# 13 CONDUCT OF OPERATIONS

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 13, "Conduct of Operations," of the NuScale Power, LLC (hereinafter referred to as the applicant) Design Certification Application (DCA), Part 2, "Final Safety Analysis Report [FSAR]," (DCA Part 2).

## 13.1 Organizational Structure

### 13.1.1 Introduction

A COL applicant's organizational structure includes the corporate-level management and technical support organization and the onsite operating organization. The management and technical support organization includes the corporate or home office offsite organization; associated functions, activities, and responsibilities; and the approximate number and qualifications of offsite personnel necessary to ensure that sufficient technical resources have been, are being, and will continue to be provided to accomplish the safe design, construction, testing, and operation of the nuclear plant. The onsite operating organization includes the structure, functions, activities, responsibilities, and the approximate number and qualifications of onsite personnel necessary to safely operate and maintain the facility.

The staff reviewed the DCA to evaluate the acceptability of combined license (COL) information items that pertain to (1) COL applicant descriptions of the corporate-level management and technical support organization and (2) COL applicant descriptions of the onsite operating organization.

### 13.1.2 Summary of Application

**DCA Part 2, Tier 1:** There is no Tier 1 information associated with this section.

**DCA Part 2, Tier 2:** The plans for a corporate-level, technical, and onsite organizational structure to support, design, construct, test, operate, and maintain the nuclear plant are not within the scope of the NuScale DCA. This responsibility resides with the COL applicant. In DCA Part 2, Tier 2, Section 13.1, "Organizational Structure," the applicant specified COL information items for the COL applicant to describe the corporate-level management and technical support organization and the onsite operating organization.

**ITAAC:** There are no inspections, tests, analyses, and acceptance criteria (ITAAC) for this area of review.

**Technical Specifications:** There are no technical specifications (TS) for this area of review.

**Technical Reports:** There are no technical reports (TRs) associated with this area of review.

### 13.1.3 Regulatory Basis

Section 13.1.1, "Management and Technical Support Organization," and Section 13.1.2 – 13.1.3, "Operating Organization," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), identifies, in part, the relevant NRC regulatory requirements for organizational structure and the associated acceptance criteria.

The applicable regulatory requirements for the organizational structure are as follows:

- 10 CFR 50.34(f)(3)(vii), as it pertains to requirements related to lessons learned from the accident at Three Mile Island (TMI) for the applicant to describe the management plan for design and construction activities of the proposed plant.

- 10 CFR 50.40(b), which requires the COL applicant to be technically qualified to engage in activities associated with the design, construction, and operation of a nuclear power plant.

- 10 CFR 50.48(a)(1)(ii), as it pertains to information that must be included in the fire protection plan of the holder of a COL under 10 CFR Part 52, specifically, the identification of the various positions within the licensee's organization that are responsible for the program.

- 10 CFR 50.54(i), (j), (k), (l), and (m), as they pertain to the organizational staffing requirements for, and responsibilities of, operators and senior operators licensed under 10 CFR Part 55, "Operators' Licenses."

- 10 CFR Part 50, Appendix B, as it pertains to organizational responsibilities for the establishment and execution of the quality assurance program.

- Sections 52.79(a)(26)–(28) and (29)(i) of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants", as they pertain to information that must be included in the FSAR that is submitted as part of the application for a COL, specifically:

  (1) the applicant's organizational structure, allocations or responsibilities and authorities, and personnel qualifications requirements for operation;
  (2) managerial and administrative controls to be used to assure safe operation as established in 10 CFR Part 50, Appendix B;
  (3) plans for preoperational testing and initial operations; and
  (4) plans for the conduct of normal operations.

The related acceptance criteria are as follows:

- Section III, "Acceptance Criteria," of SRP Section 13.1.1, Revision 6, issued August 2016.

- Section III of SRP Section 13.1.2 –13.1.3, Revision 7, issued August 2016.

- Review Criterion 6.4(2) in Section 6, "Staffing and Qualifications," of NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, issued November 2012.

- NUREG-1791, "Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m)," issued July 2005.

- NUREG/CR-6838, “Technical Basis for Regulatory Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m),” issued February 2004.

## 13.1.4 Technical Evaluation

The COL applicant is responsible for describing the corporate-level management and technical support organization and the onsite operating organization. This section presents the staff’s evaluation of the acceptability of COL information items that pertain to the COL applicant’s organizational structures.

### *13.1.4.1 Combined License Information Items*

NRC regulations require a COL applicant that references the NuScale certified design to submit the site-specific information described in the COL information items at the COL stage.

#### *13.1.4.1.1 Management and Technical Support Organization*

SRP Section 13.1.1 states that for the management and technical support organization, the COL applicant’s SAR should

(1) describe the qualification requirements for each identified position or class of positions that provide technical support to the onsite operating organization, and

(2) specify the qualification requirements for individuals holding management and supervisory positions in organizational units that provide support to the onsite operating organization.

In DCA Part Tier 2, Revision 0, Section 13.1.1, “Management and Technical Support Organization,” the applicant specified COL Item 13.1-1, which requires the COL applicant to describe the corporate-level or home office management and technical support organization and specify the necessary qualification requirements for positions within the management and technical support organization that provide technical support to the onsite operating organization. The staff finds that COL Item 13.1-1 is acceptable because it addresses the information that the COL applicant must provide for corporate-level management and technical-support organizations.

#### *13.1.4.1.2 Operating Organization*

SRP Section 13.1.2 – 13.1.3 states that the COL applicant’s SAR should describe (1) the structure, functions, and responsibilities of the onsite operating organization established to operate and maintain the plant and (2) any alternatives to the requirements involving the number of licensed personnel, as specified in 10 CFR 50.54(m). Consistent with the SRP, in DCA Part 2, Tier 2, Section 13.1.2, “Operating Organization,” COL Information Item 13.1-2 requires the COL applicant to describe the onsite operating organization, including the structure, functions, and responsibilities. In addition, COL Item 13.1-2 specifies that the proposed operating staff shall be consistent with the minimum licensed operator staffing requirements in DCA Part 2, Tier 2, Section 18.5, “Staffing and Qualification.” In DCA Part 2, Tier 2, Section 18.5, the applicant describes a staffing level and qualifications analysis that is an alternative to the requirements of 10 CFR 50.54(m). Within the context of the Chapter 13 review, the staff concludes that it is acceptable for the COL Item to reference the discussion in DCA Part 2, Tier 2, Section 18.5. Accordingly, the staff determined that COL Item 13.1-2 is acceptable. SER Chapter 18 describes the staff's evaluation of the staffing and qualification element of the NuScale human factors engineering program.

SRP Section 13.1.2 –13.1.3 states that the COL applicant's SAR should describe the education, training, and experience requirements (qualification requirements) that the applicant established to fill each management, operating, technical, and maintenance position category in the operating organization. In DCA Part 2, Tier 2, Revision 0, Section 13.1.3, COL Item 13.1-3 requires the COL applicant to describe the qualification requirements for each of the identified position categories for the operating organization. Accordingly, the staff determined that COL Item 13.1-3 is acceptable.

### 13.1.5 Combined License Information Items

Table 13.1-5 lists COL information items related to the organizational structure from DCA Part 2, Tier 2, Table 1.8-2, "Combined License Information Items." The staff verified that DCA Part 2, Tier 2, Revision 1, adequately reflects the COL information items presented in Table 13.1-5.

**Table 13.1-5 NuScale COL Information Items for Section 13.1**

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| COL Item 13.1-1 | A COL applicant that references the NuScale Power Plant design certification will provide a description of the corporate or home office management and technical support organization, including a description of the qualification requirements for (1) each identified position or class of positions that provide technical support to the onsite operating organization, and (2) individuals holding management and supervisory positions in organizational units providing technical support to the onsite operating organization. | 13.1.1 |
| COL Item 13.1-2 | A COL applicant that references the NuScale Power Plant design certification will provide a description of the proposed structure, functions, and responsibilities of the onsite organization necessary to operate and maintain the plant. The proposed operating staff shall be consistent with the minimum licensed operator staffing requirements in Section 18.5. | 13.1.2 |
| COL Item 13.1-3 | A COL applicant that references the NuScale Power Plant design certification will provide a description of the qualification requirements for each management, operating, technical, and maintenance position described in the operating organization. | 13.1.3 |

### 13.1.6 Conclusion

For the reasons given above, the staff concludes that the COL information items specified in Table 13.1-5 and included in DCA Part 2, Tier 2, Revision 1, are sufficient to identify information that the COL applicant needs to provide to meet the applicable requirements of 10 CFR 50.34(f)(3)(vii); 10 CFR 50.40(b); 10 CFR 50.48(a)(1)(ii); 10 CFR 50.54(i), (j), (k), (l), and (m); 10 CFR Part 50, Appendix B; and 10 CFR 52.79 (a)(26)-(28), (29)(i), and (40).

## 13.2 Training

### 13.2.1 Introduction

A COL applicant's training program should include (1) the initial license training program for reactor operators and senior reactor operators, (2) the licensed operator requalification program, and (3) the non-licensed plant staff training program, which consists of initial training, periodic retraining, and qualification(s) for non-licensed operators, shift supervisors, shift technical advisors, instrumentation and control (I&C) technicians, electrical maintenance personnel, mechanical maintenance personnel, radiological protection technicians, chemistry technicians, and engineering support personnel.

The staff reviewed the DCA to evaluate the acceptability of COL information items that pertain to the COL applicant's description of, and schedule for, (1) the licensed operator training program for reactor operators and senior reactor operators, including the licensed operator requalification program, and (2) the training program for the non-licensed plant staff.

### 13.2.2 Summary of Application

**DCA Part 2, Tier 1:** There is no Tier 1 information associated with this section.

**DCA Part 2, Tier 2:** The development of site-specific training programs is not within the scope of the NuScale DCA. This responsibility resides with the COL applicant. In DCA Part 2, Tier 2, Section 13.2, "Training," the applicant specified a single COL information item that requires the COL applicant to describe the initial license training program, the licensed operator requalification program, and the nonlicensed plant staff training program and to provide schedules for these programs.

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

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

**Technical Reports:** There are no TRs associated with this area of review.

### 13.2.3 Regulatory Basis

SRP Section 13.2.1, "Reactor Operator Requalification Program; Reactor Operator Training," and SRP Section 13.2.2, "Non-Licensed Plant Staff Training," identify, in part, the relevant NRC regulatory requirements for training and the associated acceptance criteria.

The applicable regulatory requirements for training are as follows:

- 10 CFR 19.12, "Instruction to workers," as it pertains to instructions provided to workers regarding protection of personnel from exposure to radiation or radioactive material.

- 10 CFR 26.29, "Training," as it pertains to employee training associated with the fitness-for-duty program.

- 10 CFR 50.34(f)(2)(ii), as it pertains to the TMI-related requirement for applicants to establish a program to begin during construction and to follow into operation for assessing and improving plant procedures applicable to operator training.

- 10 CFR 50.40(a) and (b), as they pertain to the issuance of a COL under 10 CFR Part 52 based on considerations of whether the applicant (1) is technically qualified to engage in activities associated with the design, construction, and operation of a nuclear power plant and (2) has established the licensed and non-licensed plant staff training programs necessary to provide reasonable assurance that the nuclear power plant can be safely operated.

- 10 CFR 50.54(a)(i-1), as it pertains to requirements for the establishment of a licensed operator requalification training program within 3 months after the date that the Commission makes the finding under 10 CFR Part 52.103(g) that the acceptance criteria in the COL are met.

- 10 CFR 50.120(b)(1)–(3), as they pertain to requirements for the establishment, implementation, and maintenance of training programs derived from a systems approach to training as defined in 10 CFR 55.4, "Definitions," for specific categories of nuclear power plant personnel.

- 10 CFR Part 50, Appendix B, as it pertains to the training and technical qualifications of personnel who perform activities that affect the quality of structures, systems, and components (SSCs) that are covered by the quality assurance program.

- 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," as it pertains to the requirements for emergency preparedness training of employees and other persons whose assistance may be needed in the event of a radiological emergency (e.g., local emergency services and law enforcement personnel), including participation in drill and exercise scenarios to provide performance opportunities to develop, maintain, and demonstrate key skills.

- 10 CFR 52.79(a)(26)–(28) and (29)(i), as they pertain to information to be included the COL FSAR, specifically:

  (1) the qualification requirements of licensed and non-licensed plant personnel to engage in activities associated with operation of the nuclear power plant,

  (2) the controls associated with the training of personnel who perform activities that affect the quality of SSCs that are covered by the quality assurance program as established in 10 CFR Part 50, Appendix B,

  (3) plans for licensing personnel and training non-licensed plant staff before criticality to support preoperational testing activities and initial operations, and

  (4) plans for licensed and non-licensed plant staff to receive the technical and administrative training required to operate, test, and maintain the nuclear power plant during the conduct of normal operations.

- 10 CFR 52.79(a)(14), (21), (33), (34), (36), (39), (40) and (44), as they pertain to information that must be included in the FSAR that an applicant submits as part of the application for a COL, specifically, descriptions of (1) licensed operator training required by 10 CFR Part 55 and (2) training required by 10 CFR 50.120, “Training and qualification of nuclear power plant personnel,” for specific categories of nuclear power plant personnel, and (3) non-licensed plant staff training associated with security procedures, radiological emergency plans, radiation protection, fire protection, and fitness for duty.

- 10 CFR 55.4, as it pertains to Commission-approved training programs that are based on a “systems approach to training.”

- 10 CFR 55.31(a)(4)–(5), as they pertain to the documentation requirements associated with successful completion by an applicant for an operator license of a facility licensee’s training program, when the facility licensee requests administration of the licensing exam (i.e., written examination and operating test).

- 10 CFR 55.41, “Written examination:  Operators,” as it pertains to requirements associated with the content and makeup of the NRC’s written examination for operators.

- 10 CFR 55.43, “Written examination:  Senior operators,” as it pertains to requirements associated with the content and makeup of the NRC’s written examination for senior operators.

- 10 CFR 55.45, “Operating tests,” as it pertains to requirements associated with (1) the content and makeup of the NRC’s operating test for operators and senior operators and (2) the use of a Commission approved simulation facility, a plant-referenced simulator, or the physical plant for administration of the operating test.

- 10 CFR 55.46, “Simulation Facilities,” as it pertains to requirements for the use of simulation facilities in the administration of the NRC operating test.

- 10 CFR 55.59, “Requalification,” as it pertains to requirements associated with licensed operator requalification training programs.

The related acceptance criteria are as follows:

- Section III of SRP Section 13.2.1, Revision 4, issued August 2016.

- Section III of SRP Section 13.2.2, Revision 4, issued August 2016.

- Regulatory Guide (RG) 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 3, issued May 2000.

- RG 1.149, “Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations,” Revision 4, issued April 2011.

- NUREG-0711, Revision 3.

- NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 11, issued February 2017.

- NUREG-1220, "Training Review Criteria and Procedures," Revision 1, issued January 1993.

## 13.2.4 Technical Evaluation

The COL applicant is responsible for the development of site-specific training programs. This section presents an evaluation of the acceptability of COL information items that pertain to training programs for licensed and non-licensed plant staff.

### *13.2.4.1 Combined License Information Items*

NRC regulations require the COL applicant that references the NuScale certified design to submit the site-specific information described in COL information items at the COL stage.

#### *13.2.4.1.1 Licensed and Non-licensed Plant Staff Training Programs*

SRP Section 13.2.1 states that the COL applicant should describe the description and scheduling of the licensed operator training program for reactor operators and senior reactor operates, including the licensed operator requalification program. SRP Section 13.2.2 states that the COL applicant's non-licensed plant staff training program should include the initial training, periodic retraining, and qualification that are required for non-licensed plant staff. The staff reviewed DCA Part 2, Section 13.2, COL Item 13.2-1 and found that it specifies the appropriate and necessary information for licensed plant staff. The staff reviewed DCA Part 2, Section 13.2, COL 13.2-2, and found that it specifies the appropriate and necessary information for non-licensed plant staff training programs. The staff also verified that Revision 1 of the DCA Part 2, Tier 2 adequately incorporates the COL information items presented in Table 13.2-5.

## 13.2.5 Combined License Information Items

Table 13.2-5 lists COL information item numbers and descriptions related to training from DCA Part 2, Tier 2, Table 1.8-2. Revision 1 of the DCA reflects the COL information items presented in Table 13.2-5.

**Table 13.2-5 NuScale COL Information Items for Section 13.2**

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| COL Item 13.2-1 | A COL applicant that references the NuScale Power Plant design certification will provide a description and schedule of the initial training and qualification as well as requalification programs for reactor operators and senior reactor operators. | 13.2 |
| COL Item 13.2-2 | A COL applicant that references the NuScale Power Plant design certification will provide a description and schedule of the non-licensed | 13.2 |

| | | |
|---|---|---|
| | plant staff training programs, including initial training, periodic retraining, and qualification requirements. | |

### 13.2.6 Conclusion

For the reasons given above, the staff concludes that the COL information items specified in Table 13.2-5 of this SER and included in Revision 1 of the DCA are sufficient to identify information the COL applicant needs to provide to meet the applicable requirements of 10 CFR 19.12; 10 CFR 26.29; 10 CFR 50.34; 10 CFR 50.40; 10 CFR 50.54; 10 CFR 50.120; 10 CFR Part 50, Appendix B; 10 CFR Part 50, Appendix E; 10 CFR 52.79; and 10 CFR Part 55.

## 13.3 Emergency Planning

### 13.3.1 Introduction

The NRC staff conducts its review of emergency planning (EP) in the DCA in accordance with the requirements in 10 CFR 52.47 and 10 CFR 52.48, "Standards for review of applications." The review addresses those design features, facilities, functions, and equipment that are technically relevant to the design, that are not site specific, and that affect some aspect of EP or the capability of a licensee to cope with plant emergencies. In addition, the review addresses design facilities such as a habitable technical support center (TSC) with adequate space, data retrieval capabilities, and dedicated communications equipment and an operational support center (OSC) with adequate communications. There is no minimum level of design-related EP that an application must address. The applicant may choose the extent to which the application includes EP features to be reviewed as part of the design certification.

The NRC conducted the review of design information and COL information items (designated as COL items) related to EP and documented the results in this section of Chapter 13. The COL items are listed in SER Section 13.3.5.

### 13.3.2 Summary of Application

The NuScale DCA contains DCA Part 2, which is divided into two tiers (Tier 1 and Tier 2). The sections below summarize the information submitted in Tier 1 and Tier 2.

**DCA Part 2, Tier 1:** DCA Part 2, Tier 1, Section 3.2, "Normal Control Room Heating Ventilation and Air Conditioning System," describes the normal control room heating, ventilation, and air conditioning (HVAC) system, which is also referred to as the control room ventilation system (CRVS). The CRVS serves the entire control building (CRB), which includes the TSC, and the access tunnel between the CRB and the reactor building (RXB). The CRVS is a nonsafety-related system. DCA Part 2, Tier 1, Table 3.2-2, "Normal Control Room Heating Ventilation and Air Conditioning Inspections, Tests, Analyses, and Acceptance Criteria," includes the ITAAC associated with the CRVS. SER Section 14.3 evaluates these ITAAC.

**DCA Part 2, Tier 2:** In DCA Part 2, Tier 2, Section 13.3, "Emergency Planning," the applicant stated that the NuScale design includes "design features, facilities, and equipment that are usable for up to 12 NuScale Power Modules to support emergency response functions." DCA Part 2, Tier 2, Section 13.3, describes that the TSC is located on the 30.48-meter (m) (100-foot (ft)) elevation of the CRB. Additionally, the TSC design ensures that TSC personnel are protected from radiological hazards during accident conditions (i.e., radiation dose is limited to

5 roentgen equivalent man (rem) total effective dose equivalent (TEDE) for the duration of the accident).

In the event of a loss of CRVS or if the TSC becomes otherwise uninhabitable, personnel are evacuated and the TSC functions are transferred to an alternate location. Although EP is, for the most part, the responsibility of the COL applicant, the design bases for the standard plant include design features, facilities, functions, and equipment necessary for EP. The COL applicant is responsible for the interfaces of these features with site-specific parameters. In DCA Part 2, Tier 2, Section 13.3, the applicant stated that “[i]n the event of a loss of ventilation, or if the TSC becomes otherwise uninhabitable, personnel are evacuated and the TSC functions are transferred to a location designated by the emergency plan (COL Item 13.3-3).”

The design bases for the standard plant include the following EP features:

- TSC

  A Technical Support Center (TSC) is an onsite facility that provides plant management and technical support to the plant operations personnel during emergency conditions.

- emergency response data system

  An emergency response data system (ERDS) is a direct near-real-time electronic data transmission system linked to the NRC Headquarters Operation Center that provides plant parameters from the onsite computer system. It allows the NRC to assess plant conditions and provide advice and support to the licensee and to Federal, State, and local authorities.

- OSC

  An Operations Support Center (OSC) is a facility for emergency maintenance and other support personnel to gather as a ready resource to support actions initiated by the control room during an emergency. The applicant has identified COL Item 13.3-1, which requires the COL applicant to describe the onsite OSC.

- emergency operations facility

  An emergency operations facility (EOF) is a support facility for the management of overall licensee emergency response (including coordination with Federal, State, and local officials), coordination of radiological and environmental assessments, and determination of recommended public protective actions. The applicant has identified COL Item 13.3-2, which requires the COL applicant to describe the site-specific EOF.

- TSC engineering workstations

  The TSC engineering workstations are part of the module control system (MCS) and plant control system (PCS), which provide monitoring functionality to plant processes and equipment. DCA Part 2, Tier 2, Section 7.2.13.7, “Other Information Systems,” further describes the TSC engineering workstations, and the corresponding section of this SER evaluates these workstations.

- decontamination facilities

  Decontamination facilities, located in the annex building, are provided to remove or reduce radioactive contaminants from plant equipment, protective clothing, and personnel.  DCA Part 2, Tier 2, Section 12.1.2.3, "Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA," includes more information on the decontamination facilities, and the corresponding section of this SER evaluates these facilities.

- process sampling system

  The process sampling system provides the capability to sample and analyze liquid and gaseous samples following an accident without the need for a dedicated post-accident sampling system.  DCA Part 2, Tier 2, Section 9.3.2, "Process Sampling System," and Section 12.3, "Radiation Protection Design Features," fully describe the process sampling system, and the corresponding sections of this SER evaluate it.

  The NRC considers the review of the process sampling system in Section 13.3.4.8 of the SER an open item until the staff's evaluation of the capabilities of the process sampling system to be used for obtaining samples post-accident is completed, and identifies it as **Open Item 13.3-1.**  The capability to obtain a post-accident sample is an interface item between SRP Section 9.3.2, "Process and Post-Accident Sampling Systems," and SRP Section 13.3.

The following DCA Part 2, Tier 2 sections describe the design features with an interface to EP:

- Section 2.3, "Meteorology"
- Section 6.4, "Control Room Habitability"
- Section 7.2.13.7, "Other Information Systems"
- Section 9.3.2,  "Process Sampling System"
- Section 9.4.1, "Control Room Area Ventilation System"
- Section 9.5.2, "Communication System"
- Section 12.1.2.3, "Facility Layout General Design Considerations for Maintaining Radiation Exposures ALARA"
- Section 12.3, "Radiation Protection Design Features"
- Section 12.4, "Dose Assessment"
- Section 12.5, "Operational Radiation Protection Program"

- Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors”

- Section 18.0, “Human Factors Engineering”

- Section 20.0, “Mitigation of Beyond Design Basis Events”

The respective sections of this SER address the staff’s evaluation of these additional DCA Part 2, sections.

**ITAAC:** The COL applicant will provide proposed ITAAC to support the facility’s EP, as appropriate considering site-specific information (see COL Item 14.3-1).

### 13.3.3 Regulatory Basis

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

- 10 CFR 52.79(a)(21) requires a COL application to include emergency plans complying with 10 CFR 50.47 “Emergency plans” and 10 CFR Part 50, Appendix E. Although a DCA is not required to provide this information, any EP-related information a design certification applicant requests be certified to support a future COL application must comply with these COL regulations.

- 10 CFR 50.47;

- 10 CFR 100.1, “Purpose”; 10 CFR 100.3, “Definitions”; 10 CFR 100.20, “Factors to be considered when evaluating sites”; and 10 CFR 100.21(g), as they relate to EP and emergency preparedness.

- 10 CFR Part 50, Appendix E, as it relates to EP and the ERDS.

- 10 CFR 52.48, as it relates to EP information in 10 CFR 50.47 submitted in a standard DCA.

- 10 CFR 52.47(b)(1), which requires that a DCA include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in accordance with the design certification; the provisions of the Atomic Energy Act of 1954, as amended; and the NRC’s rules and regulations.

The following guidance documents provide criteria relevant to this review and are used to confirm that the above requirements have been adequately addressed:

- SRP Section 13.3 lists the acceptance criteria that are adequate to meet the above requirements and the review interfaces with other SRP sections.

- NUREG-0654 contains specific acceptance criteria that the NRC has determined provide an acceptable means of complying with the standards in 10 CFR 50.47.

- NUREG-0696, “Functional Criteria for Emergency Response Facilities,” issued February 1981, describes the facilities and systems that nuclear power plant licensees will use to improve responses to emergencies.

- NUREG-0737, Supplement 1, describes post-TMI requirements for emergency response capabilities that have been approved for implementation.

- The NRC Office of Nuclear Security and Incident Response (NSIR)/Division of Preparedness and Response (DPR) Interim Staff Guidance (ISG) document, NSIR/DPR-ISG-01, “Interim Staff Guidance—Emergency Planning for Nuclear Power Plants,” issued November 2011, provides updated guidance for addressing EP requirements for nuclear power plants based on changes to emergency preparedness regulations in 10 CFR 50.47 and 10 CFR Part 50, Appendix E, that the NRC published in the *Federal Register* on November 23, 2011 (Volume 76 of the *Federal Register* (FR), page 72560 (76 FR 72560)).

### 13.3.4 Technical Evaluation

#### *13.3.4.1 Technical Support Center*

The staff reviewed the information in the DCA Part 2 for conformance with the applicable standards and requirements identified in SRP Section 13.3. DCD Tier 2, Section 13.3, “Emergency Planning,” and other DCA Part 2 chapters listed in SER Section 13.3.2 describe the design features of the TSC for the NuScale standard design.

The TSC is an onsite facility that provides plant management and technical support to the plant operations personnel during emergency conditions. DCA Part 2, Tier 2, Section 13.3, describes the physical location and size of the TSC. The staff issued RAI 8925, Question 13.03-2, Part iv (ADAMS Accession No. ML17206A098), asking the applicant to indicate the location in the DCA Part 2, of a diagram of the TSC in relation to the main control room (MCR). In its response to RAI 8925, Question 13.03-2 (ADAMS Accession No. ML17264B172), NuScale stated that “[t]he TSC is located directly above the MCR in the CRB, as shown in DCA Part 2, Tier 2 Figure 1.2-24 and Figure 1.2-27. Using either stairwell in the CRB, the transit time between the TSC and the MCR is less than two minutes, meeting the guidance in NUREG-0696 Section 2.2.” DCA Part 2, Tier 2, Section 13.3, notes that the TSC is located on the 100-foot elevation of the control building, which is a seismic Category I structure below the 120-foot elevation, as discussed in DCA Part 2, Tier 2, Section 3.8.4.1.2, “Control Building.”

The TSC is sized to provide working space of 75 square feet per person to avoid crowding and is designed to accommodate at least 25 people, including 5 NRC staff members and 20 utility staff members. This is consistent with the specific space and personnel accommodation criteria in NUREG-0696, Section 2.4, and is acceptable. The TSC includes 2,500 square feet for a technical evaluation room and an additional 1,000 square feet for storage, three offices, and two conference rooms.

The TSC is equipped with voice communications systems, which provide communications between the TSC and the following locations: the plant, local, and offsite emergency response facilities; the NRC; and local and State operations centers. DCA Part 2, Tier 2, Section 9.5.2.2, “System Description,” provides additional information on the TSC’s voice communications systems. The associated SER Section 9.5.2, “Communication Systems,” documents the staff’s evaluation and finding that the communications systems are acceptable.

The staff issued RAI 8925, Question 13.03-3 (ADAMS Accession No. ML17206A098), Part ii, asking the applicant to provide additional information on the backup power capabilities of the NuScale design as it pertains to the instrumentation and data system equipment in the TSC. In its response to RAI 8925, Question 13.03-3, Part ii (ADAMS Accession No. ML17264B172), NuScale stated the following:

> *TSC equipment, such as engineering workstations and communications equipment, are components of the MCS, PCS, and communications system (COMS). Power is supplied to these systems from the normal DC [direct current] power system (EDNS).*
>
> *As described in FSAR Tier 2 Section 8.3.2.1.2, the EDNS consists of batteries, battery chargers, and inverters. The EDNS battery chargers are normally supplied from the low voltage AC electrical distribution system (ELVS). Following a loss of AC electrical power supply to the EDNS battery chargers, the batteries automatically assume the loads for a minimum duration of 40 minutes.*
>
> *Additionally, in the event of a loss of normal onsite AC power, backup power to the ELVS can be provided by the backup power supply system (BPSS). Spare battery and charger terminal connection points are also provided for connection to mobile battery and charging units if necessary. The BPSS design is described in FSAR Tier 2 Section 8.3.1.1.2.*

DCA Part 2, Tier 2, Section 8.3.1.1.2, states that the BPSS is designed to provide electrical power to the NuScale power plant when normal alternating current (ac) power is not available and that it includes two redundant backup diesel generators and an auxiliary ac power source. The associated SER Chapter 8 sections document the staff's evaluation of the capability of the EDNS, ELVS, and BPSS.

Section 2.8 of NUREG-0696 states that the TSC must contain primary and backup power in order to maintain continuity of TSC functions and to immediately resume data acquisition, storage, and display of TSC data in the event of a loss of power. The applicant's description of the normal and backup power sources to the TSC equipment through the EDNS, ELVS, and BPSS show that the TSC has an adequate source of reliable power from a normal and backup system to maintain continuity of TSC functions including data acquisition, storage, and display.

The TSC has the purpose of protecting personnel from direct, airborne, in-plant radiological hazards under accident conditions. DCA Part 2, Tier 2, Section 13.3, states, in part, "The design ensures that personnel are protected from radiological hazards, including direct radiation and airborne radioactivity from in-plant sources under accident conditions (i.e., maximum of 5 rem TEDE for the duration of the accident)."

Section 2.6, "Habitability," of NUREG-0696 states that the purpose of the TSC is to provide direct management and technical support to the control room during an accident. Section II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operations," of NUREG-0737 states that any area that will, or may, require occupancy to permit an operator to aid in the mitigation of, or recovery from, an accident is designated as a "vital area." The control room and TSC must be included among those areas to which access is considered vital after an accident. Further, the design dose rate for personnel in a vital area should be such that doses do not exceed the requirements of General Design Criterion (GDC) 19, “Control room,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, during an accident. GDC 19 requires that radiation protection be adequate to ensure that radiation exposure to personnel does not exceed 0.05-sievert (Sv) (5-rem) TEDE for the duration of the accident. In addition, NUREG-0737, Supplement 1, Section 8.2.1.f, states that the TSC will be provided with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 0.05-Sv (5-rem) whole body or its equivalent to any part of the body for the duration of the accident (after NUREG-0737, Supplement 1, was published, the NRC revised GDC 2 so that the applicable dose for the DCA is 5-rem TEDE). These requirements and associated guidelines form the basic radiological habitability criteria for the TSC.

The staff issued RAI 8925, Question 13.03-2, Part ii (ADAMS Accession No. ML17206A098), asking the applicant to clarify if the TSC would be able to protect personnel from direct, airborne, in-plant radiological hazards under accident conditions in a manner that is equivalent to the MCR, as specified in Section 2.6 of NUREG-0696. In its response to RAI 8925, Question 13.03-2, Part ii (ADAMS Accession No. ML17264B172), NuScale stated the following:

> *The TSC and the main control room (MCR) share the control room ventilation system (CRVS), as described in FSAR Tier 2 Section 9.4.1. Upon detection of a high radiation level in the outside air intake, the normal outside air flow path is isolated and 100 percent of the outside air is bypassed through the air filtration unit to filter outside air and minimize radiation exposure to personnel within the MCR and TSC. This mode of operation allows the CRVS to continue providing breathable air to the MCR and TSC personnel and maintain the MCR and TSC at positive pressure with respect to the outside environment. This design meets the requirements of NUREG-0696, Section 2.6.*

SER Section 9.4.1 provides the staff’s evaluation of the CRVS. The staff’s evaluation of the radiological habitability of the TSC will be addressed in SER Section 15.0.3.4.3, “Design Basis Accident Radiological Consequence Analyses.”

The guidance in NUREG-0696, Section 2.6 “Habitability,” addresses the potential need to move the TSC functions if the TSC becomes uninhabitable. The staff notes that as part of COL Item 13.3-3, the applicant has included a provision in the DCA for a future COL applicant to address the need to designate a location in the COL application emergency plan for the transfer of TSC functions in the event of the loss of ventilation to the TSC, or if the TSC otherwise becomes uninhabitable. This is consistent with the guidance provided in NUREG-0696, Section 2.6, and is acceptable.

The staff finds the applicant’s response to RAI 8925, Questions 13.03-2 and 13.03-3 acceptable, and concludes that for the matters reviewed in this section, the information provided in the application is consistent with the guidance identified in NUREG-0696, Supplement 1 to NUREG-0737, and the SRP. Since the information is consistent with the applicable guidance it is sufficient to meet the associated regulatory requirements. Therefore, the staff determined that the information reviewed in this section meets the applicable requirements of 10 CFR 50.47(b)(8) and (11), and Subsection IV.E.3 and IV.E.8 of Appendix E to 10 CFR Part 50.

#### *13.3.4.2 Operational Support Center*

The applicant has identified COL Item 13.3-1, which requires the COL applicant to describe the onsite OSC. The staff finds that the inclusion of a COL information item associated with the OSC is acceptable because the COL applicant must describe the OSC to comply with 10 CFR 50.47(b)(8). The NRC will evaluate the acceptability of a future COL applicant's proposed OSC as part of the COL application process.

*13.3.4.3 Emergency Operations Facility*

The applicant has identified COL Item 13.3-2, which requires the COL applicant to describe the site-specific EOF. The staff finds that the inclusion of a COL information item associated with the EOF is acceptable because the COL applicant must describe the EOF to comply with 10 CFR 50.47(b)(8) and 10 CFR Part 50, Appendix E. The NRC will evaluate the acceptability of a future COL applicant's proposed EOF as part of the COL application process.

*13.3.4.4 Technical Support Center Engineering Workstations*

In Revision 1 of DCA Part 2, Tier 2, Section 13.3, the applicant noted that the TSC includes engineering workstations as described in DCA Part 2, Tier 2, Section 7.2.13.7. The staff issued RAI 8925, Question 13.03-3 (ADAMS Accession No. ML17206A098), asking the applicant to provide additional information on the safety display and indication system (SDIS) and compare it to the safety parameter display system (SPDS) and the ERDS. In its response to RAI 8925, Question 13.03-3, Part i (ADAMS Accession No. ML17264B172), NuScale stated the following:

> *The reference to the safety display and indication system (SDIS) in Section 13.3 is being changed to refer to the technical support center (TSC) engineering workstations, which are part of the module control system (MCS) and plant control system (PCS). The post-accident monitoring (PAM) variables displayed on SDIS in the main control room are also displayed on MCS or PCS and are available to the TSC engineering workstations. The PAM instrumentation includes the required functions, range, and accuracy for each variable monitored. Variables and their type classification are based on their accident management function as identified in abnormal operating procedures, emergency operating procedures, and emergency procedure guidelines. The PAM Type B, C, D, and E variables are summarized in FSAR Tier 2 Table 7.1-7.*
>
> *The engineering workstation PCS displays in the technical support center (TSC) provide monitoring functionality to plant processes and equipment.*
>
> *There is a unidirectional communication interface between the MCS and PCS networks and the plant network. The MCS and PCS systems provide monitoring data via one-way communication interfaces to the plant network which provides data recording, trending, and historical retention that can be called up on the TSC engineering workstations*.

The guidance in NUREG-0696, Section 2.8, "Instrumentation, Data System Equipment, and Power Supplies," calls for the TSC to have equipment to gather, store, and display data needed in the TSC to analyze plant conditions. The system which provides this capability is generically referred to as the SPDS. The applicant has explained that a TSC engineering workstation is provided which serves the same function as the generic SPDS. The TSC engineering workstation, which is part of the PCS, receives information from MCS and PAM, and has the capability to gather, store, and display data needed in the TSC. The staff finds that the TSC engineering workstations satisfy the guidance in NUREG-0696 and provide for equipment capable of gathering, storing, and displaying data needed in the TSC to analyze plant conditions. For this reason, the staff finds the applicant's response to RAI 8925, Questions 13.03-3 acceptable, and determined that this information meets the applicable requirements of 10 CFR 50.47(b)(8) and 10 CFR Part 50, Appendix E, Section IV.E.8.a(i).

*13.3.4.5 Emergency Response Data System*

In DCA Part 2, Tier 2, Section 13.3, the applicant stated,

> *An emergency response data system compliant with Section VI of 10 CFR [Part] 50, Appendix E, provides a direct near-real-time electronic data link of selected parameters between the onsite computer system and the NRC Operations Center in the event of an emergency.*

The staff issued RAI 8925, Question 13.03-3, Part i (ADAMS Accession No. ML17206A098), as discussed in SER Section 13.3.4.4. In its response to RAI 8925, Question 13.03-3, Part i (ADAMS Accession No. ML17264B172), NuScale stated the following with regard to the ERDS:

> *Additionally, there is a link from the plant network to the NRC emergency response data system (ERDS) via dedicated communications servers that connect to the plant network and provide data communication of required plant data to off-site emergency response facilities as shown in FSAR Tier 2 Figure 7.0-1.*

The staff also issued RAI 8925, Question 13.03-3, Part iii (ADAMS Accession No. ML17206A098), asking the applicant to provide additional information on the data variables to be transmitted to the NRC. In its response to RAI 8925, Question 13.03-3, Part iii (ADAMS Accession No. ML17264B172), NuScale noted, in part, that "the capability exists for the data variables identified in FSAR Table 7.1-7 to be communicated to the NRC ERDS. The specific data to be transmitted via ERDS will be identified as part of emergency planning (COL Item 13.3-3)." In consideration of COL Item 13.3-3, the NRC staff finds it acceptable for a future COL applicant to address the specific data transmitted through the ERDS as part of its emergency plan.

The staff concludes that the application has provided an adequate description of a direct near real-time electronic data link between the onsite computer system and the NRC's ERDS that provides for the automated transmission of a limited data set of selected parameters. This meets the requirements in Section VI, "Emergency Response Data System," of Appendix E to 10 CFR Part 50 to have a connection to the ERDS.

*13.3.4.6 Enhanced Emergency Response Capabilities for Beyond-Design-Basis Events*

The applicant has identified COL Items 20.4-1 through 20.4-6 to address enhanced emergency response capabilities for beyond-design-basis events (BDBEs). These COL Items are being addressed in Chapter 20 of the SER.

*13.3.4.7 Decontamination Facilities*

Decontamination facilities are provided to remove or reduce radioactive contaminants from plant equipment, protective clothing, and personnel. As described in DCA Part 2, Tier 2, Section

12.1.2.3, and the associated section of the SER, personnel and equipment decontamination areas are located in the annex building.

The staff finds that the information provided in the application on the decontamination rooms is consistent with the guidance in Section II.K, “Radiological Exposure Control,” of NUREG-0654 that such a facility should be provided. For this reason, the staff determined that this information meets the applicable requirements of 10 CFR 50.47(b)(8) and 10 CFR Part 50, Appendix E, Section IV.E.3.

*13.3.4.8 Process Sampling System*

DCA Part 2, Tier 2, Section, 9.3.2, states that “[t]he function of the process sampling system (PSS) is to provide the means to obtain representative liquid and gaseous samples from various primary and secondary process streams and components for monitoring and analyzing the chemical and radiochemical conditions. The PSS capability is used during normal plant operations and following accident conditions without the need for a dedicated post-accident sampling system.” The corresponding section of the SER provides the staff’s evaluation of the process sampling system.

The NRC considers this area of the SER an open item until review of the process sampling system is completed, and identifies it as **Open Item 13.3-1**. The capability to obtain a post-accident sample is an interface item between SRP Section 9.3.2, “Process Sampling Systems,” and SRP Section 13.3. The guidance in NUREG-0654, Evaluation Criteria I.2, includes a post-accident sampling capability as part of the methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition to meet the regulatory requirements found in 10 CFR 50.47(b)(9). If the PSS is determined to be acceptable as a means for obtaining a post-accident sample, then this open item, associated with the process sampling system will be resolved and NRC staff will be able to close this item and finalize Section 13.3.4.8.

*13.3.4.9 Inspections, Tests, Analyses, and Acceptance Criteria*

The applicant did not provide EP-specific ITAAC for the design and specified COL Item 14.3-1 for a future COL applicant to address ITAAC, as listed in SER Section 13.3.5. The NRC will evaluate the acceptability of a future COL applicant’s proposed ITAAC as part of the COL application process. SER Section 14.3.10 also mentions the COL item.

### 13.3.5 Combined License Information Items

Table 13.3-5 lists COL information items related to EP, as provided in DCA Part 2, Tier 2, Sections 13.3, 14.3, and 20.4. DCA Part 2, Tier 2, Section 13.3, provides information related to those aspects of EP that are nonsite-specific EP features and that are technically relevant to the design (i.e., facilities and equipment). However, the COL applicant that references the certified standard design is responsible for the programmatic aspects of EP and emergency preparedness. The NRC staff reviewed COL Information Items 13.3-1 through 13.3-3 and found them to be consistent with the regulatory standards in 10 CFR 50.47(b) and 10 CFR Part 52 and with the guidance in the SRP. Therefore, the proposed COL information items are acceptable.

COL Items 20.4-1 through 20.4-6 are being addressed in Chapter 20 of the SER.

**Table 13.3-5 NuScale COL Information Items for Sections 13.3, 14.3, and 20.4**

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| COL Item 13.3-1 | A COL applicant that references the NuScale Power Plant design certification will provide a description of the onsite operational support center (OSC) including the direct communication system or systems between the OSC and the control room. | 13.3 |
| COL Item 13.3-2 | A COL applicant that references the NuScale Power Plant design certification will provide a description of an emergency operations facility for management of overall licensee emergency response and which complies with the guidance in NUREG-0696, “Functional Criteria for Emergency Response Facilities,” NUREG-0737 Supplement 1, “Clarification of TMI Action Plan Requirements – Requirements for Emergency Response Capability,” and NSIR/DPR-ISG-01, “Interim Staff Guidance—Emergency Planning for Nuclear Power Plants.” | 13.3 |
| COL Item 13.3-3 | A COL applicant that references the NuScale Power Plant design certification will provide a comprehensive emergency plan in accordance with 10 CFR 50.47, 10 CFR 50, Appendix E, 10 CFR 52.48, and 10 CFR 52.79(a)(21). | 13.3 |
| COL Item 14.3-1 | A COL applicant that references the NuScale Power Plant design certification will provide the site-specific selection methodology and Inspections, Tests, Analyses, and Acceptance Criteria for emergency planning. | 14.3 |
| 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. | 20.4 |
| 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 |
| 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 |
| 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. | 20.4 |

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| 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, severe accident mitigation guidelines, and extensive damage 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 |
| 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. | 20.4 |

### 13.3.6 Conclusion

With the exception of **Open Item 13.3-1** identified in Section 13.3.4.8 concerning the capability to obtain a post-accident sample, the staff concludes, on the basis of its review as described above, that the applicant has adequately addressed the EP design-related features for the NuScale power plant design. Therefore, with the exception of application information related to post-accident sampling, the information is acceptable and meets the applicable requirements listed in SER Section 13.3.3. When the process sampling system review is completed, the NRC staff will update this conclusion to reflect the disposition of **Open Item 13.3-1**.

## 13.4 Operational Programs

### 13.4.1 Introduction

COL applicants are required by 10 CFR 52.79 to describe operational programs, but similar requirements do not exist for DCAs. NuScale provided a COL item describing a future COL applicant’s obligation to provide operational program information. Staff evaluated this section using Draft Revision 4 of SRP 13.4, which was published in September 2018.

### 13.4.2 Summary of Application

In DCA Part 2, Tier 2, Section 13.4, “Operational Programs,” the applicant provided COL information item 13.4-1 which stated that a COL applicant that references the NuScale power plant design certification will provide site-specific information, including an implementation schedule, for the listed operational programs.

**DCA Part 2, Tier 1:** There is no Tier 1 information for the operational programs.

**DCA Part 2, Tier 2:** DCA Part 2, Tier 2, Section 13.4, provides the applicant’s COL information item on operational programs.

**ITAAC:** There are no ITAAC associated with the operational programs.

**Technical Specifications:** There are no TS associated with the operational programs.

**Technical Reports:** There is no TR associated with the operational programs.

### 13.4.3 Regulatory Basis

There are no regulatory requirements regarding operational programs for a design certification applicant. A DC applicant is required to have a quality assurance program meeting the requirements of 10 CFR Part 50, Appendix B. Chapter 17 of this SER describes how the applicant meets that requirement. Similarly, Section 13.6 of this SER describes how the applicant meets the information security requirements of 10 CFR Part 73.

### 13.4.4 Technical Evaluation

Staff compared the list of operational programs in COL Item 13.4-1 with the recommended list in SRP Section 13.4. Staff found that the applicant's list included all of the programs the SRP recommended.

### 13.4.5 Combined License Information Items

Table 13.4-5 lists a COL information item related to Operational Programs, from DCA Part 2, Tier 2, Table 1.8-2: "Combined License Information Items".

**Table 13.4-5 NuScale COL Information Items for Section 13.4**

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| COL Item 13.4-1 | A COL applicant that references the NuScale Power Plant design certification will provide site-specific information, including implementation schedule, for operational programs:<br><br>• Inservice inspection programs (refer to Section 5.2, Section 5.4, and Section 6.6)<br>• Inservice testing programs (refer to Section 3.9 and Section 5.2)<br>• Environmental qualification program (refer to Section 3.11)<br>• Pre-service inspection program (refer to Section 5.2 and Section 5.4)<br>• Reactor vessel material surveillance program (refer to Section 5.3)<br>• Pre-service testing program (refer to Section 3.9.6, Section 5.2, and Section 6.6)<br>• Containment leakage rate testing program (refer to Section 6.2)<br>• Fire protection program (refer to Section 9.5)<br>• Process and effluent monitoring and sampling program (refer to Section 11.5)<br>• Radiation protection program (refer to Section 12.5)<br>• Non-licensed plant staff training program (refer to Section 13.2)<br>• Reactor operator training program (refer to Section 13.2)<br>• Reactor operator requalification program (refer to Section 13.2) | 13.4 |

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| | • Emergency planning (refer to Section 13.3)<br>• Process control program (PCP) (refer to Section 11.4)<br>• Security (refer to Section 13.6)<br>• Quality assurance program (refer to Section 17.5)<br>• Maintenance rule (refer to Section 17.6)<br>• Motor-operated valve testing (refer to Section 3.9)<br>• Initial test program (refer to Section 14.2) | |

### 13.4.6 Conclusion

The staff determined that the COL item listed above is acceptable because the applicant appropriately directs the COL applicant to develop operational programs, consistent with the list provided in SRP Section 13.4, draft Revision 4.

## 13.5 Plant Procedures

### 13.5.1 Introduction

A COL holder's plant procedures include (1) administrative procedures that provide for administrative control over safety-related activities for the operation of the facility, (2) operating procedures and EOPs used to ensure that routine operating, off-normal (i.e., abnormal), and emergency activities are conducted in a safe manner, and (3) procedures for other safety-related plant operating activities, including related maintenance activities, that the operating program or EOP program does not cover.

The staff reviewed the DCA to (1) evaluate the acceptability of COL information items pertaining to COL applicant descriptions of plant procedures, (2) evaluate the acceptability of COL information items pertaining to the COL applicant's program for development and implementation of plant procedures, and (3) evaluate the technical adequacy of the NuScale generic technical guidelines (GTGs) and determine their acceptability as a basis for development of COL applicant plant-specific technical guidelines (P-STGs).

### 13.5.2 Summary of Application

**DCA Part 2, Tier 1:** This section does not include a technical review of Tier 1 information; however, the staff verified that the GTGs were consistent with the Tier 1 Accident Monitoring variables.

**DCA Part 2, Tier 2:** Procedure development is not within the scope of the NuScale DCA. This responsibility resides with the COL applicant. DCA Part 2, Tier 2, Revision 1, Section 13.5, "Plant Procedures," specifies COL information items for the COL applicant to describe the administrative, operating, and maintenance procedures. The applicant submitted the NuScale Technical Report "NuScale Generic Technical Guidelines," TR-1117-57216 in a supplemental response to RAI 8738, Question 13.05.02.01-1, dated November 30, 2017 (ADAMS Accession Nos. ML17334B739 (letter) and ML17334B822 (GTG)).

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

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

**Technical Reports:** There is one TR for this area of review:

- TR-1117-57216, "NuScale Generic Technical Guidelines," Draft Revision 1, issued August 16, 2018 (ADAMS Accession No. ML18228A855).

### 13.5.3 Regulatory Basis

SRP Section 13.5.1.1, "Administrative Procedures—General," and SRP Section 13.5.2.1, "Operating and Emergency Operating Procedures," identify, in part, the relevant NRC regulatory requirements for plant procedures and the associated acceptance criteria.

The applicable regulatory requirements for plant procedures are as follows:

- 10 CFR 50.34(f)(2)(ii), as it pertains to the TMI-related requirement for applicants to establish a program to begin during construction and to follow into operation for assessing and improving plant emergency procedures.

- 10 CFR 50.34(f)(3)(i), as it pertains to the TMI-related requirement to provide administrative procedures that evaluate and provide feedback on operating experience, design experience, and construction experience.

- 10 CFR 50.40(a), as it pertains to the issuance of a COL under 10 CFR Part 52 based on considerations of whether the applicant has developed operating procedures that are sufficient to provide reasonable assurance that the nuclear power plant can be safely operated.

- 10 CFR Part 50, Appendix B, as it pertains to the establishment of criteria for the development, approval, and control of procedures for all activities affecting quality.

- 10 CFR 52.79(a)(27), (29)(i), and (29)(ii), as they pertain to information that must be included in the FSAR submitted as part of the application for a COL, specifically, (1) the managerial and administrative controls associated with procedures used to perform activities that affect the quality of SSCs covered under the quality assurance program, as established in 10 CFR Part 50, Appendix B, and (2) plans for the development and implementation of plant procedures used for emergency operations (other than emergency planning) and the conduct of normal operations, including maintenance, surveillance, and periodic testing of SSCs.

The related acceptance criteria are as follows:

- RG 1.33, "Quality Assurance Program Requirements (Operation)," Revision 3, issued June 2013.

- Appendix A, "Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors," to American National Standards Institute (ANSI)/American

Nuclear Society (ANS) 3.2-2012, "Managerial, Administrative, and Quality Assurance Controls for Operational Phase of Nuclear Power Plants."

- Section III of SRP Section 13.5.1.1, Revision 2, issued August 2016.
- SRP Section 13.5.2.1, Revision 2, issued March 2007.
- Section I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents," of NUREG-0737, issued November 1980.
- Section 7, "Upgrade Emergency Operating Procedures (EOPs)," of Supplement 1 to NUREG-0737, issued January 1983.
- NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures—Resolution of Comments on NUREG-0799," issued August 1982.

## 13.5.4 Technical Evaluation

DCA Part 2, Tier 2, Revision 1, Section 13.5, identifies procedure development as the COL applicant's responsibility. This section (1) evaluates the acceptability of the COL information items for plant procedures and (2) evaluates the technical adequacy of the NuScale GTGs and determines their acceptability for use in the development of COL applicant P-STGs.

### *13.5.4.1 Combined License Information Items*

The NRC requires a COL applicant that references the NuScale certified design to submit the site-specific information described in the COL information items at the COL stage.

#### *13.5.4.1.1 Administrative Procedures*

SRP Section 13.5.1.1 describes administrative procedures as those that provide for administrative control over safety-related activities for the operation of the facility. The staff's review of the NuScale DCA using SRP Section 13.5.1.1 focused on the evaluation of COL information items pertaining to administrative procedures. COL Item 13.5-1 in DCA Part 2, Tier 2, Revision 1, Section 13.5.1, "Administrative Procedures," requires the COL applicant to describe site-specific procedures that provide administrative control for activities that are important for the safe operation of the facility consistent with the guidance in RG 1.33, Revision 3, which endorses ANSI/ANS 3.2-2012. Accordingly, the staff determined that COL Item 13.5-1 is acceptable.

SRP Section 13.5.1.1 provides the technical rationale for applying SRP acceptance criteria to the establishment of a program for the development and implementation of administrative procedures. DCA Part 2, Tier 2, Revision 1, Section 13.5.1, COL Item 13.5-4, requires the COL applicant to provide a program for the development and implementation of the administrative procedures and a plan for the development, implementation, and control of administrative procedures, including preliminary schedules for preparation and target completion dates. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures. The staff determined that COL Item 13.5-4 is acceptable because it is consistent with provisions in SRP Section 13.5.1.1.

*13.5.4.1.2 Operating and Maintenance Procedures*

SRP Section 13.5.2.1 states that the applicant's SAR should describe the different classifications of procedures that the operators will use in the control room and locally in the plant for plant operations. DCA Part 2, Tier 2, Section 13.5.2, "Operating and Emergency Operating Procedures," Revision 1, COL Item 13.5-2, requires the COL applicant to describe the site-specific procedures that operators use in the MCR and locally in the plant, including normal operating procedures, abnormal operating procedures, and EOPs. The COL applicant will also describe the classification system for these procedures and the general format and content of the different classifications. The staff determined that COL Information Item 13.5-2 is acceptable because it requires the COL applicants to describe the different classifications of the site-specific procedures that licensed operators and non-licensed operators perform.

SRP Section 13.5.2.1 states that the applicant's SAR should describe plant procedures that will be used by the operating organization (i.e., plant staff). DCA Part 2, Tier 2, Section 13.5.2.2, Revision 1, COL Item 13.5-3, requires COL applicants to describe the site-specific program for developing maintenance and other operating procedures. It also requires COL applicants to describe how these procedures are classified, including the general format and content of the different classifications. This COL information item includes a list of the categories of procedures to be included. The staff determined that COL Item 13.5-3 is acceptable because it requires the COL applicant to describe the different classifications of procedures for developing maintenance and other operating procedures (i.e., procedures for activities not procedurally covered under the operating procedures or EOPs identified in Section I.1 of SRP Section 13.5.2.1).

SRP Section 13.5.2.1 provides the technical rationale for applying SRP acceptance criteria to the establishment of programs for the development and implementation of operating and maintenance procedures. Thus, the applicant should include COL information items that require the COL applicant to provide programs for development and implementation of the operating and maintenance procedures. DCA Part 2, Tier 2, Section 13.5.2, "Operating and Maintenance Procedures," Revision 1, COL Item 13.5-5, requires a COL applicant to provide a plan for the development, implementation, and control of operating procedures, including preliminary schedules for preparation and target completion dates. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures. COL Item 13.5-7 requires a COL applicant to provide a plan for the development, implementation, and control of EOPs, including preliminary schedules for preparation and target completion dates. In its submittal, the COL applicant is to include the procedures generation package, which comprises the P-STGs, a plant-specific writer's guide for preparing EOPs based on the P-STGs, a description of the program for verification and validation of the EOPs, and a description of the program for training operators on the EOPs. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures. COL Item 13.5-8 requires a COL applicant to provide a plan for the development, implementation, and control of maintenance and other operating procedures, including preliminary schedules for preparation and target completion dates. Additionally, the COL applicant will identify the group or groups within the operating organization that will be responsible for maintaining and following these procedures. The staff concludes that COL Items 13.5-5, 13.5-7, and 13.5-8 are acceptable because they require the COL applicant to provide programs for the development, implementation, and control of operating and maintenance procedures.

### *13.5.4.2 NuScale Generic Technical Guidelines*

NUREG-0737, Section I.C.1, and NUREG-0737, Supplement 1, Section 7, both provide that technical guidelines for development of the EOPs should be prepared and submitted to the NRC for review.

SRP Section 13.5.2.1 states that COL applicants can use design-specific GTGs (also referred to as emergency operating guidelines) to develop P-STGs from which they will develop their EOPs. The design certification applicant is responsible for the GTGs.

The NuScale GTGs, TR-1117-57216, are incorporated by reference by DCA, Part 2, Tier 2, Revision 1, Table 1.6-2. The staff's evaluation of the NuScale GTGs below is based on TR-1117-57216-P (proprietary), Draft Revision 1 (ML18228A855) that was submitted on the docket in August 16, 2018, following a series of public and closed meetings held in February 2018 in which the staff discussed its initial review of technical report TR-1117-57216, Rev. 0, with the applicant. Further information on these meetings is provided in a comprehensive meeting summary dated June 21, 2018 (ML18169A172).

Section 6.0, "Validation," of TR-1117-57216, Draft Revision 1, states the following:

- Draft Revision 1 of the GTGs was issued with the understanding that the selected setpoints are based on analysis and engineering judgement and that instrument selection is not complete in all cases.

- Draft Revision 1 of the GTGs has been incorporated into the NuScale simulator, has undergone limited scenario testing, and is not considered to be validated.

- Current actions and logic have been used in simulations to provide high confidence in the mitigating strategies.

- The GTGs submitted with the DCA will be validated during performance of the NuScale ISV testing.

- The GTGs will be updated to document any changes resulting from the ISV testing.

Given this, the staff's review of the NuScale GTGs focused on (1) the three critical safety functions (CSFs) defined for the NuScale power plant, (2) the methodology used to identify operator actions, and (3) the CSF flowchart logic and operator actions necessary to assess and maintain the CSFs, including the bases. The staff issued six RAIs with a total of 17 questions that identified issues associated with GTG structure and use, assignment of CSFs, final setpoint selection information, PAM variable information inconsistencies, CSF flowchart functional decision logic, and incomplete information. As discussed below, the applicant has submitted satisfactory responses to these RAIs, but the staff is presently unable to conclude that the NuScale GTGs are acceptable for use as a basis for the development of COL applicant P-STGs. A determination of acceptability is contingent upon the achievement of satisfactory results and the subsequent incorporation of any necessary changes to the GTGs and the associated PAM variables that result from ISV testing and validation activities. These ISV activities will occur during the review of the DCA and are being tracked as **Open Item 13.5-1**.

The NuScale GTGs cover the content in existing industry operating procedure classes, including EOPs, severe accident management guidelines, and ELAP, and loss of large area (LOLA) event guidance. The GTGs describe the actions assumed within the analysis of accident conditions that occur outside of the design basis. The NuScale transient and accident analyses conducted for design-basis events (DBEs) do not identify any required operator actions. Accordingly, the GTGs contain those actions that have been credited by probabilistic risk assessments (PRAs) and described in the DCA Part 2, to respond to BDBEs such as severe accidents, ELAP, and LOLAs. In addition, the GTGs include some actions based on task analysis activities performed as part of the human factors engineering program.

The NuScale GTGs are "symptom-based" procedural guidelines that allow the operator to respond directly to indications presented as part of an accident progression. Symptom-based procedures do not require the operator to attempt to diagnose the accident in progress; instead, they 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 necessary to maintain them within the prescribed limits instead of responding in a predetermined sequence based on a diagnosed accident. Legacy plant generic guidelines include "event-based" descriptions (i.e., events based on the transient and accident analysis events and associated operator actions described under "Transient and Accident Analyses" for a specific design). Because the NuScale design has no credited manual actions in DCA Part 2, Tier 2, Chapter 15, the symptom-based approach enables mitigating strategies to be effective with multiple failures, regardless of the combination. The evaluation of symptoms in the NuScale GTGs is grouped into critical safety functions (CSFs). The following three CSFs, listed in order of priority, have been defined for the NuScale design:

(1) containment integrity (CI);

(2) reactivity; and

(3) core heat removal (CHR).

In its review of the NuScale GTGs, the staff evaluated the selected CSFs, the methodology used to identify operator actions, and the CSF flowchart logic and operator actions necessary to assess and maintain the CSFs, including the bases.

*13.5.4.2.1 Critical Safety Functions*

Section 4.2, "Critical Safety Functions," of the NuScale GTGs identify the CI, reactivity, and CHR CSFs. The NRC staff compared the CSFs identified in the NuScale GTGs against existing guidance in Institute of Electrical and Electronics Engineers (IEEE) 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," which RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 4, issued June 2006, endorses, and against the previously approved emergency response guidelines (ERGs) of the Pressurized-Water Reactor Owners Group. This comparison shows that the reactivity, CHR, and CI CSFs included in the NuScale GTGs are consistent with industry guidance and previously approved ERGs. In addition, the NuScale GTGs provide subfunctions within the CHR CSF that address functions analogous to the heat sink and inventory CSFs identified in previously approved ERGs (i.e., the CHR CSF addresses the alignment and level requirements for the safety systems to perform their functions).

Section 4.2 of the GTGs briefly discusses the reasons that the secondary heat sink CSF is defined for other pressurized-water reactor (PWR) designs but not for the NuScale plant design. This section also explains that the applicant did not include RCS integrity as a stand-alone CSF because this function is monitored by the core heat removal safety function. The primary actuation that mitigates a loss of RCS integrity is the ECCS. When the ECCS actuation valves open, a natural circulation path is created, allowing heat to be removed by the containment vessel to the ultimate heat sink.

Section 4.2 of the GTGs does not provide any additional discussion, insights, or evaluations on the suitability of excluding the inventory CSF (defined for other PWR designs but not the NuScale design) other than to make the following statement:

> Additional safety functions are not needed due to the simplicity and reliance on passive systems in the NuScale design.

Similarly, DCA Part 2, Tier 2, Chapter 7, "Instrumentation and Controls," does not provide any additional discussion, insights, or evaluations on the suitability of excluding the RCS integrity CSF or the inventory CSF from the NuScale design, other than to state the following in DCA Part 2, Tier 2, Section 7.1.1.2.2, "Post-Accident Monitoring":

> *The remove fuel assembly heat critical safety function includes the aspects of reactor coolant system (RCS) integrity. This is due to the integral nature of emergency core cooling system (ECCS) and RCS integrity—actuating ECCS opens valves to allow steam release to the containment and return of water back to the RCS—it is done to maintain core cooling and protect the fuel clad fission product barrier. This is automatically actuated when there is an existing loss of RCS as indicated by low reactor pressure vessel (RPV) riser water level or high containment water level.*

Previously approved ERGs for other PWR designs verified the integrity of the RCS to ensure that the pressure-temperature limits of the RCS are not violated and include an inventory control safety function. The NuScale design does not include a separate function for RCS inventory because it is considered an integral part of the core heat removal safety function. In the NuScale design, the ECCS provides passive core cooling by retaining primary coolant inside the containment vessel (CNV), which facilitates the transfer of heat from the fuel to the UHS. In regard to the absence of an RCS integrity CSF, the applicant stated the following:

> *RCS integrity is addressed under the core heat removal CSF by ensuring that the low temperature overpressure (LTOP) system automatically actuates when required. The LTOP system fully actuates ECCS when pressure is above the temperature dependent pressure setpoint. When ECCS actuates, an intentional hydraulic connection between the RCS and CNV occurs which establishes a natural circulation heat removal path outside of the RCS, but within the containment. A pressurized thermal shock event is not credible at NuScale because of the following factors:*
>
> 1. *All sources of makeup are isolated by the containment isolation system.*
>
> 2. *Actuation of the ECCS system precludes pressurization of the RCS system.*

3. *The NuScale reactor pressure vessel is designed to withstand the maximum passive system cooldown rate.*

The staff finds that the three CSFs that NuScale provided are consistent with industry guidance and previously approved ERGs. The staff concludes that separate CSFs for integrity control and inventory control are not necessary for the NuScale design because they are inherent to the selected CSFs. The staff evaluated the applicant's identification of the CSFs in SER Section 18.7 and found that the applicant adequately described CSFs and applied guidance for the identification of design specific CSFs.

*13.5.4.2.2 Methodology to Identify Operator Actions*

Section 3.0, "Methodology," of the NuScale GTGs discusses the methodology used to determine appropriate mitigating strategies and actions based on plant design. The NuScale GTGs state that operator actions are identified by reviewing the following:

- the plant response to DBEs described in DCA Part 2, Tier 2, Chapter 15;
- the I&C failure defense-in-depth analysis described in DCA Part 2, Tier 2, Chapter 7;
- operator actions assumed in the BDBE analysis described in DCA Part 2, Tier 2, Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation";
- operator actions assumed in a BDBE evaluation described in DCA Part 2, Tier 2, Chapter 20;
- multiunit considerations described in DCA Part 2, Tier 2, Chapter 21;
- NuScale generic technical specifications
- system requirements and limitations as defined in system description documents; and
- human factors engineering task analysis results described in DCA Part 2, Tier 2, Section 18.4.

DCA Part 2, Tier 2, Section 15.0.0.5, "Limiting Single Failures," states that no operator actions are required for 72 hours for any DBE. Plant automation places and maintains the unit in a safe state (i.e., a condition reached when the initiating event is mitigated) for at least 72 hours after a DBE with assumed failures. The staff reviewed the analyses in DCA Part 2, Tier 2, Chapter 15, and determined that operator action is not credited to mitigate events in the transient and accident analyses. In addition, NuScale incorporated by reference TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," Revision 0 (ADAMS Accession No. ML15306A065; October 2015), which states, in part, that operator action is not required to achieve safe shutdown, core cooling, containment isolation and integrity, and reactor coolant pressure boundary integrity in response to DBEs. Although no operator action is credited for the mitigation of DBEs, the NuScale GTGs direct the operators to verify the automatic actuation of safety systems. Because no operator actions are required to mitigate

DBEs, the staff finds that the DCA Part 2, Tier 2, Chapter 15, transient and accident analyses are acceptable for supporting the automatic action verification provided in the NuScale GTGs.

The I&C analysis of diversity and defense-in-depth did not identify any manual operator actions for a DBE. DCA Part 2, Tier 2, Section 7.1.5.1.14, "Guideline 14—Manual Operator Action," discusses manual operator action as pertaining to diversity and defense-in-depth. DCA Part 2, Tier 2, Section 7.1.5.1.14, clarifies that the NuScale design has no Type A variables (i.e., variables that provide information essential for the direct accomplishment of CSFs that require manual action) but that manual control blocks (i.e., safety-related manual switches) provide an independent and diverse method of manually actuating the automatic safety-related functions. Based on the discussion provided in DCA Part 2, Tier 2, Section 7.1.5.1.14, the staff finds that the only operator actions associated with diversity and defense-in-depth are the automatic action verifications in the NuScale GTGs.

The PRA analysis (documented in DCA Part 2, Tier 2, Chapter 19) examined BDBEs that lead to core damage and identified seven human actions, including two that are important human actions. Table 4-1, "PRA Credited Operator Actions to Mitigate Core Damage," of the NuScale GTGs lists these actions. The staff responsible for reviewing DCA Part 2, Tier 2, Chapter 19, with regard to the PRA and severe accident evaluation considered the human actions important to the mitigation of BDBEs listed in Table 4-1 in Revision 0 of the GTGs and found the set of actions to be consistent with those actions modeled in the PRA. In its review of the human reliability portion of the PRA, the staff considered whether additional human actions may be important to the mitigation of BDBEs, but it did not identify any such actions. Therefore, the staff has concluded that the set of human actions for mitigation of BDBEs described in Revision 0 of the GTGs is a complete set that should be addressed in the development of the P-STGs.

Additional evaluations for BDBEs that the applicant performed and documented in DCA Part 2, Tier 2, Chapter 20, identified two actions meant to prevent core damage during an ELAP or to mitigate core damage and minimize the spread of radioactive release during a LOLA event. Table 4-3, "Operator Actions Described in DCA Part 2, for Chapter 20 Beyond-Design-Basis Events," of the NuScale GTGs lists these actions. The staff responsible for reviewing DCA Part 2, Tier 2, Section 20.1, "Mitigating Strategies for Beyond Design-Basis External Events," with regard to ELAP and DCA Part 2, Tier 2, Section 20.2, "Loss of Large Areas of the Plant due to Explosions and Fires," with regard to LOLAs resulting from explosions or fires has evaluated the operator actions important to the mitigation of the BDBEs listed in Table 4-3 of Revision 0 of the NuScale GTGs and found that (1) the operator actions were consistent with those actions described in DCA Part 2, Tier 2, Section 20.1, and TR-0816-50797-NP, "Mitigation Strategies for Extended Loss of AC Power Event," Revision 0 (November 2016) (ADAMS Accession No. ML17005A120), and (2) the operator actions were consistent with those actions described in DCA Part 2, Tier 2, Section 20.2, and Nuclear Energy Institute 06-12, "B.5.b Phase 2 and 3 Submittal Guideline," Revision 3. Therefore, the staff has concluded that the operator actions for mitigation of BDBEs described in Revision 0 of the GTGs are a complete set that should be addressed in the development of the P-STGs.

In addition to the manual actions specifically required by analysis, manual actions have been included when they are determined to be appropriate by subject matter expert experience or to maintain consistency among the mitigating strategies. For example, the PRA analysis only credits manual actuation of the ECCS and the containment isolation signal (CIS) when these systems fail to actuate as designed. For consistent performance and standardization of operator actions, when any engineered safety features function fails to actuate, operators will take action to manually perform the actuation.

*13.5.4.2.3 Critical Safety Function Flowchart Logic and Operator Actions*

Section 4.3, "Structure and Use," of the NuScale GTGs describes how the GTGs are formatted, displays and states options implementation including both electronic and paper copy methods and any differences in these methods.

Section 4.8, "Implementation Strategy," provides detailed guidance regarding implementation and execution strategies for CSF flowcharts including how operators should prioritize both the CSFs and defense-in-depth functions and when and how often operators should evaluate these functions.

The staff finds that the structure and instructions for use of the flowcharts and the implementation strategy of the NuScale GTGs are acceptable because they (1) are complete and logically structured, (2) appropriately prioritize safety and defense-in-depth functions, (3) adequately describe initial and follow-up evaluation when CSFs are challenged or not met, (4) can be practically implemented, and (5) provide adequate design-specific information for a COL applicant to use in the development of P-STGs and subsequent EOPs.

*Containment Integrity Safety Function*

Section 5.1, "Containment Integrity Safety Function," of the GTGs describes purpose of the CI safety function and the actions necessary to address the CI safety function. Specifically, Section 5.1 contains (1) the CI safety function flowchart that depicts the functional logic and specifies the operator actions necessary to assess and maintain the CSF and (2) the accompanying bases information.

DCA Part 2, Tier 2, Section 7.1.1.2.2, "Post-Accident Monitoring," states that the containment water level and the inside bioshield area radiation monitor are Type B variables that are used to provide direct indication and to support the CI CSF. Further, DCA Part 2, Tier 2, Chapter 7, Table 7.1-7, "Summary of Type A, B, C, D, and E Variables," also identifies the containment water level and the inside bioshield area radiation monitor as Type B variables. The containment water level and the inside bioshield area radiation monitor are not evaluated in the CI safety function flowchart. Because DCA Part 2, Tier 2, Chapter 7, identifies the containment water level and the inside bioshield area radiation monitor as Type B variables that provide direct indication and that support the CI CSF, the staff questioned why the CI safety function flowchart does not use these variables as decision variables to determine whether the CI CSF can be met. As a result, the NRC issued RAI 9430, Question 13.05.02.01-9 (ADAMS Accession No. ML18101A640), to address this concern. In its response to RAI 9430, Question 13.05.02.01-9 (ADAMS Accession No. ML18162A359), the applicant explained that it used the guidance in RG 1.97, Revision 4, to select the PAM variables that it submitted with the application. RG 1.97 recommends compiling a conservative list of variables that would then be revised as procedures are developed. The applicant explained that during GTG development, the containment water level was found to be a poor indicator of CI during transients because of water volume changes as a result of temperature changes. Therefore, the containment water level was not included in the GTGs for use during actions for the CI CSF. The inside bioshield area radiation monitor provides an indication of failed fuel that would be used for subsequent long-term recovery actions, and it does not prompt GTG-related actions. Finally, the applicant stated that the GTGs will be validated during the performance of the integrated system validation (ISV), which may result in changes to DCA Part 2, Tier 2, Chapter 7, and the GTGs. The applicant's response to RAI 9430 explained the reason for differences between the Type B PAM variables listed in DCA Part 2, Tier 2, Chapter 7, and the variables used in the GTG flowcharts. However, the GTGs must be consistent with the design; therefore, the staff is tracking the resolution of these inconsistencies and the validation of the GTGs during ISV as **Open Item 13.5-1.** RAI 9430, Question 13.05.02.01-9 is marked as awaiting supplemental response for the staff's verification of these activities and review of any changes to the GTGs and the Type B PAM variables.

The staff finds the omission of a setpoint for containment $H_2$ concentration in Section 5.1.1, "Containment Atmosphere," of the GTGs is acceptable because the applicant explained the basis for including $O_2$ and excluding $H_2$ in the setpoint basis discussion and a COL applicant would address the specific setpoints for $H_2$ as part of the P-STGs, which the NRC staff will review.

With the exception of the **Open Item 13.5-1**, the staff finds the Containment Integrity Safety Function flowchart and operator actions adequate because they provide logical steps necessary for operators to maintain this safety function as it is described in the GTGs. As a result, the staff finds that this information is acceptable for the purpose of providing a basis for development of COL P-STGs.

*Reactivity Safety Function*

Section 5.2, "Reactivity Safety Function," of the GTGs describes the purpose of the reactivity safety function and the actions necessary to address the reactivity safety function. Specifically, Section 5.2 contains (1) the reactivity safety function flowchart, which depicts the functional logic and specifies the operator actions necessary to assess and maintain the CSF and (2) the accompanying bases information. The goal of the reactivity safety function is to ensure that the reactor has been tripped with all control rods inserted when required and that the reactor is maintained in a shutdown condition. Actions are prioritized by what can be performed in the MCR to those that are performed locally. The mitigating strategy is to take actions to trip the reactor if it has not tripped automatically when required by plant conditions. After actions to trip the reactor are performed, the remaining actions are to verify that the control rods are inserted and that no uncontrolled addition of positive reactivity is occurring and to borate, as needed, using the chemical and volume control system (CVCS) to maintain adequate shutdown margin. The reactivity safety flowchart in Section 5.2 of the GTGs depicts the logic and specifies the operator actions necessary to assess and maintain the reactivity safety function.

Because there is a possibility for DBEs to result in a CIS, the staff reviewed how the reactivity safety function is met during a scenario where an incomplete reactor trip occurs coincident with a CIS (e.g., resulting from a low-low pressurizer level) and operators are precluded from bypassing the CIS to borate/makeup unless a degraded core cooling condition exists. In this type of scenario, the staff questioned how the reactivity safety function can be met following a reactor trip with the power module not in a safe shutdown state in the event of a control rod malfunction coincident with a CIS. The applicant provided additional information explaining how the plant is designed to respond during an anticipated transient without scram (ATWS) event:

> *At full power, the core exhibits a large negative temperature coefficient, even at beginning of cycle conditions. The initial loss of feedwater, due to the containment isolation, causes a loss of RCS cooling which results in a rise in*

*RCS temperature. Due to the negative temperature coefficient, the core is subcritical shortly after the loss of feedwater, even without inserting control rods. The long-term ATWS response is unique because of the excess heat transfer capacity of the passive cooling systems. This excess heat transfer results from the relatively small core size, a large coolant mass-to-power ratio, and the efficient passive heat transfer systems. Return to power occurs only after passive heat transfer to the ultimate heat sink (UHS) has been established and RCS temperature is significantly reduced. The strong negative temperature coefficient and reduction in RCS temperature causes core fission power to increase until it is in equilibrium with the passive heat removal capacity. The core fission power never exceeds the passive heat removal capacity for an extended duration. The resulting fission power is well within the capacity of the passive cooling systems and UHS, and the core is protected.*

Based on this explanation, the staff finds that the NuScale design ensures that the reactivity safety function is met in this scenario.

Operators verify the isolation of dilution sources as part of the reactivity safety function. The staff compared the reactor trip parameters in DCA Part 2, Tier 2, Table 7.1-3, "Reactor Trip Functions," with the Demineralized Water System Isolation (DWSI) parameters in DCA Part 2, Tier 2, Table 7.1-4, "Engineered Safety Features Actuation System Functions," and determined that all reactor trip parameters also result in DWSI. In addition to the reactor trip parameters, a DWSI is also initiated by high subcritical multiplication and low RCS flow. Based on the discussion in this paragraph, the staff determined that Section 5.2 of the GTGs provides verification of the automatic actions identified in the safety analysis to address both CVCS isolation and DWSI and provides contingency actions to address potential failures of these safety systems to actuate.

DCA Part 2, Tier 2, Section 7.1.1.2.2, states that the core inlet and exit temperatures are Type B variables that provide a direct indication and are used to assess the process of accomplishing or maintaining the reactivity CSF. Further, DCA Part 2, Tier 2, Chapter 7, Table 7.1-7, identifies the core inlet and exit temperatures as Type B variables. Core inlet and exit temperatures are not evaluated in the reactivity safety function flowchart. Because DCA Part 2, Tier 2, Chapter 7, identifies the core inlet and exit temperatures as Type B variables that provide direct indication and are used to assess the process of accomplishing or maintaining the reactivity CSF, the staff questioned why these variables are not used as decision variables in the reactivity safety function flowchart to determine whether the reactivity CSF can be met. Therefore, the NRC issued RAI 9430, Question 13.05.02.01-7 (ADAMS Accession No. ML18101A640), to address this concern. In its response to RAI 9430, Question 13.05.02.01-7 (ADAMS Accession No. ML18162A359), the applicant stated that it selected the core inlet and exit temperatures as PAM variables before the creation of GTGs, which form the basis for the EOPs, specifically in accordance with RG 1.97 by compiling a conservative list of variables that it would then revise as procedures were developed. The applicant stated that the GTGs will be validated during ISV testing and that changes to DCA Part 2, Tier 2, Chapter 7, and the GTGs may result. The applicant's response to RAI 9430 explained the reason for differences between the Type B PAM variables listed in DCA Part 2, Tier 2, Chapter 7, and the variables used in the GTG flowcharts. However, the GTGs must be consistent with the design; therefore, the staff is tracking the resolution of these inconsistencies and the validation of the GTGs during ISV as **Open Item 13.5-1**. RAI 9430, Question 13.05.02.01-7 is marked as awaiting supplemental response for the staff's verification of these activities.

With the exception of the **Open Item 13.5-1**, the staff finds the Reactivity Safety Function flowchart and operator actions adequate because they provide the logical steps necessary for operators to maintain this safety function as it is described in the GTGs. As a result, the staff finds that this information is acceptable for the purpose of providing a basis for development of COL P-STGs.

*Core Heat Removal Safety Function*

Section 5.3, "Core Heat Removal Safety Function," of the GTGs, describes the purpose of the CHR safety function and the actions necessary to address the CHR safety function. Specifically, Section 5.3 contains (1) the CHR safety function flowchart, which depicts the functional logic and specifies the operator actions necessary to assess and maintain the CSF and (2) the accompanying bases information.

The staff reviewed the actions (grouped by subfunctions) of the CHR safety function. The staff compared the UHS inventory verification subfunction with the analysis presented in TR-0816-50797-NP and found the evaluation parameter and contingency actions to be consistent with the analyses that support mitigating strategies. The staff also compared the fuel clad protection verification subfunction with the accident sequence analyses presented in DCA Part 2, Tier 2, Chapter 19, and found the contingency actions and sequencing of those actions to be consistent with the accident sequence analyses presented in DCA Part 2, Tier 2, Chapter 19.

Section 5.3 of the GTGs includes actions to verify automatic system actuation for the ECCS, LTOP, DHRS, and the pressurizer heaters. The staff compared the CHR safety function flowchart to the ECCS actuation parameters in DCA Part 2, Table 7.1-4 and determined that the flowchart provides for verification of all the ECCS automatic actuation parameters. The staff compared the CHR safety function flowchart to the DHRS actuation parameters in DCA Part 2, Table 7.1-4 and determined that the flowchart provides for verification of all DHRS automatic actuation parameters except for the low-voltage ac power to the highly reliable direct current power system battery chargers for which there is a 60-second actuation delay for the ac voltage monitored at the buses that supply power to these chargers. The staff determined that exclusion of this parameter is acceptable because (1) it is an indication of an electrical power interruption that is not the direct result of, or caused by, a degraded core decay heat removal condition and (2) the power interruption is not a primary indication of the actual physical plant parameter response that would be indicative of the potential for, or the existence of, an actual degraded core decay heat removal condition. The staff compared the CHR safety function flowchart to the pressurizer heater trip parameters in DCA Part 2, Table 7.1-4 and found that the flowchart provides for verification of all heater trip parameters when both trains of the DHRS are aligned and directs operators to manually secure the pressurizer heaters otherwise.

In addition to verifying the automatic actuation of the ECCS/LTOP, DHRS, and pressurizer heater trip, the CHR safety function flowchart provides for contingency operator actions if the automatic actuations are not successful. Based on the comparison of the core heat CHR safety function flowchart with the parameters provided in DCA Part 2, Table 7.1-4 and on the operator actions provided in Section 5.3 of the GTGs, the staff has determined that Section 5.3 of the GTGs provides for verification of the automatic actions identified in the safety analysis and provides contingency actions to address potential failures of safety systems to perform their functions.

DCA Part 2, Tier 2, Section 7.1.1.2.2, states that degrees of subcooling is a Type B variable that provides direct indication and verification and is used to assess the process of accomplishing or maintaining the CHR CSF. Further, DCA Part 2, Tier 2, Table 7.1-7, also identifies degrees of subcooling as a Type B variable. The CHR safety function flowchart does not evaluate degrees of subcooling. Because DCA Part 2, Tier 2, Chapter 7, identifies degrees of subcooling as a Type B variable that provides direct indication and verification and is used to assess the process of accomplishing or maintaining the CHR CSF, the staff questioned why the CHR safety function flowchart does not use this variable as a decision variable to determine whether the CHR CSF can be met. Therefore, the NRC issued RAI 9430, Question 13.05.02.01-8 (ADAMS Accession No. ML18101A640), to address this concern. In its response to RAI 9430, Question 13.05.02.01-8 (ADAMS Accession No. ML18162A358), the applicant explained that RCS subcooling was selected as PAM variables prior to the creation of GTGs and that the PAM variables were selected in accordance with RG 1.97 by compiling a conservative list of variables that would then be revised as procedures were developed. The applicant stated that RCS subcooling can be used to determine whether adequate CHR is occurring but that it was not required once emergency procedure guidance was developed based on the NuScale design. The applicant states that the GTGs will be validated during ISV testing and that changes to DCA Part 2, Tier 2, Chapter 7, and the GTGs may result. The applicant's response to RAI 9430 explained the reason for differences between the Type B PAM variables listed in DCA Part 2, Tier 2, Chapter 7, and the variables used in the GTG flowcharts. However, the GTGs must be consistent with the design; therefore, the staff is tracking the resolution of these inconsistencies and the validation of the GTGs during ISV as **Open Item 13.5-1**. RAI 9430, Question 13.05.02.01-8, is marked as awaiting supplemental response for the staff's verification of these activities.

With the exception of the **Open Item 13.5-1**, the staff finds the Core Heat Removal Safety Function flowchart and operator actions adequate because they provide logical steps necessary for operators to maintain this safety function as it is described in the GTGs. As a result, the staff finds that this information is acceptable for the purpose of providing a basis for development of COL P-STGs.

*Miscellaneous Items*

During a public/closed meeting conducted on February 2, 2018, and closed meetings held on February 9, 2018, and February 15, 2018, the staff discussed its initial review of TR-1117-57216, Revision 0, with the applicant. A comprehensive meeting summary dated June 21, 2018 (ADAMS Accession No. ML18169A173), provides further information on these meetings. Based on these discussions, the staff issued several questions in RAI 9430 (ADAMS Accession No. ML18101A640) for the following disparities/inconsistencies that it identified between DCA Part 2, Tier 2, Chapter 7, and the NuScale GTG TR-1117-57216, Revision 0, with regard to the PAM variable information specified for the indications and instrumentation:

- containment isolation valve position indication (RAI 9430, Question 13.05.02.01-10).
- RCS $T_{HOT}$ instrumentation (RAI 9430, Question 13.05.02.01-11).
- containment pressure instrumentation (RAI 9430, Question 13.05.02.01-12).
- pressurizer level instrumentation (RAI 9430, Question 13.05.02.01-13).

- neutron flux instrumentation (RAI 9430, Question 13.05.02.01-14).

- low-voltage ac ELVS, 24-hour internal monitoring and protection system timer (RAI 9430, Question 13.05.02.01-15).

In its response to RAI 9430, Questions 13.05.02.01-10 through 13.05.02.01-15 (ADAMS Accession No. ML18162A359), the applicant addressed the inconsistencies and provided markups to the GTGs to resolve these inconsistencies. The staff reviewed the applicant's responses and the markups and finds them acceptable. **RAI 9430, Question 13.05.02.01-12 is a confirmatory item** pending verification that the next revision of the DCA incorporates these markup changes in DCA Part 2, Tier 2, Section 7.1.1.2.2, DCA Part 2, Tier 2, Table 7.1-7, Table 3.11-1, and Tier 1, Table 2.5-5.

### 13.5.5 Combined License Information Items

Table 13.5-5 lists COL information item numbers and descriptions related to plant procedures from DCA Part 2, Tier 2, Revision 1, Table 1.8-2. The COL information items presented in Table 13.5-5 are reflected in Revision 1 of the DCA.

Note that the applicant deleted COL Information Item 13.5-6, which separately addresses the application and usage of site-specific EOPs, from the original application. In its revised response to RAI 8827, Question 13.05.02.01-3 (ADAMS Accession No. ML17282A017), the applicant explained that it did not include the COL information item after it had determined that the item was unnecessary based on discussions held during the July 25, 2017 public teleconference (ADAMS Accession No. ML17215B148). Accordingly, COL 13.5-6 will not be used as indicated in Table 13.5-5.

**Table 13.5-5 NuScale COL Information Items for Section 13.5**

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| COL Item 13.5-1 | A COL applicant that references the NuScale Power Plant design certification will describe the site-specific procedures that provide administrative control for activities that are important for the safe operation of the facility consistent with the guidance provided in RG 1.33, Revision 3. | 13.5.1 |
| COL Item 13.5-2 | A COL applicant that references the NuScale Power Plant design certification will describe the site-specific procedures that operators use in the main control room and locally in the plant, including normal operating procedures, abnormal operating procedures, and emergency operating procedures (EOPs). The COL applicant will describe the classification system for these procedures, and the general format and content of the different classifications. | 13.5.2.1 |
| COL Item 13.5-3 | A COL applicant that references the NuScale Power Plant design certification will describe the site-specific maintenance and other operating procedures, including how these procedures are classified, and the general format and content of the different classifications. The categories of procedures listed below should be included: | 13.5.2.2 |

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| | • plant radiation protection procedures;<br>• emergency preparedness procedures;<br>• calibration and test procedures;<br>• chemical-radiochemical control procedures;<br>• radioactive waste management procedures;<br>• maintenance and modification procedures;<br>• material control procedures; and<br>• plant security procedures. | |
| COL Item 13.5-4 | A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of administrative procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures. | 13.5.1 |
| COL Item 13.5-5 | A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures. | 13.5.2.1 |
| COL Item 13.5-6 | Not used. | N/A |
| COL Item 13.5-7 | A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of EOPs, including preliminary schedules for preparation and target dates for completion. Included in the submittal is the Procedures Generation Package, consisting of the following:<br>• Plant Specific Technical Guidelines, which are guidelines based on analysis of transients and accidents that are specific to the COL applicant's plant design and operating philosophy.<br>• A plant-specific writer's guide that details the specific methods to be used by the COL applicant in preparing EOPs based on the Plant Specific Technical Guidelines.<br>• A description of the program for verification and validation of the EOPs.<br>• A description of the program for training operators on the EOPs.<br>Additionally, the COL applicant will identify the group within the operating organization responsible for maintaining these procedures. | 13.5.2.1 |

| Item No. | Description | DCA Part 2, Tier 2 Section |
|---|---|---|
| COL Item 13.5-8 | A COL applicant that references the NuScale Power Plant design certification will provide a plan for the development, implementation, and control of maintenance and other operating procedures, including preliminary schedules for preparation and target dates for completion. Additionally, the COL applicant will identify what group or groups within the operating organization have the responsibility for maintaining and following these procedures | 13.5.2.2 |

### 13.5.6 Conclusion

The COL applicant is responsible for the development of plant procedures. In its review of DCA Part 2, Tier 2, Revision 1, Section 13.5, the staff (1) evaluated the acceptability of seven COL information items and (2) evaluated the technical adequacy of the NuScale GTGs and their acceptability for use in the development of the COL applicant's P-STGs. The staff determined that the seven COL information items are acceptable. No open items are associated with COL information items in DCA Part 2, Tier 2, Revision 1, Section 13.5.

Therefore, the staff concludes that the COL information items specified in Table 13.5-5 are sufficient to identify information that the COL applicant needs to provide to satisfy the applicable requirements of 10 CFR 50.34; 10 CFR 50.40; 10 CFR Part 50, Appendix B; and 10 CFR 52.79.

To this point, the staff's review of the NuScale GTGs focused on (1) the three CSFs defined for the NuScale power plant, (2) the methodology used to identify operator actions, and (3) the CSF flowchart logic and operator actions necessary to assess and maintain the CSFs, including the bases. The applicant has submitted satisfactory responses to the staff's RAIs, but the staff is presently unable to conclude that the NuScale GTGs are acceptable for use as a basis for the development of COL applicant P-STGs. A determination of acceptability is contingent upon the achievement of satisfactory results from ISV testing and validation activities and the subsequent incorporation of any necessary changes to the GTGs and the associated PAM variables. These ISV activities will occur during the review of the DCA and are being tracked as **Open Item 13.5-1**.