry into the generic technical guidance. The table lists the parameter, limit, number of channels, and the actuation logic with an arrow indicating if the parameter is rising or lowering to actuate. Each of the reactor trip parameters are described in the Safety Analysis Analytical Limits Report (Reference 7.2.1, Table 7.1-4).

Table 10-1 List of reactor trip parameters

Parameter	Analytical Limit	Channels	Logic
High Power Range Linear Power	High-1 = 25% RTP High-2 = 120% RTP	4	2/4↑
High Intermediate Range Log Power Rate	3 dpm	4	2/4↑
High Power Range Positive and Negative Rate	+/- 15% RTP/minute	4	2/4↕
High Source Range Count Rate	5.0×10^5 cps	4	2/4↑
High Source Range Log Power Rate	3 dpm	4	2/4↑
High Range RCS Thot (NR RCS T_{hot})	610°F	4	2/4↑
High Range Containment Pressure	9.5 psia	4	2/4↑
High Pressurizer Pressure	2000 psia	4	2/4↑
Low Pressurizer Pressure	1720 psia (when T_{hot} > 600°F)	4	2/4↓
Low Low Pressurizer Pressure	1600 psia	4	2/4↓
High Pressurizer Level	80%	4	2/4↑
Low Pressurizer Level	35%	4	2/4↓
High Main Steam Pressure	800 psia	4	2/4↑
Low Main Steam Pressure	300 psia (when NI power > 15%)	4	2/4↓
Low Low Main Steam Pressure	20 psia	4	2/4↓
High Steam Superheat (MS Temperature and Pressure)	150°F	4	2/4↑
Low Steam Superheat (MS Temperature and Pressure)	0.0°F	4	2/4↓
Low Low RCS Flow	0.0 ft^3/sec	4	2/4↓
Low ELVS 480 VAC to EDSS Battery Chargers	80% normal ELVS voltage with actuation delay of 60 seconds	4	2/4↓
High Under-the-Bioshield Temperature	250°F	4	2/4↑

The following parameters are used within the generic technical guidance as decision points for the operators. The parameters are arranged by safety function and, in most cases, are the same as the actuation setpoints for the associated ESFAS functions.

Table 10-2 List of general technical guidelines decision variables

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Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0519-65790

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

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

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

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

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

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

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

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

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the process by which NuScale develops its generic technical guidelines.

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

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

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

(4) The information sought to be withheld is in the enclosed report entitled "NuScale Generic Technical Guidelines," TR-1117-57216-P, Revision 1.The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon

the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR § 2.390(a)(4) and 9.17(a)(4).

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

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

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

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

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

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 31, 2019.

Thomas A. Bergman