14 INITIAL TEST PROGRAM AND INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA

THIS NRC STAFF DRAFT SE HAS BEEN PREPARED AND IS BEING RELEASED TO SUPPORT INTERACTIONS WITH THE ACRS. THIS DRAFT SE HAS NOT BEEN SUBJECT TO FULL NRC MANAGEMENT AND LEGAL REVIEWS AND APPROVALS, AND ITS CONTENTS SHOULD NOT BE INTERPRETED AS OFFICIAL AGENCY POSITION.

The ITP and ITAAC may change to address information needed to make a safety finding in high effort chapters. Any changes that impact this safety evaluation will be reflected as part of the high effort chapter's safety evaluation and revised in this safety evaluation, as necessary.

14.1 Introduction

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereafter referred to as the staff) review of Chapter 14, "Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria," of the NuScale Power, LLC (hereafter referred to as the applicant), Standard Design Approval Application (SDAA), Part 2, "Final Safety Analysis Report." The NRC staff's regulatory findings documented in this report are based on Revision 1 of the SDAA, dated October 31, 2023 (Agencywide Documents Access and Management System Accession No. ML23304A364).

The precise parameter values, as reviewed by the staff in this safety evaluation (SE), are provided by the applicant in the SDAA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this SE to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

In this chapter, the NRC staff uses the term "non-safety-related" to refer to structures, systems and components (SSCs) that are not classified as "safety-related SSCs," as described in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.2, "Definitions." However, among the "non-safety-related" SSCs, there are those that are "important to safety," as that term is used in the general design criteria (GDC) listed in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and others that are not considered "important to safety."

14.2 Initial Test Program

14.2.1 Generic Guidelines for Initial Test Programs

14.2.1.1 Introduction

The applicant for an operating license under 10 CFR Part 50 or a combined license (COL) under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," is responsible for ensuring that a suitable initial (preoperational and startup) test program will be conducted for the facility. The initial test program (ITP) includes system and component tests, monitoring of SSC performance, and inspection and surveillance test activities for plant SSCs. An ITP satisfying these objectives should provide the necessary assurance that the facility can be operated in accordance with design requirements and in a manner that will not endanger public health and safety.

Initial startup testing consists of equipment performance tests completed during and after fuel loading. These performance tests are normally conducted during the fuel loading, precritical, initial criticality, low-power, and power ascension phases to confirm the design-basis and demonstrate, to the extent practical, that the plant will operate in accordance with the design and can respond to anticipated transients and postulated accidents as specified in the SDAA.

The ITP is designed to demonstrate the performance of SSCs and integrated plant design features that will be used during normal facility operations, as well as standby systems and features that must function to maintain the plant in a safe condition in the event of malfunctions or accidents. The startup tests are sequenced so that plant safety is never entirely dependent on the performance of untested SSCs.

Regulatory Guide (RG) 1.68, Revision 4, “Initial Test Programs for Nuclear Power Plants,” issued June 2013, describes the general scope and depth of the ITP acceptable to the NRC staff for light-water-cooled nuclear power plants. Additionally, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 14.2, Revision 3, “Initial Plant Test Program—Design Certification and New License Applicants,” issued March 2007, provides guidance to the NRC staff for the review of a proposed ITP. For small modular reactor designs, SECY-11-0024, “Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews,” dated February 18, 2011, requested Commission approval of the staff’s recommendation to develop a risk-informed and integrated framework for the review of the integral pressurized-water reactor designs. On May 11, 2011, the Commission approved the staff’s approach and provided additional direction (ML111320551). In response, the NRC staff subsequently developed a design-specific review standard (DSRS) for the NuScale design. NuScale DSRS Section 14.2, “Initial Plant Test Program—Design Certification and New License Applicants,” dated July 11, 2016, provides guidance to the NRC staff in reviewing the proposed NuScale ITP.

DSRS Section 14.2 notes that a design certification (DC) applicant is not required to provide an ITP submittal under 10 CFR Part 52, Subpart B, “Standard Design Certifications.” Likewise, a standard design approval (SDA) applicant is not required to submit an ITP under 10 CFR Part 52, Subpart E, “Standard Design Approvals.” For this design, however, the applicant elected to request NRC review of its program; therefore, the staff reviewed the test abstracts for completeness and suitability for the development of an ITP against the guidance in SRP Section 14.2 and RG 1.68.

14.2.1.2 Summary of Application

SDAA Part 2: The applicant discussed its program in SDAA Part 2, Section 14.2.1, “Summary of Initial Test Program and Objectives,” which is summarized here in part:

> The Initial Test Program (ITP) consists of a series of preoperational and startup tests conducted by the Startup organization. Preoperational testing is conducted for each NuScale Power Module (NPM) following completion of construction testing but before fuel load. Completion of preoperational testing for each NPM is necessary to ensure the NPM is ready for fuel loading and startup testing.

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

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

Technical Reports: There are no technical reports for this area of review.

14.2.1.3 Regulatory Basis

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

- 10 CFR 50.34(b)(6)(iii), which requires the applicant to provide plans for preoperational testing and initial operations

- 10 CFR 30.53(c), as it relates to testing radiation detection and monitoring instruments

- Criterion XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, as it relates to test programs established to ensure that SSCs will perform satisfactorily in service

- Section III.A.4 of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50, as it relates to the preoperational leakage testing of the primary reactor containment and related systems and components penetrating the primary containment pressure boundary

- 10 CFR 50.43(e)(1)(i), which states that an application for a DC or an SDA that proposes nuclear reactor designs that differ significantly from light-water reactor (LWR) designs that were licensed before 1997, or use simplified, inherent, passive, or other innovative means to accomplish their safety functions, will only be approved if the performance of each safety feature of the design has been demonstrated through analysis, appropriate test programs, experience, or a combination thereof

- 10 CFR 52.137(b), which requires that an application for approval of a standard design for a reactor that differs significantly from the LWR designs of plants that have been licensed and in commercial operation before April 18, 1989, or that use simplified, inherent, passive, or other innovative means to accomplish its safety functions, must meet the requirements of 10 CFR 50.43(e)

- 10 CFR 52.79(a)(28), which requires COL applicants to provide plans for preoperational testing and initial operations

Additionally, the guidance in DSRS Section 14.2 lists acceptance criteria adequate to meet the above requirements, as well as review interfaces with other DSRS sections.

14.2.1.4 Technical Evaluation

The applicant provided the technical information associated with the ITP in SDAA Part 2, Section 14.2, "Initial Plant Test Program." This information applies to the preoperational testing phase, as well as the initial startup testing phase. Preoperational testing consists of tests conducted following completion of construction and construction-related inspections and tests but before fuel loading. Preoperational testing demonstrates the capability of the plant systems to meet relevant performance requirements. Startup tests, which begin with initial fuel loading, demonstrate the capability of the integrated plant to meet performance requirements. The staff reviewed the NuScale ITP in accordance with the guidance in RG 1.68 and DSRS Section 14.2.

For each phase of the ITP, a license applicant may define organizational responsibilities, describe administrative controls for the development of the test program, and provide test abstracts, which include the objectives of each test, as well as a summary of prerequisites, test methods, and specific acceptance criteria. These test abstracts should address the criteria outlined in RG 1.68 and, specific to the NuScale application, DSRS Section 14.2. The DSRS also states that the applicant should describe how it considered the use of reactor operating and testing experience, the trial use of plant operating and emergency procedures, and conformance with applicable RGs. Conformance of a proposed test program to the above guidelines provides reasonable assurance that the facility can be operated in accordance with its design criteria and in a manner that will not endanger public health and safety.

The staff noted that the applicant provided administrative test attributes, consistent with the DSRS, in the areas of organization and staffing, conformance with RGs, test procedure control, reactor operating and testing experience, plant operating and emergency procedures, and test program scheduling and sequencing. In addition, the applicant provided individual test descriptions, test performance requirements, and acceptance criteria for each preoperational and startup test.

14.2.1.4.1 *Initial Test Program Objectives*

The staff reviewed the preoperational and initial startup testing objectives as described in SDAA Part 2, Section 14.2, against the guidance in RG 1.68 and DSRS Section 14.2. Consistent with this guidance, the staff noted that the applicant's proposed test program includes controls to (1) provide assurance that SSCs operate in accordance with their design, (2) provide assurance that construction and installation of equipment in the facility have been completed in accordance with the design, (3) demonstrate, to the extent practical, the validity of analytical models used to predict plant responses to anticipated transients and postulated accidents, as well as the correctness and conservatism of assumptions used in those models, (4) familiarize the plant's operating and technical staff with the operation of the facility, (5) perform testing, to the extent practical, using the plant conditions that simulate the actual operating, abnormal operating occurrences, and emergency conditions to which the SSCs may be subjected, (6) verify, to the extent practical, by trial use, that the facility's operating, surveillance, and emergency procedures are adequate, (7) verify that system interfaces and component interactions are in accordance with the design, and (8) complete and document the ITP testing required to satisfy preoperational and startup testing requirements, thus providing reasonable assurance that the plant can be brought safely to its rated power and can be safely operated during sustained power operations.

Consistent with guidance, in the preoperational and startup testing phase description, the staff noted that the applicant's testing is performed on those SSCs that are (1) relied upon for safe shutdown and cooldown of the NPM under normal conditions for maintaining a safe condition for an extended shutdown period, (2) relied upon for safe shutdown and cooldown of the NPM under transient and postulated accident conditions and for maintaining a safe condition for an extended shutdown period following such conditions, (3) relied upon for establishing conformance with safety limits or limiting conditions for operation that are included in the TS,

(4) assumed to function or for which credit is taken in the accident analysis as described in SDAA Part 2, Chapter 15, “Transient and Accident Analyses,” (5) used to process, store, control, or limit the release of radioactive materials, (6) relied upon to maintain their structural integrity during normal operation, anticipated transients, simulated test parameters, and design-basis event conditions to avoid damage to safety-related SSCs, and (7) identified as risk significant in the probabilistic risk assessment.

Based on the discussion above, in the initial startup testing phase description and test abstracts, the staff noted that the applicant provided controls consistent with guidance to ensure (1) a safe core loading, (2) a safe and orderly approach to initial criticality, and (3) the plant’s ability to meet test acceptance criteria during low-power and power ascension testing based on sufficient testing.

14.2.1.4.2 *Organizational Staffing Responsibilities*

DSRS Section 14.2 states that the applicant is responsible for providing a detailed description of management organizations and staff responsibilities, authorities, and qualifications. As such, in SDAA Part 2, Section 14.2.2, “Organization and Staffing,” the applicant provided COL Item 14.2-1, which states, “An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific organizations that manage, supervise, or execute the ITP, including the associated training requirements.” The staff finds this consistent with the guidance in DSRS Section 14.2, as the COL item indicates that the license applicant will implement adequate organization and staffing when testing is conducted.

14.2.1.4.3 *Initial Test Program Test Procedures*

The staff reviewed the methodology submitted by the applicant that will be used to develop, review, and approve individual test procedures to ensure that they are consistent with relevant guidance in RG 1.68 and DSRS Section 14.2 or propose to meet the regulatory requirements in a different way. Section 14.2 of the DSRS specifies that the applicant should provide a summary description of the general guidance to control ITP activities. This description should include administrative controls that will be used to develop, review, and approve individual test procedures; coordinate with organizations involved in the test program; confirm participation of plant operating and technical staff; and review, evaluate, and approve test results.

In SDAA Part 2, Section 14.2.3.1, “Initial Test Program Procedures,” the staff noted that the applicant provided general guidance for the development and review of test specifications and procedures. Specifically, the SDAA states that the preoperational and startup testing procedures will contain the following administrative controls: (1) test procedure format, (2) application, to the extent practical, of normal plant operating procedures, emergency operating procedures, and surveillance procedures in support of test procedure development, (3) test procedure review and approval, and (4) test procedure change and revision. Further, the SDAA states that the content of the procedures will address objectives, detailed step-by-step instructions specifying how testing is to be performed, special precautions, test instrumentation, test equipment calibration, initial test conditions, methods to direct and control test performance, acceptance criteria by which testing is evaluated, test prerequisites, identification of the data to be collected and method of documentation, actions to take if unanticipated errors or malfunctions occur while testing, remedial actions to take if acceptance criteria are not satisfied, and actions to take if an unexpected or unanalyzed condition occurs. Additionally, SDAA Part 2, Section 14.2.3.4, “Generic Component Testing,” discusses procedures to be developed for generic component testing, which is generally executed after a system's transfer from the construction organization to the startup organization.

SDAA Part 2, Section 14.2.3.2, "Graded Approach to Testing," outlines the graded approach to testing, consistent with the requirements of GDC 1, "Quality Standards and Records." It requires, in part, that SSCs important to safety be tested to quality standards commensurate with the importance of the safety functions to be performed. The NuScale subject matter experts identified all functions of each system during the SSC classification process and compared them to safety functional requirements as described in SDAA Part 2, Section 17.4, "Reliability Assurance Program." As noted in the test abstracts in SDAA Part 2, Section 14.2.12, "Individual Test Descriptions," the testable functions contain a safety and risk categorization.

RG 1.68 and DSRS Section 14.2 describe certain tests that should be included in the ITP, such as first-of-a-kind tests, which are new, unique, or special tests used to verify design features that the NRC has not previously reviewed. As such, SDAA Part 2, Section 14.2.3.3, "Testing of First-of-a-Kind Design Features," highlights the four tests and refers to Table 14.2-104, "ITP Testing of New Design Features," which summarizes the ITP testing for new design features.

The staff finds that the general test specifications and test procedure guidelines specified in SDAA Part 2, Section 14.2.3, "Test Procedures," are acceptable for the SDA because the specifications and guidelines are consistent with RG 1.68 and DSRS Section 14.2. Because plant-specific design information will be needed, the staff concludes that it is acceptable to defer responsibility for the development of detailed preoperational and startup test specifications and test procedures to the license applicant.

14.2.1.4.4 *Initial Test Program's Conformance with Regulatory Guides*

The staff reviewed the methodology used by the applicant to verify that the ITP is consistent with the guidance in the RGs. DSRS Section 14.2 states, in part, that the applicant should establish and describe an ITP that is consistent with the regulatory positions outlined in RG 1.68 and identifies supplemental RGs that provide more detailed information pertaining to the testing. Appendix A to RG 1.68 references a set of supplemental RGs that provide additional guidance for particular tests during the preoperational and initial startup phases. The supplemental RGs contain additional information to help determine whether performance of the tests in the proposed manner will accomplish the objectives of certain plant tests.

In SDAA Part 2, Section 14.2.7, "Test Programs Conformance with Regulatory Guides," the applicant listed the RGs used in the development of the NuScale ITP. In addition, SDAA Part 2, Table 1.9-2, "Conformance with Regulatory Guides," lists the RGs applicable to the NuScale design. The staff reviewed this table to ensure that the applicable RGs were included in the development of the ITP. In cases where the applicant determined that RGs did not apply to the NuScale design, or where the applicant proposed a deviation from the guidance in the RGs, the staff review found that the applicant's proposed testing scope was acceptable to meet the applicable regulatory guidance.

The staff reviewed the list of RGs that the applicant had determined are not applicable to the NuScale design, which include the following:

- RG 1.9, Revision 4, "Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants," issued March 2007

- RG 1.52, Revision 4, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” issued September 2012

- RG 1.79.1, “Initial Test Program of Emergency Core Cooling Systems for New Boiling-Water Reactors,” issued October 2013

- RG 1.160, Revision 3, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” issued May 2012

The staff determined that RGs 1.9 and 1.52 do not apply to the NuScale SDAA because the design does not require or include safety-related emergency diesel generators or containment atmosphere controls, respectively. RG 1.79.1 does not apply to the NuScale design as it is specific to boiling-water reactors, while RG 1.160 does not apply as it contains guidance for meeting requirements that are the responsibility of the license applicant. Thus, the staff concludes that those RGs do not apply to the NuScale SDAA.

Based on the above review, the staff finds that the NuScale ITP adequately conforms to the general scope and depth of test programs, as described in RG 1.68, and also conforms to the test program regulatory positions stated in DSRS Section 14.2. In addition, the staff finds that the applicant has adequately justified the categorization of certain RGs as inapplicable to the NuScale SDAA review.

14.2.1.4.5 *Use of Reactor Operating and Testing Experience in the Development of the Initial Test Program*

The staff reviewed the methodology submitted by the applicant to include reactor operating and testing experience in the development of the ITP. DSRS Section 14.2 and RG 1.68 state that the applicant should describe how it used the operating and testing experiences of other facilities in the development of the ITP.

In SDAA Part 2, Section 14.2.8, “Utilization of Reactor Operating and Testing Experience in Test Program Development,” the applicant considered the use of operational and testing experience gained from previous pressurized-water reactor plant designs,[1] as well as operating and testing experience obtained from NRC licensee event reports, NRC generic communications, and Institute of Nuclear Power Operations issuances. The applicant stated that the administrative procedures control the review of reactor operating experience and its incorporation in the ITP. In SDAA Part 2, Section 14.2.4, “Conduct of the Test Program,” the applicant stated that it will be responsible for providing test specifications and test procedures for preoperational and startup tests for NRC review and for the preparation of the startup administration manual, which will contain the processes and standards that govern the activities associated with the plant ITP. COL Item 14.2-2 directs that an applicant that references the NuScale Power Plant US460 standard design is responsible for the development of the startup administration manual, which will contain the administrative procedures and requirements that control the activities associated with the ITP.

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[1] COL Item 14.2-3 states, in part, “an applicant that references the NuScale Power Plant US460 standard design will identify the specific operator training to be conducted during low-power testing related to the resolution of Three Mile Island Action Plan Item I.G.1.”

The staff finds that the applicant provided adequate controls for the use of reactor operating and testing experience as described in RG 1.68 and DSRS Section 14.2. However, development of ITP test procedures will require the applicant to review detailed, plant-specific design information; thus, the staff concludes that it is acceptable to defer the review of the use of operating and testing experience to the applicant.

14.2.1.4.6 *Trial Use of Plant Operating Procedures, Emergency Procedures, and Surveillance Procedures*

The staff reviewed the proposed trial use of plant operating, emergency, and surveillance procedures during the performance of the ITP. DSRS Section 14.2 states that the applicant should incorporate plant operating, emergency, and surveillance procedures into the test program, or otherwise verify these procedures through use, to the extent practicable, during the ITP.

In SDAA Part 2, Section 14.2.9, "Trial Use of Plant Operating Procedures, Emergency Procedures, and Surveillance Procedures," the staff noted that the applicant included provisions to ensure that the plant's normal, surveillance, abnormal, and emergency operating procedures will be, to the extent practical, developed, trial tested, and corrected throughout the preoperational and initial startup tests.

A license applicant that references the NuScale Power Plant US460 SDA is responsible for the development of the Startup Administration Manual, which will contain the administrative procedures and requirements that control the activities associated with the ITP. The applicant should provide a milestone for completing the Startup Administrative Manual and making it available for NRC inspection (COL Item 14.2-2).

The staff also notes that the applicant's quality assurance controls should ensure that procedures are appropriate and include quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Based on the above review, the staff finds that NuScale's approach is acceptable to develop, trial test, and correct the plant's normal, surveillance, abnormal, and emergency operating procedures throughout the preoperational and initial startup tests, to the extent practical, during preoperational and initial startup test activities.

14.2.1.4.7 *Initial Fuel Loading and Initial Criticality*

The staff reviewed the measures provided by the applicant for use during initial fuel loading and initial criticality. RG 1.68 and DSRS Section 14.2 provide general guidance on the conduct of the ITP after the completion of preoperational testing. As stated in the regulatory guidance, initial fuel loading and precritical tests ensure that (1) initial core loading is safe, (2) provisions are in place to maintain a shutdown margin, and (3) the facility is in a final state of readiness to achieve criticality and to perform low-power testing.

In SDAA Part 2, Section 14.2.10, "Initial Fuel Loading, and Initial Criticality," the applicant included provisions for prefuel load checks, initial fuel loading, precriticality, and initial criticality in accordance with RG 1.68 and DSRS Section 14.2. The staff noted that these provisions included TS compliance, proper verification of boron concentration limits, calibration and testing of nuclear instrumentation, shutdown margin verifications at predetermined intervals, and control rod functionality tests. These controls are consistent with the regulatory positions in RG 1.68 and are therefore acceptable to the staff.

Based on the above review, the staff concludes that the ITP adequately addresses the initial fuel loading and initial criticality testing by meeting the associated guidance in RG 1.68 and DSRS Section 14.2.

14.2.1.4.8 *Initial Test Program Schedule and Sequence*

The staff reviewed the methodology submitted by the applicant that will be used to develop the ITP schedule and sequence. RG 1.68 and DSRS Section 14.2 discuss the guidelines for the test program schedule and sequence, stating that the applicant should develop a schedule for conducting each major phase of the ITP and that the schedule should establish that the safety of the plant will not depend on the performance of untested SSCs.

The staff noted that, in SDAA Part 2, Section 14.2.11, "Test Program Schedule and Sequence," the applicant provided measures for conducting each major phase of the ITP relative to the initial fuel load. The SDAA states that the license applicant will provide a schedule showing the timetable for the generation, review, and approval of procedures, as well as the actual testing and analysis of the results. The applicant also stated that approved test procedures will be available to the staff no later than 60 days before their intended use.

The staff reviewed the controls that will be implemented during the preoperational and initial startup testing phases. The staff found that the applicant provided general controls to ensure that, during the preoperational testing phase, testing is performed as systems and equipment availability allows. Additionally, the staff noted that the applicant stated that test sequencing is accomplished as early in the test program as feasible and that the safety of the plant is not dependent on the performance of untested systems, components, or features.

Based on the above review, the staff finds that the information provided by the applicant is consistent with the guidance contained in RG 1.68 and DSRS Section 14.2. Since a license applicant is designated as responsible for the test program schedule, the staff finds that it is acceptable to defer the detailed test program schedule and sequence to the COL stage. The license applicant should provide a milestone for completing the detailed testing schedule and make it available to the NRC (COL Item 14.2-4).

14.2.1.4.9 *Individual Test Descriptions*

Tables 14.2-1 through 14.2-108 of SDAA Part 2, Section 14.2, provide the individual test abstracts. Each abstract identifies each test by title and gives the test objectives, prerequisites, test methods, and acceptance criteria. These test abstracts will be used in the development of detailed preoperational and startup test procedures. Based on the risk-informed approach specified in the NuScale DSRS and consistent with Commission direction, the staff performed a risk-informed review of the test abstracts and adapted the depth of the review based on the safety significance of the test. The test abstracts were binned into three categories:

(1) Test abstracts associated with SSCs identified to be safety-related, as having high risk or safety significance, or as being referenced by ITAAC were given a detailed review.

(2) Test abstracts associated with SSCs identified to be of low risk and low safety significance with no safety impact during initial plant startup were given a more limited review.

(3) Test abstracts that will be developed by the license applicant and are not reviewed at the SDA stage are indicated in Table 14.2-2 of this SER and will be reviewed at the COL stage.

In accordance with RG 1.68 and DSRS 14.2, the staff confirmed that the following test abstracts contained in Table 14.2-1 are adequate.

Table 14.2-1 NuScale Section 14.2 Test Abstracts Reviewed at the SDAA Stage

Abstract	Test Title
Table 14.2-1[2]	Pool Cooling and Cleanup System Test #1
Table 14.2-2[2]	Ultimate Heat Sink Test #2
Table 14.2-3[2]	Pool Leakage Detection System Test #3
Table 14.2-4[2]	Reactor Component Cooling Water System Test #4
Table 14.2-5	Chilled Water System Test #5
Table 14.2-6[2]	Auxiliary Boiler System Test #6
Table 14.2-7[2]	Air Cooled Condenser System Test #7
Table 14.2-8[2]	Site Cooling Water System Test #8
Table 14.2-10[2]	Utility Water System Test #10
Table 14.2-11[2]	Demineralized Water System Test #11
Table 14.2-12	Nitrogen Distribution System Test #12
Table 14.2-13[2]	Service Air System Test #13
Table 14.2-14[2]	Instrument Air System Test #14
Table 14.2-15[2]	Control Room Habitability System Test #15
Table 14.2-16[2]	Normal Control Room HVAC System #16
Table 14.2-17[2]	Reactor Building HVAC System Test #17
Table 14.2-18	Radioactive Waste Building HVAC System Test #18
Table 14.2-19	Turbine Building HVAC System Test #19
Table 14.2-20	Radioactive Waste Drain System Test #20
Table 14.2-21[2]	Balance-of-Plant Drain System Test #21
Table 14.2-22[2]	Fire Protection System Test #22
Table 14.2-23[2]	Fire Detection System Test #23
Table 14.2-24	Main Steam System Test #24
Table 14.2-25[2]	Condensate and Feedwater System Test #25
Table 14.2-26[2]	Feedwater Treatment System Test #26
Table 14.2-27[2]	Condensate Polisher Resin Regeneration System Test #27
Table 14.2-28[2]	Feedwater Heater Vents and Drains System Test #28
Table 14.2-29	Turbine Generator System Test #29
Table 14.2-30[2]	Liquid Radioactive Waste System Test #30

[2] The NRC staff has identified these test abstracts as having low risk and low safety significance. Accordingly, commensurate with the low risk and low safety significance of these abstracts and in accordance with the SRP, Introduction Part 2, the NRC staff limited the depth of the review for these abstracts.

Abstract	Test Title
Table 14.2-31[2]	Gaseous Radioactive Waste System Test #31
Table 14.2-32[2]	Solid Radioactive Waste System Test #32
Table 14.2-33	Chemical and Volume Control System Test #33
Table 14.2-34[2]	Boron Addition System Test #34
Table 14.2-35	Module Heatup System Test #35
Table 14.2-36	Containment Evacuation System Test #36
Table 14.2-37[2]	Containment Flooding and Drain Test #37
Table 14.2-38	Containment System Test #38
Table 14.2-39[2]	Reactor Coolant System Test #39
Table 14.2-40[2]	Emergency Core Cooling System Test #40
Table 14.2-41	Decay Heat Removal System Test #41
Table 14.2-42	In-Core Instrumentation System Test #42
Table 14.2-43	Module Assembly Equipment Test #43
Table 14.2-44	Fuel Handling Equipment Test # 44
Table 14.2-45	Reactor Building Cranes Test # 45
Table 14.2-46	Process Sampling System Test #46
Table 14.2-47	High Voltage AC Electrical Distribution System Test #47
Table 14.2-48	Medium Voltage AC Electrical Distribution System Test #48
Table 14.2-49[2]	Low Voltage AC Electrical Distribution System Test #49
Table 14.2-50	Augmented DC Power System Test #50
Table 14.2-51	Normal DC Power System Test #51
Table 14.2-52	Backup Power Supply System Test #52
Table 14.2-53[2]	Plant Lighting System Test #53
Table 14.2-54[2]	Module Control System Test #54
Table 14.2-55[2]	Plant Control System Test #55
Table 14.2-56[2]	Module Protection System Test #56
Table 14.2-57[2]	Plant Protection System Test #57
Table 14.2-58[2]	Neutron Monitoring System Test #58
Table 14.2-59[2]	Safety Display and Indication System Test #59
Table 14.2-60	Fixed-Area Radiation Monitoring System Test #60
Table 14.2-61[2]	Communication System Test #61
Table 14.2-63	Hot Functional Testing Test #63
Table 14.2-64[2]	Module Assembly Equipment Bolting Test #64
Table 14.2-65[2]	Steam Generator Flow-Induced Vibration Test #65
Table 14.2-66	Security Access Control Test #66
Table 14.2-67[2]	Security Detection and Alarm Test #67
Table 14.2-68	Initial Fuel Loading and Precritical Test #68
Table 14.2-69	Initial Fuel Load Test #69
Table 14.2-70	Reactor Coolant System Flow Measurement #70
Table 14.2-71[2]	NuScale Power Module Temperatures Test #71
Table 14.2-72	Primary and Secondary System Chemistry Test #72
Table 14.2-73	Control Rod Drive System - Manual Operation, Rod Speed, and Rod Position Indication Test #73
Table 14.2-74	Control Rod Assembly Full-Height Drop Time Test #74
Table 14.2-75[2]	Control Rod Assembly Ambient Temperature Full-Height Drop Time Test #75
Table 14.2-76	Pressurizer Spray Bypass Flow #76

Abstract	Test Title
Table 14.2-77	Initial Criticality Test #77
Table 14.2-78	Post-Critical Reactivity Computer Checkout Test #78
Table 14.2-79	Low-Power Test Sequence Test#79
Table 14.2-80	Determination of Zero-Power Physics Testing Range Test #80
Table 14.2-81	All Rods Out Boron Endpoint Determination Test #81
Table 14.2-82	Isothermal Temperature Coefficient Measurement Test #82
Table 14.2-83	Bank Worth Measurement Test #83
Table 14.2-84	Power-Ascension Test #84
Table 14.2-85[2]	Core Power Distribution Map Test #85
Table 14.2-86	Neutron Monitoring System Power Range Flux Calibration Test #86
Table 14.2-87	Reactor Coolant System Temperature Instrument Calibration Test #87
Table 14.2-88	Reactor Coolant System Flow Calibration Test #88
Table 14.2-89	Radiation Shield Survey Test #89
Table 14.2-90[2]	Reactor Building Ventilation System Capability Test #90
Table 14.2-91	Thermal Expansion Test #91
Table 14.2-92	Control Rod Assembly Misalignment Test #92
Table 14.2-93	Steam Generator Level Control Test #93
Table 14.2-94	Ramp Change in Load Demand Test #94
Table 14.2-95	Step Change in Load Demand Test #95
Table 14.2-96	Loss of Feedwater Heater Test #96
Table 14.2-97	100 Percent Load Rejection Test #97
Table 14.2-98	Reactor Trip from 100 Percent Power Test #98
Table 14.2-99	Island Mode Test for the First NuScale Power Module Test #99
Table 14.2-100	Island Mode Test for Multiple NuScale Power Modules Test #100
Table 14.2-101	Remote Shutdown Controls and Monitoring Test #101
Table 14.2-102	NuScale Power Module Vibration #102

The staff confirmed that each of the test abstracts identified above from NuScale SDAA Part 2, Section 14.2, contains the necessary prerequisites, acceptance criteria, and test methods to satisfy the guidance in DSRS Section 14.2 and RG 1.68 for the SDAA review.

Although the NRC is approving only the test abstracts listed in the table above, the staff notes that additional test abstracts that the NRC is not reviewing at this stage are included in SDAA Part 2, Section 14.2.12. Test abstracts not listed in the table above are not approved by the staff and must be addressed by any license applicant. Table 14.2-2 lists the tests that the staff did not evaluate at the SDAA stage and must be reviewed at the COL stage. Further, Section 9.3.2 and Section 9.3.4 of this SER discuss the programmatic leakage control program requirements of 10 CFR 50.34(f)(2)(xxvi), including the requirement for an associated ITP, which must be addressed at the COL stage.

Table 14.2-2 NuScale Section 14.2 Test Abstracts Not Reviewed at the SDAA Stage

Abstract	Test Title
Table 14.2-9[3]	Potable Water System Test #09
Table 14.2-62[4]	Seismic Monitoring System Test #62

14.2.1.5 Combined License Information Items

SDAA Part 2, Table 1.8-2, "Combined License Information Items," lists COL information item numbers and descriptions related to Section 14.2, from SDAA Part 2.

Table 14.2-3 COL Information Items, SDAA Part 2, Table 1.8-1

Item No.	Description	SDAA Section
COL Item 14.2-1	An applicant that references the NuScale Power Plant US460 standard design will describe the site-specific organizations that manage, supervise, or execute the Initial Test Program, including the associated training requirements.	14.2
COL Item 14.2-2	An applicant that references the NuScale Power Plant US460 standard design will develop the Startup Administration Manual that will contain the administrative procedures and requirements that control the activities associated with the Initial Test Program. The applicant will provide a milestone for completing the Startup Administrative Manual and making it available for Nuclear Regulatory Commission inspection.	14.2
COL Item 14.2-3	An applicant that references the NuScale Power Plant US460 standard design will identify the specific operator training to be conducted during low-power testing related to the resolution of Three Mile Island Action Plan Item I.G.1, as described in NUREG-0660, NUREG-0694, and NUREG-0737.	14.2
COL Item 14.2-4	An applicant that references the NuScale Power Plant US460 standard design will provide a schedule for the Initial Test Program.	14.2
COL Item 14.2-5	An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the potable water system pre-operational testing.	14.2
COL Item 14.2-6	An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the seismic monitoring system pre-operational testing.	14.2

14.2.1.6 Conclusion

The staff completed its review of the NuScale ITP at the SDAA stage in accordance with the requirements of 10 CFR 30.53, "Tests"; 10 CFR 50.43, "Additional standards and provisions affecting class 103 licenses and certifications for commercial power"; 10 CFR 52.137, "Contents of applications; technical Information"; 10 CFR 50.34, "Contents of applications; technical information"; 10 CFR 52.79, "Contents of applications; technical information in final safety

[3] COL Item 14.2-5 states, "An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the potable water system pre-operational testing."
[4] COL Item 14.2-6 states, "An applicant that references the NuScale Power Plant US460 standard design will provide a test abstract for the seismic monitoring system pre-operational testing."

analysis report”; Section III.A.4 of Appendix J to 10 CFR Part 50; and Criterion XI of Appendix B to 10 CFR Part 50. The staff concludes that the applicant has provided sufficient information in the ITP for the test abstracts indicated in Table 14.2-1 above and adequately addressed the methods and the applicable guidance in DSRS Section 14.2 and RG 1.68. As previously stated, the test abstracts contained in Table 14.2-2 above were not evaluated as part of the staff’s SDAA ITP review and will need to be reviewed and approved at the COL stage. Except for the tests outlined in Table 14.2-2, the staff concludes that the applicant’s ITP is acceptable.

14.3 Inspections, Tests, Analyses, and Acceptance Criteria

Basis for Inclusion of ITAAC in an SDAA

The requirements for contents of applications and technical information for an SDA in Subpart E to 10 CFR 52.137 do not specify the inclusion of ITAAC. The applicant has voluntarily submitted ITAAC as part of the NuScale US460 SDAA. In assessing the appropriateness of including ITAAC in an SDAA, the staff notes that the 2007 10 CFR Part 52 final rule contains information as to how the Commission views the contents of an application for an SDA compared to a DC application. Specifically, in the section of changes to 10 CFR Part 52, the discussion of changes to Subpart E (72 FR 49391) contains the following statement:

> The Commission has decided that the contents of applications for design approvals should contain essentially the same technical information that is required of design certification applications…

The contents of applications for a DC include the requirements for the inclusion of ITAAC in 10 CFR 52.47(b)(1) as follows:

> The proposed inspections, tests, analyses, and acceptance criteria 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 conformity with the design certification, the provisions of the Act, and the Commission's rules and regulations…

Therefore, because the Commission has decided that the contents of an SDA should contain essentially the same technical information as a certified design, and ITAAC requirements are included in the contents of an application for a certified design, it is appropriate for ITAAC to be included in the SDAA if the applicant chooses to do so.

Relationship Between Top-Level Design Features and ITAAC

Final Safety Analysis Report (FSAR) Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria,” includes information on how NuScale developed the ITAAC included in the US460 SDAA. The section describes criteria and methods used to identify top-level design features, a portion of which were identified and selected to be verified by ITAAC. Tier 1 information from the NuScale US600 DC represents information that is certified and is part of the rulemaking for the certified design. The Tier 1 information also forms the basis for the US600 ITAAC. For the US460, which is an SDA, there is no Tier 1 information. As a result, NuScale used top-level design features (analogous to Tier 1 information in the US600 DC) to develop the ITAAC for the US460. The staff agrees that this is an acceptable approach.

Section 14.3.1 of this SER evaluates the process used to develop the ITAAC for the NuScale US460.

Regarding ITAAC, the staff provided NuScale with two letters, dated April 8, 2016, and June 21, 2016 (ML16096A121 and ML16160A179, respectively) that included a set of standardized ITAAC. The staff endorsed this standardized ITAAC guidance for use in the NuScale US600 DC application. NuScale subsequently used this standardized ITAAC guidance in its US460 SDAA. The staff has evaluated the ITAAC contained in the US460 SDAA against the standard ITAAC guidance. SDAA Part 8, "License Conditions; Inspections, Tests, Analyses, and Acceptance Criteria," contains the top-level design features, system descriptions, and design commitments that form the basis for the ITAAC. In addition, Part 8 includes the ITAAC for the NuScale US460 design. Sections 14.3.2 through 14.3.13 of this SER evaluate the ITAAC for the NuScale US460 design.

14.3.1 Selection Criteria for Standard Design Approval Application

14.3.1.1 Introduction

The NuScale SDAA Part 8 information includes the following:

- definitions and general provisions
- design descriptions
- ITAAC
- ITAAC additional information tables

The purpose of the ITAAC portion of the SDAA Part 8 information is to propose ITAAC that, if the licensee performs the inspections, tests, and analyses and the acceptance criteria are met, the facility has been constructed and will be operated in accordance with the license; the Atomic Energy Act of 1954, as amended (AEA); and applicable regulations. The principal performance characteristics and safety functions of the SSCs are verified by the appropriate ITAAC.

14.3.1.2 Summary of Application

SDAA Part 8: The SDAA Part 8 information is summarized below.

Definitions and General Provisions: The definitions and general provisions are provided in SDAA Part 8, Sections 1.1, "Definitions," and 1.2, "General Provisions."

Design Descriptions: Design descriptions are provided in each subsection of SDAA Part 8, Section 2.0, "Module-Specific Structures, Systems, and Components Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design Descriptions and ITAAC," and Section 3.0, "Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Based Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design Descriptions and ITAAC." The unit-specific descriptions in SDAA Part 8, Section 2.0, "Module-Specific Structures, Systems, and Components Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design Descriptions and ITAAC," apply to each NuScale module, while SDAA Part 8, Section 3.0, "Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Based Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design

Descriptions and ITAAC," addresses SSCs that support multiple NuScale modules. The design description consists of the system description and design commitments. The ITAAC are used to verify the design features in these commitments.

ITAAC: Sections 2.0 and 3.0 of SDAA Part 8 contain the ITAAC. Section 2.0 provides the ITAAC tables for SSCs that support a single NPM and should be completed for each module of a multiunit plant. Section 3.0 provides the ITAAC tables for SSCs that are shared by multiple NPMs.

ITAAC Additional Information: SDAA Part 8, Table 2.1-2, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information," provides additional information for each entry in ITAAC Table 2.1-1, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.01.XXX)." Similarly. there are accompanying tables of additional information for each ITAAC table in SDAA Part 8.

SDAA Part 2: SDAA Part 2, Section 14.3, discusses the development of ITAAC. SDAA Part 2, Section 14.3.2, "Top-Level Design Features and Inspections, Tests, Analyses, and Acceptance Criteria First Principles," describes the criteria used to identify the scope of top-level design features and the scope of the ITAAC. SDAA Part 2, Section 14.3.3, "Inspections, Tests, Analyses, and Acceptance Criteria Information," describes the information presented in the ITAAC tables. SDAA Part 2, Section 14.3.4, "Treatment of Module-Specific and Shared Structures, Systems, and Components in Inspections, Tests, Analyses, and Acceptance Criteria," describes how the application treats ITAAC for module-specific and shared SSCs.

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

Technical Reports: There are no technical reports for this area of review.

14.3.1.3 Regulatory Basis

As discussed above, while ITAAC are not required to be part of an SDAA, NuScale has voluntarily included ITAAC as part of its application. As concluded above, the staff applied the requirements of ITAAC for DCs specified in 10 CFR 52.47(b)(1). These requirements specify that 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 DC has been constructed and will be operated in conformity with the DC, the provisions of the AEA, and the Commission's rules and regulations.

Guidance supporting the staff's review includes the following:

- SRP Section 14.3
- SRP Sections 14.3.2–14.3.4, 14.3.6–14.3.9, 14.3.11, and 14.3.12
- Section 14.3.5 of the DSRS for the NuScale small modular reactor design
- NRC Regulatory Issue Summary (RIS) 2008-05, Revision 1, "Lessons Learned to Improve Inspections, Tests, Analyses, and Acceptance Criteria Submittal," dated September 23, 2010

- Regulatory Guide 1.206, Revision 1, “Applications for Nuclear Power Plants,” issued October 2018

- Regulatory Guide 1.68, Revision 4, “Initial Test Programs for Water-Cooled Nuclear Power Plants,” issued June 2013

- Regulatory Guide 1.45, Revision 1, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage,” issued May 2008

- Regulatory Guide 1.143, Revision 2, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” issued November 2001

- SECY-19-0034 “Improving Design Certification Content,” dated April 8, 2019

In addition, in letters dated April 8, 2016, and June 21, 2016 (ML16096A121 and ML16160A179, respectively), the staff transmitted to NuScale a set of standardized ITAAC that the staff endorsed for use in the NuScale US600 DC application. NuScale used this standardized ITAAC guidance in its US460 SDAA.

14.3.1.4 Technical Evaluation

14.3.1.4.1 *Standard Design Approval Application Part 2, Chapter 14*

SDAA Part 2, Section 14.3, discusses the development of ITAAC. SDAA Part 2, Section 14.3.1, “Introduction,” introduces and lists two COL information items, discussed in Section 14.3.1.5 of this SER. SDAA Part 2, Section 14.3.2, describes the criteria used to identify the scope of top-level design features and the scope of the ITAAC. The staff excludes SDAA Part 2, Section 14.3.2, from its review of this SDAA and does not take a position on the “first principles” described in that section. SDAA Part 2, Section 14.3.2, will not be incorporated into an SDA for the NuScale US460 design. SDAA Part 2, Section 14.3.3, describes the content and organization of ITAAC information. SDAA Part 2, Section 14.3.4, describes the treatment of module-specific and shared SSCs in ITAAC. The staff reviewed the information in these sections and finds it consistent with guidance in SRP Section 14.3 and therefore acceptable. The staff notes that the applicant organized the ITAAC design descriptions and ITAAC based on the structures and systems of the NuScale design rather than on the format of the SRP. Therefore, the subsections in Section 14.3 of this SER are not an evaluation of their corresponding sections in the SDAA.

14.3.1.4.2 *Standard Design Approval Application, Part 8*

14.3.1.4.2.1 Definitions, General Provisions, Design Descriptions, and ITAAC

In accordance with SRP Section 14.3, top-level information should include the principal performance characteristics and safety functions of the standard design and should be verified appropriately by ITAAC. The SDAA has included this information in Part 8 for matters within the SDAA. The design information includes design commitments that identify those features and capabilities that are necessary for compliance with the AEA and NRC rules and regulations and that are to be verified by ITAAC. As stated above, the requirements of ITAAC for DCs are being applied to the SDAA. Specifically, 10 CFR 52.47(b)(1) states that the application must contain 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 DC has been constructed and will be operated in conformity with the DC, the provisions of the AEA, and the Commission's rules and regulations.

For the ITAAC to be “sufficient” as required by 10 CFR 52.47(b)(1), (1) the inspections, tests, and analyses must clearly identify those activities necessary to demonstrate that the acceptance criteria are met, (2) the acceptance criteria must state clear design or performance objectives demonstrating that the design commitments are satisfied, (3) the ITAAC must be consistent with each other and the design commitment, (4) the ITAAC must be capable of being performed and satisfied prior to fuel load, and (5) a facility that incorporates the set of SDA ITAAC, as a whole, must provide reasonable assurance that, if the ITAAC are satisfied, the facility has been constructed and will be operated in conformity with the license, the AEA, and the NRC’s rules and regulations.

Sections 14.3.2 through 14.3.13 of this SER document the staff’s review of the technical adequacy of the ITAAC listed in SDAA Part 8 and its determination as to whether the ITAAC design descriptions have the type of information and the level of detail discussed in SRP Section 14.3 and SECY-19-0034.

The staff reviewed SDAA Part 8 definitions, general provisions, and ITAAC tables for form and clarity in accordance with the guidance provided in SRP Section 14.3 and RIS 2008-05. The staff concludes that the form and clarity of the definitions, general provisions, and design descriptions are acceptable. Furthermore, the staff concludes that (1) the inspections, tests, and analyses and acceptance criteria are consistent with each other and with the design commitment, (2) the inspections, tests, and analyses are clearly stated, and (3) the acceptance criteria are clear and objective.

14.3.1.5 Combined License Information Items

Table 14.3.1-1 lists COL information item numbers and descriptions related to this area of the review from SDAA Part 2, Table 1.8-1. Section 13.3.4.5 of this SER evaluates COL Item 14.3-1. Regarding COL Item 14.3-2, the staff agrees that it is a license applicant’s responsibility to provide the site-specific selection methodology and ITAAC for site-specific SSCs.

Table 14.3.1-1 NuScale COL Information Items, SDAA Part 2, Section 14.3.1

Item No.	Description	SDAA Part 2 Section
14.3-1	An applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for emergency planning.	14.3-1
14.3-2	An applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for structures, systems, and components within their scope.	14.3-1

14.3.1.6 Conclusion

The staff concludes that the form and clarity of the definitions, general provisions, and design descriptions found in SDAA Part 8 are acceptable. Furthermore, the staff concludes that (1) the inspections, tests, and analyses and the acceptance criteria are consistent with each other and with the design commitment, (2) the inspections, tests, and analyses are clearly stated, and (3) the acceptance criteria are clear and objective. Based on the ITAAC review documented in Sections 14.3.2 through 14.3.13 of this SER, the staff finds that the requirements of 10 CFR 52.47(b)(1) are satisfied for matters contained within the SDA because the NuScale ITAAC 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 set of SDA ITAAC has been constructed and will be operated in accordance with the license, the AEA, and the NRC's rules and regulations for those matters considered within the scope of the SDA.

14.3.2 Structural and Systems Engineering—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.2.1 Introduction

This section reviews ITAAC and top-level design descriptions applicable to structural and systems engineering. The following SDAA Part 8 tables contain the ITAAC applicable to this review area:

- Table 3.11-1, "Reactor Building ITAAC," Numbers 6 and 7
- Table 3.12-1, "Radioactive Waste Building ITAAC," Number 3
- Table 3.13-1, "Control Building ITAAC," Numbers 4 and 5

The purpose of ITAAC Number 6 in Table 3.11-2, ITAAC Number 3 in Table 3.12-2, and ITAAC Number 4 in Table 3.13-1 is to verify that the as-built reactor building (RXB), radioactive waste building (RWB), and control room building (CRB) maintain their structural integrity in accordance with the approved design under the actual design-basis loads and that the in-structure responses for the as-built structure are enveloped by those in the approved design. The purpose of ITAAC Number 7 in SDAA Part 8, Table 3.11-2, and ITAAC Number 5 in SDAA Part 8, Table 3.13-1, is to verify that as-built nonseismic Category I SSCs located where a potential for adverse interaction with a seismic Category I SSC exists will not impair the ability of the seismic Category I SSC to perform its safety functions during or following a safe-shutdown earthquake (SSE).

14.3.2.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.2.3 Regulatory Framework for ITAAC Methodology Development

See Section 14.3.1.3 of this SER. SRP Section 14.3.2, Revision 0, "Structural and Systems Engineering—Inspections, Tests, Analyses, and Acceptance Criteria," issued March 2007, provides acceptance criteria and additional guidance for this review area.

14.3.2.4 Technical Evaluation

The staff reviewed the ITAAC information in SDAA Part 8, described in Section 14.3.2.1. The staff reviewed the structural design descriptions for the RXB, RWB, and CRB structures in

SDAA Part 8, Sections 3.11, 3.12, and 3.13, and finds that the level of structural information provided is consistent with that included in the enclosure to SECY-19-0034 covering the level of structural information that should be in the top-level design features.

As noted in SECY-19-0034, the acceptance criteria for ITAAC addressing a complex set of activities (e.g., structural integrity analyses, American Society of Mechanical Engineers (ASME) code compliance) usually require reports demonstrating that the top-level design commitments are met. This is accomplished for systems such as the reactor coolant system (RCS), containment pressure boundary (PB), RXB, RWB, CRB, and various safety and non-safety SSCs through various supplemental design-relevant TRs (www.nrc.gov/reactors/new-reactors/smr/licensing-activities/current-licensing-reviews/nuscale-us460/topical-reports.html), some of which are listed in Table 14.3.2-1.

Table 14.3.2-1 NuScale Topical Reports (TR) in support of the SDAA

Topical Report	Title
TR-0118-58005-P-A	"Improvements in Frequency Domain Soil-Structure-Fluid Interaction Analysis," Revision 2
TR-0920-71621-A	"Building Design and Analysis Methodology for Safety-Related Structures," Revision 1
TR-108553-P-A	"Framatome Fuel and Structural Response Methodologies Applicability to NuScale," Supplement 1 to TR-0116-20825-P-A, Revision 1 and Supplement 1 to TR-0716-50351-P-A, Revision 1," Revision 0
TR-0116-21012-P-A	"Applicability Range Extension of NSP4 Critical Heat Flux Correlation, Supplement 1 to TR-0116-21012-P-A, Revision 1," TR-107522, Revision 1
TR-124587	"Extended Passive Cooling and Reactivity Control Methodology," Revision 0
TR-0516-49416	"Non-Loss-of-Coolant Accident Analysis Methodology," Revision 4
TR-0516-49422	"Loss-of-Coolant Accident Evaluation Model," Revision 3
TR-0915-17772-P-A	"Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," Revision 3
TR-141299	"NuScale Power Plant Design Capability to Mitigate Beyond-Design-Basis Events Defined by 10 CFR 50.155," Revision 0

The staff review of the reports confirms the validity of the SDAA ITAAC for assignment of top-level design features for the seismic Category I RXB, the CRB, and the radioactive waste category RW-IIa RWB structures. The following subsections examine specific ITAAC, indicating their importance in future evaluations and potential closure of COL action items.

14.3.2.4.1 *As-Built Reconciliation*

The staff's review is based on the requirement of 10 CFR 52.47(b)(1) that proposed ITAAC 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 has been constructed and will operate in accordance with the design, the AEA, and NRC rules and regulations. Details in their development are subject to specific input in Sections 3.7 and 3.8 of SDAA Part 2, Chapter 3, "Design of Structures, Systems, Components and Equipment" of the FSAR (ML23304A321), supplemented by TRs that describe analysis results for loads, loading conditions, and effects for the specific construction to be used in the US460. For example, TR-121515-NP, "US460 NuScale Power Module Seismic Analysis," issued December 2022, details the coupling of NPMs to the RXB to yield US460-specific loads for fluid structure interaction, for in-structure response spectra, and for in-structure time histories for the mechanical design of seismic Category I SSCs for the RXB for the US460 NPMs.

The staff notes that COL Item 14.3-2, reviewed and evaluated in Section 14.3.1 of this SE, attunes to the US460 NPMs as it states that "an applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for structures, systems, and components within their scope." An applicant referencing an SDA verifies that the ITAAC apply to those portions of the facility final design that are approved. The as-built reports reconcile design deviations, if any, from the approved final design for construction, concluding that the as-built structures maintain their structural integrity. This provides the staff reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a plant that incorporates the design approval has been constructed and will be operated as required.

The as-built structures, therefore, are acceptable under the actual design-basis loads to demonstrate that their structural integrity is maintained such that the in-structure responses are enveloped by those in the approved design and that a potential adverse interaction between as-built nonseismic Category I SSCs and a seismic Category I SSC will not impair the ability of the seismic Category I SSC to perform its safety functions during or following an SSE. A design summary report documents that the seismic Category I structures meet the acceptance criteria specified in Section 3.7 and Section 3.8 of the FSAR. The following subsections document the staff's review of the ITAAC against these criteria.

14.3.2.4.2 *ITAAC for Structural Integrity of Safety-Related Structures*

Consistent with SECY-19-0034 for "the structural integrity review, the ITAAC [are] generally sufficient if they verify that the as-built safety-significant structures maintain their structural integrity under design-basis loads in accordance with the supporting Tier 2 information, and that as-built Seismic Category I structures are appropriately protected from adverse interaction with SSCs that are not Seismic Category I." The staff notes that, for the structural integrity review, the ITAAC are sufficient if they verify that the as-built safety-significant structures conform to design-basis loads, loading conditions, and environment; are within the design code limits; and meet their intended function(s) without loss of structural integrity or compromise of their safety-related functions noted in the FSAR. As noted above, SRP Section 14.3.2 provides guidance to identify top-level design features that need ITAAC to ensure that all seismic Category I building structures and their safety-related SSCs are verified.

The staff reviewed the structural integrity ITAAC for seismic effects for the RXB, RWB, and CRB housing the main control room (MCR) structures identified in SDAA Part 8. The staff notes that the ITAAC in the SDAA for the RXB, RWB, and CRB are Number 6 in Table 3.11-1, Number 3 in Table 3.12-1, and Number 4 in Table 3.13-1, respectively. These ITAAC state that, based on inspection, reconciliation analyses will be performed of the as-built RXB, RWB, and CRB structures. The ITAAC acceptance criteria state that a design summary report will document the reconciliation analysis and conclude that (1) the as-built building structure maintains its structural integrity, in accordance with the approved design under the actual design-basis loads for the as-built structure, and (2) the in-structure responses for the as-built buildings are enveloped by those in the approved design.

The staff notes that SDAA Section 3.8.4.5.1, "Design Summary Report," of Chapter 3, "Design of Structures, Systems, Components and Equipment, Final Safety Analysis Report," Revision 1, states that, in its Appendix 3B, "Design Reports and Critical Section Details," the seismic Category I structures meet acceptance criteria presented in Section 3.7 and Section 3.8.

Appendix 3B summarizes the structural design and analysis of the RXB and CRB and states the following:

> Section 3.8.4 and Section 3.8.5 describe these structures, their foundations, and the primary loads and load combinations. This appendix describes how those loads are combined and how the design is checked for adequacy. In addition, analysis and design of a selection of critical sections are described in detail. These sections are critical in that they represent parts of the structure that: (1) perform a safety-critical function, (2) are subjected to large stress demand, (3) are considered difficult to design or construct, or (4) are considered to be representative of the structural design. For example, the walls and basemat slab at the NuScale Power Module (NPM) bays satisfy the first three criteria.

The staff notes that SDAA Section 3.8.4.5.1 aligns with the Commission's intent in SECY-19-0034 for a design summary report documenting the results of a reconciliation analysis of the cumulative effect of changes between the approved design and the as-built SSCs for seismic Category I structures where "[d]eviations from the design are tracked as required by 10 CFR Part 50, Appendix B, and evaluated consistent with the methods and procedures of Section 3.7 and Section 3.8." The staff also notes that, depending on the extent of the deviation, the evaluation may range from documentation of the basis of an engineering judgment to inclusion of the change in the performance of a revised analysis.

The staff also reviewed the corresponding discussions of the SDAA ITAAC supporting information for the RXB, RWB, and CRB housing the MCR in NuScale US460 Plant SDAA Part 8, Tables 3.11-2 (ITAAC No. 03.11.06), 3.12-2 (ITAAC No. 03.12.03), and 3.13-2 (ITAAC No. 03.13.04), and finds that the SDAA ITAAC are consistent with the FSAR supportive information. The staff notes that for each of the seismic Category I portions of the RXB and CRB, the design complies with the regulatory framework described above and is designed to withstand the effects of earthquakes without loss of capability to perform the safety functions verified by ITAAC. The staff further notes that the RWB is a reinforced concrete building, and its primary function is to house non-safety-related SSCs. Its below-grade portion, classified as an RW-IIa structure, maintains its structural integrity to seismic loads verified by ITAAC. Based on the applicant's description of the design summary report for as-built structures, the staff finds that the ITAAC for the RXB, CRB, and RW-IIa (RWB-below grade) structures are acceptable to verify the structural integrity of the as-built structures under the actual design-basis loads.

14.3.2.4.3 *ITAAC for Seismic Interaction of Seismic Category I Structures, Systems, and Components/Nonseismic Category I Structures, Systems, and Components*

The applicant provided ITAAC to verify that as-built seismic Category I SSCs are designed to withstand the effects and to maintain the specified design functions following the SSE. Seismic Category II and nonseismic structures are designed or physically arranged (or both) so that the SSE could not cause unacceptable structural interactions potentially impairing the ability of seismic Category I SSCs to perform their safety functions during or following an SSE. These ITAAC for the RXB and the CRB are identified as Number 7 and Number 5 in SDAA Part 8, Tables 3.11-1 and 3.13-1. Supporting information for the ITAAC found in SDAA Part 8, Tables 3.11-2 (ITAAC No. 03.11.07) and 3.13-2 (ITAAC No. 03.13.05), reference FSAR Chapter 3 (e.g., Section 3.2.1). These ITAAC, along with the corresponding discussions in FSAR Chapter 14, Table 14.3-2, conform to the standardized ITAAC design commitments and associated FSAR discussion in the standardized ITAAC guidance. Also, the staff finds that the top-level design descriptions and ITAAC are based on and are consistent with the FSAR material. Therefore, the staff finds ITAAC Number 7 in SDAA Table 3.11-2 for the RXB, and ITAAC Number 5 in SDAA Table 3.13-2 for the CRB, referencing the FSAR are sufficient to verify that as-built seismic Category I structures are protected from adverse seismic interactions with nonseismic Category I SSCs.

14.3.2.5 Combined License Information Items

The applicant did not identify any COL information items associated with ITAAC for the RXB, RWB, and CRB structures.

14.3.2.6 Conclusion

The staff reviewed Section 14.3.2 of SDAA Chapter 14 for the ITAAC development methodology and found it to be a standalone part of the application providing insights for identifying top-level design features necessitating ITAAC for SSCs that are safety related, the non-safety-related failure of which could affect those that are safety-related, and those determined by probabilistic risk assessment to be risk significant and with specific application to RXB, RWB, and CRB SSCs. The staff found that SDAA Part 8 discusses ITAAC for reconciliation analyses, inspections, and testing of the as-built RXB, RWB, and CRB structures substantiated by SDAA Part 2, Chapter 3, in accordance with those in SDAA Part 8 ITAAC and SDAA Part 2, Chapter 14, supplemental information, and in SECY-19-0034 and therefore acceptable. The staff also reviewed the applicable regulatory framework and noted that ITAAC need not be resolved by 10 CFR Part 52, Subpart E. However, the development of the SDAA SE is critical to support a COL application, as it may include ITAAC from the SDAA with updated change of design information. The staff therefore concludes that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the NuScale SDAA has been constructed and will be operated in accordance with the design, the provisions of the AEA, and NRC rules and regulations.

14.3.3 Piping Systems and Components—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.3.1 Introduction

This section reviews ITAAC and top-level design features descriptions applicable to piping and components. The following SDAA Part 8 tables contain the ITAAC applicable to this review area:

- Table 2.1-1, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 1–4
- Table 2.1-1, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 13–15
- Table 2.1-1, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 5, 11, and 18–21
- Table 2.2-1, "Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 1 and 2
- Table 2.4-2, "Equipment Qualification—Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 1, 3, 6, and 7
- Table 3.1-1, "Control Room Habitability System Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 2 and 3
- Table 3.11-1, "Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria," Number 8
- Table 3.14-1, "Equipment Qualification—Shared Equipment Inspections, Tests, Analyses, and Acceptance Criteria," Number 1

14.3.3.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.3.3 Regulatory Basis

See Section 14.3.1.3 of this SER. SRP Section 14.3.3, "Piping Systems and Components—Inspections, Tests, Analyses, and Acceptance Criteria," provides acceptance criteria and additional guidance for this review area.

14.3.3.4 Technical Evaluation

SRP Section 14.3.3 discusses nine specific areas related to piping and components: piping stress analysis, pipe break analysis, leak-before-break (LBB) evaluation, as-built reconciliation, piping and component safety classification, fabrication (welding), hydrostatic testing, seismic and dynamic qualification of equipment, and valve qualification. The staff gives its technical evaluation of the piping stress analysis, pipe break analysis, LBB analysis, piping and component safety classification, environmental and seismic and dynamic qualification of equipment, and valve qualification in Sections 3.12, 3.6.2, 3.6.3, 3.2.2, 3.10, 3.11, and 3.9.6 of this SER, respectively.

The staff has confirmed that the information in SDAA Part 8 associated with this review area is consistent with the guidance in SRP Section 14.3.3. The staff has also reviewed the contents of SDAA Part 8 and ensured that it contains the top-level design features expected for the piping and components of the design. These top-level design features include compliance with the ASME Boiler and Pressure Vessel (BPV) Code, Section III, protection of safety-related SSCs from dynamic and environmental effects of postulated piping failures, LBB analysis, safety classification, and qualification and testing of equipment.

14.3.3.4.1 *Generic Piping Design*

14.3.3.4.1.1 Piping Stress Analysis

SER Section 3.12 discusses the staff's review of the piping stress analysis. ITAAC Number 1 in SDAA Part 8, Table 2.1-1, requires the ASME BPV Code Class 1, 2, and 3 as-built piping systems to comply with ASME BPV Code, Section III, requirements through the completion of ASME BPV Code, Section III, Design Reports. As noted in SDAA Part 8, Table 2.1-2, discussion for ITAAC Number 1 in SDAA Part 8, Table 2.1-1, the ASME BPV Code requires a Design Report for each ASME BPV Code Class 1, 2, and 3 component (including piping systems), in accordance with NCA-3550, "Requirements for Design Output Documents." This Design Report must be reconciled with the as-built component in accordance with NCA-3554, "Modification of Documents and Reconciliation with Design Report." The applicant's proposed ITAAC and the associated discussion in SDAA Part 8, Table 2.1-2, for compliance with ASME BPV Code, Section III, are consistent with the standardized ITAAC guidance. Based on the technical review conducted in Section 3.12 of this SER, which concludes that the piping stress analysis methodologies are consistent with ASME BPV Code, Section III, requirements, and based on the applicant's use of proposed ITAAC and associated discussion in SDAA Part 8, Table 2.1-2, that are consistent with the standardized ITAAC guidance, the NRC staff finds these ITAAC acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for piping stress analysis within the scope of the SDA.

14.3.3.4.1.2 Pipe Break Analysis

Section 3.6.2 of this SER discusses the staff's review of the pipe break analysis. The ITAAC for protection of safety-related SSCs from dynamic and environmental effects (SDAA Part 8, Table 2.1-1, ITAAC Number 4) is in the section for the NPM and includes the areas up to and including the reactor pool bay wall. Those areas beyond the reactor pool bay wall were covered by COL Item 3.6-1 and an additional ITAAC (SDAA Part 8, Table 3.11-1, ITAAC Number 8). The staff has confirmed that ITAAC cover the full scope of the plant area for the standard design where pipe breaks may be postulated.

The applicant's proposed ITAAC, located in ITAAC tables for the NPM and the RXB, and the associated discussion in SDAA Part 8, Table 2.1-2 and Table 3.11-2, for pipe break analysis, are consistent with the standardized ITAAC guidance. Therefore, based on the technical review conducted in SER Section 3.6.2, which determines the technical adequacy of the applicant's pipe break analysis methodologies, and the applicant's use of proposed ITAAC and the associated top-level design features discussion consistent with the standardized ITAAC guidance, the staff finds that these ITAAC and associated discussion are acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for pipe break analysis within the scope of the SDA.

14.3.3.4.1.3 Leak-Before-Break Evaluation

Section 3.6.3 of this SER discusses the staff's review of LBB analysis. NuScale states in SDAA Part 2, Section 3.6.3, that the US460 power plant does not use the LBB methodology, so there are no ITAAC to evaluate for this area of review.

14.3.3.4.1.4 As-Built Reconciliation

The topic of as-built reconciliation is covered through ITAAC, requiring that as-built ASME BPV Code piping and components meet the requirements of ASME BPV Code, Section III. As noted in SDAA Part 8, Table 2.1-2, and the discussion for ITAAC Number 1 in SDAA Part 8, Table 2.1-1, the ASME BPV Code requires a Design Report for each ASME BPV Code Class 1, 2, and 3 component in accordance with NCA-3550. This Design Report must be reconciled with the as-built component in accordance with NCA-3554. This reconciled Design Report ensures that the as-built design meets the ASME BPV Code requirements but does not ensure the adequacy of construction activities. ASME BPV Code, Section III, also requires that a Data Report be prepared to verify that the ASME BPV Code requirements are met for the as-built components. A Data Report (which references the previously mentioned reconciled Design Report) addresses the adequacy of construction for each component and ensures that the as-built component meets the ASME BPV Code requirements.

ITAAC Number 1 in SDAA Part 8, Table 2.1-1, verifies compliance with ASME BPV Code, Section III, requirements for ASME BPV Code Class 1, 2, and 3 piping systems through inspection of ASME BPV Code, Section III, Design Reports for as-built piping systems. These ITAAC and the associated discussion are acceptable, as discussed in SER Section 14.3.3.4.1.1. ITAAC Number 1 in SDAA Part 8, Table 2.1-1, verifies compliance with ASME BPV Code, Section III, requirements for ASME BPV Code Class 1, 2, and 3 components and core support (CS) components through inspection of ASME BPV Code, Section III, Design Reports, for as-built components. These ITAAC and the associated top-level design features discussions are acceptable, as discussed in SER Sections 3.9.3.4.1 and 3.9.5.4.6.2. ITAAC Numbers 2 and 3 in SDAA Part 8, Table 2.1-1, verify that the ASME BPV Code Class 1, 2, 3, and CS components and interconnecting piping comply with ASME BPV Code, Section III, requirements through the completion of ASME BPV Code, Section III, Data Reports for the ASME BPV Code Class 1, 2, 3, and CS components and interconnecting piping. The applicant's proposed ITAAC and the associated discussion in SDAA Part 8, Table 2.1-2, for as-built reconciliation are consistent with the standardized ITAAC guidance. As these ITAAC and associated discussion are aligned with the staff-approved standardized ITAAC guidance for as-built reconciliation for ASME BPV Code, Section III, compliance, the staff finds these ITAAC and associated discussion acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for as-built reconciliation for ASME BPV Code, Section III, components and interconnecting piping within the scope of the SDA.

Finally, ITAAC Number 5 in SDAA Part 8, Table 2.1-1, verifies the installation of the ECCS supplemental boron dissolvers and containment vessel (CNV) lower mixing tubes in accordance with the associated installation specification. The proposed ITAAC is similarly formatted to ITAAC Number 11 in SDAA Part 8, Table 2.1-1, which is discussed later in this SER for the NPM valves. The staff has reviewed proposed ITAAC Number 5 and the associated discussion in SDAA Part 8, Table 2.1-2, and finds them acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for matters within the scope of the SDA because they verify the installation of the ECCS supplemental boron dissolvers and CNV lower mixing tubes, which is important for their proper functioning. Proper functioning of these features is also confirmed through a proposed first-of-a-kind test, as discussed in Section 6.3.

14.3.3.4.2 *Verifications of Components and Systems*

14.3.3.4.2.1 Piping and Component Safety Classification

Section 3.2.2 of this SER discusses the staff's review of piping and component safety classification. The safety classification of piping and components is a topic that is resolved during the SDAA phase, except for any site-specific elements, which will be reviewed at the time of a COL application. Based on the technical review conducted in SER Section 3.2.2, the staff has identified no specific ITAAC that are required for a piping and component safety classification to meet the requirements of 10 CFR 52.47(b)(1) for matters within the scope of the SDA. The safety classifications assigned in the SDAA will be confirmed through the previously mentioned as-built reconciliation ITAAC, which will ensure that the as-built piping and components are constructed in accordance with the assigned classifications.

14.3.3.4.2.2 Fabrication (Welding)

This section covers the topic of welding primarily through compliance with ASME BPV Code requirements. As previously discussed, the ASME BPV Code requires reports verifying that systems and components meet ASME BPV Code requirements, including welding. Because the topic of ASME BPV Code compliance has previously been discussed and found acceptable in Section 14.3.3.4.1.4 of this SER, no additional issues are identified for this review area.

14.3.3.4.2.3 Pressure Testing

The staff's review of pressure testing is typically covered through compliance with ASME BPV Code requirements, as the pressure test (typically hydrostatic, but in some cases pneumatic) is a required element of ASME BPV Code compliance. The ASME BPV Code ITAAC proposed by the applicant satisfy the pressure testing requirement, in that they require that the applicable ASME BPV Code report demonstrate that the system meets ASME BPV Code requirements, which include pressure testing. Because the topic of ASME BPV Code compliance has previously been discussed and found acceptable in Section 14.3.3.4.1.4 of this SER, no additional issues are identified for this review area.

14.3.3.4.2.4 Environmental and Seismic and Dynamic Qualification of Equipment

Section 3.10 of this SER discusses the staff's review of seismic and dynamic qualification of equipment. Section 3.11 of this SER discusses the staff's review of environmental qualification of mechanical and electrical equipment. ITAAC Number 1 in SDAA Part 8, Table 2.4-2, and ITAAC Number 1 in SDAA Part 8, Table 3.14-1, verify the seismic and dynamic qualification of seismic Category I equipment, including its associated supports and anchorages. The scope of these ITAAC is limited to specific SSCs listed in SDAA Part 8, Table 2.4-3, "Module Specific Mechanical and Electrical/Instrumentation and Controls Equipment," and SDAA Part 8, Table 3.14-3, "Mechanical and Electrical/Instrumentation and Controls Shared Equipment." The staff confirmed that the SDAA Part 8 tables contain the required seismic Category I SSCs. The ITAAC and the associated discussion in SDAA Part 8, Table 2.4-2 and Table 3.14-2, ensure that SSCs listed in the SDAA Part 8 tables will be designed and built to the appropriate standard and remain functional during and after the design-basis earthquake. As the proposed ITAAC and associated top-level design features discussion are consistent with the standardized ITAAC guidance and based on the technical review conducted in SER Section 3.10, the staff finds these ITAAC and the associated discussion acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for the seismic and dynamic qualification of SSCs within the scope of the SDA.

The applicant proposed ITAAC Number 3 in SDAA Part 8, Table 2.4-1, “Equipment Qualification—Module Specific ITAAC,” for the environmental qualification of nonmetallic parts, materials, and lubricants used in safety-related mechanical equipment. The proposed ITAAC and the associated discussion in SDAA Part 8, Table 2.4-2, is consistent with the standardized ITAAC guidance. Therefore, based on this consistency and the technical review conducted in SER Section 3.11, the staff finds this ITAAC and the associated top-level design features discussion acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for the environmental qualification of nonmetallic parts, materials, and lubricants used in safety-related mechanical equipment within the scope of the SDA.

14.3.3.4.2.5 Valve Qualification

Section 3.9.6 of this SER discusses the NRC staff’s review of valve qualification. Based on the safety significance of the proper performance of power-operated valves, the staff considers the process of demonstrating the functional capability of safety-related power-operated valves in the NuScale US460 Power Plant to be appropriate as an SDAA Part 8 top-level design feature requirement that should not be modified without prior NRC review. The tables in SDAA Part 8 discuss the use of ASME QME-1 as referenced in FSAR Section 3.10, which provides a controlled process for demonstrating the qualification of safety-related power-operated valves. The staff finds that the wording of ITAAC Number 6 in SDAA Part 8, Table 2.4-1, and the associated discussion in SDAA Part 8, Table 2.4-2, is consistent with the standardized ITAAC guidance after incorporation of these changes. Based on this, as well as the technical review in SER Section 3.9.6, the NRC staff finds that ITAAC Number 6 in SDAA Part 8, Table 2.4-1, and the associated discussion in SDAA Part 8, Table 2.4-2, are acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for the functional qualification of safety-related valves within the scope of the SDA.

The staff reviewed SDAA Part 8, Section 2.4, “Equipment Qualification—Module Specific,” to ensure that it included all applicable safety-related valves in the NPM and that ITAAC Number 6 in SDAA Part 8, Table 2.4-1, for equipment qualification, included qualification of safety-related valves in all environments, and their applicable fluid conditions. The proposed ITAAC Number 6 and the associated discussion in SDAA Part 8, Table 2.4-2, is consistent with the standardized ITAAC guidance. Because the applicant included ITAAC and associated top-level design features discussions, for the equipment qualification of all safety-related valves in the NPM in all environments and their applicable fluid conditions, that are consistent with the standardized ITAAC guidance, the staff finds the proposed ITAAC and associated discussion sufficient for the equipment qualification of safety-related valves in the NPM.

NuScale proposed ITAAC for performance testing under preoperational temperature, differential pressure, and flow conditions for the containment system (CNTS) isolation valves (ITAAC Number 13 in SDAA Part 8, Table 2.1-1), emergency core cooling system (ECCS) valves (ITAAC Number 14 in Table 2.1-1), decay heat removal system (DHRS) valves (ITAAC

Number 15 in Table 2.1-1), CNTS check valves (ITAAC Number 21 in Table 2.1-1), chemical and volume control system (CVCS) demineralized water supply isolation valves (ITAAC Number 1 in SDAA Part 8, Table 2.2-1), and control room habitability system (CRHS) valves (ITAAC Number 2 in SDAA Part 8, Table 3.1-1). The proposed ITAAC and associated discussion in SDAA Part 8, Table 2.1-2, Table 2.2-2, and Table 3.1-2, are consistent with the standardized ITAAC guidance for these preoperational tests. Based on this, as well as the technical review in SER Section 3.9.6, the staff finds that the ITAAC and associated SDAA Part 8 discussion noted above are acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for performance testing for valves under preoperational temperature, differential pressure, and flow conditions within the scope of the SDA.

NuScale proposed ITAAC for loss of motive power testing under preoperational temperature, differential pressure, and flow conditions for the CNTS hydraulic-operated valves (ITAAC Number 18 in SDAA Part 8, Table 2.1-1), ECCS reactor recirculation valves and reactor vent valves (ITAAC Number 19 in Table 2.1-1), DHRS hydraulic-operated valves (ITAAC Number 20 in Table 2.1-1), CVCS demineralized water supply isolation valves (ITAAC Number 2 in SDAA Part 8, Table 2.2-1), and CRHS solenoid-operated valves (ITAAC Number 3 in SDAA Part 8, Table 3.1-1). The proposed ITAAC and associated discussion in SDAA Part 8, Table 2.1-2, Table 2.2-2, and Table 3.1-2, are consistent with the standardized ITAAC guidance for loss of motive power testing. Based on this, as well as the technical review in SER Section 3.9.6, the staff finds that the ITAAC and associated discussion listed above are acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for loss of motive power testing under preoperational temperature, differential pressure, and flow conditions for valves within the scope of the SDA. NuScale also proposed ITAAC Number 7 in SDAA Part 8, Table 2.4-1, to demonstrate the set pressure, capacity, and overpressure design requirements for safety-related relief valves. The proposed ITAAC and associated discussion in SDAA Part 8, Table 2.4-2, cover the full scope of safety-related relief valves and are consistent with the standardized ITAAC guidance for ASME BPV Code, Section III, Relief Valve Capacity Qualification. Therefore, the staff finds this ITAAC and the SDAA Part 8 associated discussion acceptable for meeting the requirements of 10 CFR 52.47(b)(1) to demonstrate the set pressure, capacity, and overpressure design requirements for safety-related relief valves within the scope of the SDA. NuScale included ITAAC Number 11 in SDAA Part 8, Table 2.1-1, to verify the installation of the ECCS valves, containment isolation valves (CIV), and DHRS actuation valves and their associated hydraulic lines. The staff has reviewed the proposed ITAAC and the associated discussion in SDAA Part 8, Table 2.1-2, and finds them acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for matters within the scope of the SDA because they verify the installation of the NPM valves, which is important for their proper functioning.

14.3.3.5 Combined License Information Items

There are no COL items for this section.

14.3.3.6 Conclusion

The NRC staff reviewed the SDAA Part 8 information in accordance with the guidance in SRP Section 14.3.3 and SECY-19-0034. The staff finds that, for the topics discussed above, the applicant has met the requirements of 10 CFR 52.47(b)(1) for matters within the scope of the SDA by proposing ITAAC that are sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the piping systems and components have been constructed and installed in conformity with the license and will be operated in conformity with the license, the provisions of the AEA, and the NRC's rules and regulations. The staff also concludes that the applicant has included sufficient top-level design information in SDAA Part 8, consistent with SECY-19-0034.

14.3.4 Reactor Systems—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.4.1 Introduction

This section reviews ITAAC and top-level design descriptions applicable to reactor systems. The following SDAA Part 8 tables contain the ITAAC that are reviewed in this section:

- Table 2.1-1, "NuScale Power Module ITAAC, Numbers 5, 8, 11, 14,15, 19, 20
- Table 2.2-1, "Chemical and Volume Control System ITAAC,"
- Table 2.4-1, "Equipment Qualification- Module Specific ITAAC," Number 8

SER Section 14.3.4.4 references additional ITAAC for reactor systems that are evaluated elsewhere in this report. The staff addresses reactor systems piping and mechanical ITAAC in Sections 14.3.3.4.3.4 and 14.3.3.4.3.5 of this SER. Section 14.3.3.4.3.4 addresses component electrical ITAAC, and Section 14.3.5.4 addresses ITAAC for automatic reactor trip functions, engineered safety functions, and manual switches. These ITAAC are identified in their respective sections.

14.3.4.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.4.3 Regulatory Basis

See Section 14.3.1.3 of this SER. SRP Section 14.3.4, "Reactor Systems—Inspections, Tests, Analyses, and Acceptance Criteria," provides acceptance criteria and additional guidance for this review area.

14.3.4.4 Technical Evaluation

Consistent with SRP Section 14.3.4 and SECY-19-0034, the design descriptions and ITAAC adequately describe the top-level design features and performance characteristics that are significant to safety. The staff reviewed the design description and system ITAAC to confirm completeness and consistency with the system design-basis as described in various SDAA Part 8 sections and concludes that the design description and ITAAC are consistent. The reactor systems ITAAC, along with the corresponding discussions in SDAA Part 8, Tables 2.1-1, 2.1-2, 2.2-1, 2.2-2, 2.4-1, and 2.4-2, generally conform to the standardized SDAA ITAAC, design commitments, and associated discussions in the standardized ITAAC guidance.

The requirements of 10 CFR 52.47(b)(1) are met, in part, by identifying ITAAC to verify the top-level design features of the reactor systems in the SDAA.

The subsections below present the staff's review of the reactor systems' ITAAC, listed in order of the associated SDAA Part 8 sections.

14.3.4.4.1 *Fuel Assembly Design (SDAA Part 2, Section 4.2)*

For this section, no proposed ITAAC are designated in SDAA Part 2, Section 14.3.2.1.5, since ITAAC must be satisfied before fuel load, but the fuel assembly design cannot be reasonably verified until fuel load. Therefore, the staff performed an in-depth review of the fuel assembly design in Chapter 4 of this SER.

14.3.4.4.2 *Control Rod Drive System (SDAA Part 2, Section 4.6)*

SDAA Part 2, Table 2.4-2, provides ITAAC for control rod drive system (CRDS) piping and components. The as-built mechanical equipment must comply with ASME BPV Code, Section III, requirements, and electrical equipment must perform its operational function as described in the ITAAC. The staff reviewed the design information and ITAAC associated with the CRDS and concludes that they are complete and adequately describe and verify the design requirements for the CRDS. The staff evaluates each ITAAC identified above, and the associated design descriptions and Section 14.3 material, in other parts of Section 14.3 of this SER, as noted in Section 14.3.4.1.

14.3.4.4.3 *Overpressure Protection System (SDAA Part 2, Section 5.2.2)*

SDAA Part 2, Table 2.4-2, provides ITAAC for overpressure protection system mechanical and electrical equipment. The as-built mechanical equipment must comply with ASME BPV Code, Section III, requirements, and electrical equipment must perform its operational function as described in the ITAAC. The staff reviewed the design information and ITAAC associated with the overpressure protection system and concludes that they are complete and adequately describe and verify the design requirements for the overpressure protection system. The staff evaluates each ITAAC identified above, and the associated design descriptions and Section 14.3 material, in other parts of Section 14.3 of this SER, as noted above in Section 14.3.4.1.

14.3.4.4.4 *Decay Heat Removal System (SDAA Part 2, Section 5.4.3)*

SDAA Part 2, Tables 2.1-1, 2.1-3, 2.1-4, and 2.1-5, provide ITAAC for DHRS piping and mechanical and electrical equipment. The as-built piping and mechanical equipment must comply with ASME BPV Code, Section III, requirements, and electrical equipment must perform its operational function as described in the ITAAC. SDAA Part 8, Table 2.5-2, provides module protection system (MPS) ITAAC for the DHRS related to the automatic reactor trip functions, engineered safety functions, and manual switches, The staff evaluates these ITAAC, and the associated design descriptions and Section 14.3 material, in other parts of Section 14.3 of this SER, as noted above in Section 14.3.4.1.

ITAAC Number 02.04.08 in SDAA Part 8, Table 2.4-2, verifies that the DHRS condensers have the capacity to transfer their design heat load. The staff used the acceptance criteria in SRP Section 14.3 and SRP Section 14.3.11, “Containment Systems—Inspections, Tests, Analyses, and Acceptance Criteria,” and the ITAAC-related guidance in RG 1.206, Section C.2.9, to evaluate the ITAAC. The staff finds the ITAAC acceptable because the DHRS heat removal capability is credited in the Chapter 15 transient analyses for mitigation of non-loss-of-coolant accident design-basis events. Since DHRS heat removal is, in part, a function of the condensers’ capacity to transfer their design heat load, it is essential to confirm the condensers’ heat transfer capacity prior to fuel load.

The staff finds that the DHRS-related ITAAC items mentioned above are necessary and sufficient to verify the SDAA Part 8 design commitments for the operation of the components in the DHRS, as this set of ITAAC, if satisfied, demonstrates that the structural and functional performance requirements of the system are met. The staff has reasonable assurance that, if the proposed inspections, tests, and analyses are performed and the acceptance criteria are met, the as-built top-level design parameters described in SDAA would be in conformity with the standard design with respect to the parameter values used in the safety analyses corresponding to the DHRS. Based on this review, the staff concludes that the top-level functional design for the DHRS is appropriately described in SDAA and is acceptable. Consequently, the staff finds that the NuScale DHRS meets the requirements of 10 CFR 52.47(b)(1).

14.3.4.4.5 *Reactor Coolant System High-Point Vents (SDAA Part 2, Section 5.4.4)*

The RCS does not include a separate safety-related high-point vent capability. However, a non-safety-related high-point degasification line connected to the upper head of the reactor pressure vessel permits venting the pressurizer to the liquid radioactive waste system via the CVCS. SDAA Part 8, Tables 2.1-2, 2.1-3, 2.1-4, 2.4-2, and 2.4-3, provide ITAAC for piping and mechanical and electrical equipment. The as-built piping and mechanical equipment must comply with ASME BPV Code, Section III, requirements, and electrical equipment must perform its operational function as described in the ITAAC. The staff reviewed the design information and ITAAC associated with the high-point vent capabilities and concludes that they are complete and adequately describe and verify the design requirements for the vents. The staff evaluates each ITAAC identified above, and the associated design descriptions and Section 14.3 material, in other sections of Section 14.3 of this SER, as noted above in Section 14.3.4.1.

14.3.4.4.6 *Emergency Core Cooling System (SDAA Part 2, Section 6.3)*

SDAA Part 8, Tables 2.1-2, 2.1-3, 2.1-4, 2.4-2, and 2.4-3, provide ITAAC for ECCS piping and components. The as-built mechanical components must comply with ASME BPV Code, Section III, requirements, and electrical equipment must perform its operational function, as described in the ITAAC. SDAA Part 8, Tables 2.5-1 and 2.5-2, provide MPS ITAAC for the ECCS related to the automatic engineered safety functions and manual switches. The staff reviewed the design information and ITAAC associated with the ECCS and concludes that they are complete and adequately describe and verify the design requirements for the ECCS. The staff evaluates each ITAAC identified above, and the associated design descriptions and Section 14.3 material, in other sections of Section 14.3 of this SER, as noted above in Section 14.3.4.1.

14.3.4.4.7 *Chemical and Volume Control System (SDAA Part 2, Section 9.3.4)*

SDAA Part 8, Tables 2.1-4, 2.1-5, and 2.4-3, provide ITAAC for CVCS piping and components. The ASME BPV Code Class 3 as-built piping and isolation valves connected to the reactor pressure vessel must comply with ASME BPV Code, Section III, requirements. The staff reviewed the design information and ITAAC associated with the CVCS and concludes that they are complete and adequately describe and verify the design requirements for the CVCS. The staff evaluates each ITAAC identified above, and the associated design descriptions and Section 14.3 material, in other sections of Section 14.3 of this SER as noted above in Section 14.3.4.1.

14.3.4.5 Combined License Information Items

SDAA Part 2, Section 14.3, contains no COL information items related to this area of review.

14.3.4.6 Conclusion

The staff reviewed the SDAA Part 8 information related to reactor systems in accordance with the guidance in SRP Section 14.3.4 and SECY-19-0034. The staff finds that the top-level design features and performance characteristics of the reactor system SSCs are appropriately described and are acceptable. In addition, the staff finds that ITAAC Number 02.04.08 for the DHRS in SDAA Part 8, Table 2.4-2, complies with 10 CFR 52.47(b)(1). The staff discusses its conclusions regarding mechanical, electrical, and instrumentation and control (I&C) aspects of reactor systems ITAAC in other sections of Section 14.3 of this SER, as noted above in Section 14.3.4.1.

14.3.5 Instrumentation and Controls—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.5.1 Introduction

This section reviews ITAAC and top-level design descriptions applicable to I&C. The applicant presented information developed for the NuScale I&C-related ITAAC in SDAA Part 8, Section 2.5, “Module Protection System and Safety Display and Indication System”; Section 2.6, “Neutron Monitoring System”; and Section 2.4. The following SDAA Part 8 tables list I&C systems-related ITAAC, along with associated design commitments:

- Tables 2.5-1, 2.5-2, and 2.5-3, “Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 1 through 13
- Tables 2.6-1, 2.6-2, and 2.6-3, “Neutron Monitoring Systems Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 1, 2, and 3
- Tables 2.4-1, 2.4-2, and 2.4-3, “Equipment Qualification—Module Specific Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 4 and 5

14.3.5.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.5.3 Regulatory Basis

See Section 14.3.1.3 of this SER. DSRS Section 14.3.5, “Instrumentation and Controls—Inspections, Tests, Analyses, and Acceptance Criteria,” provides acceptance criteria and additional guidance for this review area.

14.3.5.4 Technical Evaluation

This SER section evaluates the ITAAC information provided in SDAA Part 8, Sections 2.4, 2.5, and 2.6, in accordance with NuScale DSRS Section 14.3.5.

Based on the review of the I&C-related information in SDAA Part 2, Chapter 7, the staff concludes the following regarding I&C-related information:

- Consistent with NuScale DSRS Section 14.3.5 and SECY-19-0034, the ITAAC system descriptions, design commitments, and ITAAC adequately describe the top-level I&C design features and performance characteristics that are significant to safety. For safety-related systems, this included a description of system purpose, safety functions, equipment quality (e.g., meeting the functional requirements of Institute of Electrical and Electronics Engineers (IEEE) Std. 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," and the digital system life cycle design process), automatic decision-making and trip logic functions, manual initiation functions, and design features (e.g., system architecture) provided to achieve high functional reliability.

- Consistent with NuScale DSRS Section 14.3.5 and SECY-19-0034, the functions and characteristics of other I&C systems important to safety are adequately discussed to the extent that the functions and characteristics are necessary to support remote shutdown, support operator actions or assessment of plant conditions and safety system performance, maintain safety systems in a state that ensures their availability during an accident, minimize or mitigate control system failures that would interfere with or cause unnecessary challenges to safety systems, or provide diverse backups to safety systems.

- Consistent with NuScale DSRS Section 14.3.5 and SECY-19-0034, the ITAAC verify the significant features of the I&C systems on which the staff is relying to provide assurance of compliance with each NRC requirement identified in DSRS Chapter 7, "Instrumentation and Controls." Tests, analyses, and acceptance criteria associated with each design commitment, when taken together, are sufficient to provide reasonable assurance that the final as-built I&C system fulfills NRC requirements. The sufficiency of the ITAAC is discussed in greater detail below.

The NRC staff also evaluated whether ITAAC were needed from an I&C perspective for active systems. Based on the I&C design information provided in SDAA Part 2, Chapter 7, the NRC staff finds that no active systems are needed for reactor coolant makeup or decay heat removal, and therefore, no ITAAC is required from an I&C perspective.

14.3.5.4.1 *Module Protection System and Safety Display and Indication System ITAAC*

In SDAA Part 8, Table 2.5-1, the applicant provides ITAAC verifying design features for the MPS and its associated components in the safety display and indication system (SDIS). SDAA Part 8, Section 2.5.1, "ITAAC Design Description," identifies the MPS safety-related functions verified by ITAAC, design commitments associated with MPS and SDIS safety-critical design features, and development life cycle activities.

These ITAAC, along with the corresponding discussions in SDAA Part 8, Table 2.5-2, for ITAAC numbers 02.05.01 through 02.05.13, generally conform to the standardized ITAAC, design commitments, and associated discussion in the standardized ITAAC guidance. The NRC staff finds that the ITAAC are sufficient to demonstrate that the MPS and SDIS perform the safety-related and non-safety-related system functions identified in SDAA Part 8, Section 2.5.1.

Section 14.3.9 of this SER discusses the review of the capability to remotely shut down the reactor outside of the MCR.

14.3.5.4.2 *Neutron Monitoring System ITAAC*

The ITAAC design description in SDAA Part 2, Section 2.6, "Neutron Monitoring System," refers to FSAR Section 7.0.4 for a description of the neutron monitoring system (NMS) and identifies the NMS safety-related functions that are verified by ITAAC.

There is no ITAAC to verify the capability of the as-built NMS to monitor the neutron flux levels in the reactor core because ITAAC must be satisfied prior to initial loading of fuel into the reactor. However, there are ITAAC to appropriately verify physical separation and electrical isolation for NMS Class 1E circuits. The NRC staff finds that the ITAAC in SDAA Part 8, Table 2.6-1, along with the corresponding discussions in SDAA Part 8, Table 2.6-2, for ITAAC 02.06.01 through 02.06.03 conform to the standardized ITAAC, design commitments, and associated discussion in the standardized ITAAC guidance.

Based on the above, the NRC staff finds that the ITAAC for the NMS in SDAA Part 8, Section 2.6, comply with 10 CFR 52.47(b)(1).

14.3.5.4.3 *Equipment Qualification ITAAC*

The applicant provided ITAAC verifying design features for the safety-related digital I&C in SDAA Part 8, Section 2.4. SDAA Part 8, Section 2.4.1, in part, states the following:

> The Class 1E computer-based instrumentation and control systems listed in Table 2.4-3 located in a mild environment withstand design-basis mild environmental conditions without loss of safety-related functions.
>
> The Class 1E digital equipment listed in Table 2.4-3 performs its safety-related function when subjected to the design-basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA.

The ITAAC in SDAA Part 8, Table 2.4-1, ITAAC Numbers 02.04.04 and 02.04.05, along with the corresponding discussions in SDAA Part 8, Table 2.4-2, for ITAAC Numbers 02.04.04 and 02.04.05, conform to the standardized ITAAC, design commitments, and associated Tier 2 discussion in the standardized ITAAC guidance. Therefore, the staff finds the ITAAC are sufficient to verify the qualification of the Class 1E computer-based I&C systems for a mild environment and verify the capability of the Class 1E digital equipment to withstand electromagnetic interference, radiofrequency interference, and electrical surge. Based on the above, the staff finds that the ITAAC for equipment qualification of the safety-related digital I&C in SDAA Part 8, Table 2.4-2, ITAAC Numbers 02.04.04 and 02.04.05, comply with 10 CFR 52.47(b)(1).

14.3.5.5 Combined License Information Items

No COL information items are listed in SDAA Part 2, Chapter 1, Table 1.8-2, for this area of review.

14.3.5.6 Conclusion

The staff finds that the SDAA Part 8 design descriptions and ITAAC for the I&C system satisfy the requirements in 10 CFR 52.47(b)(1) and meet the relevant DSRS Section 14.3.5 and SECY-19-0034 acceptance criteria for design content. The NRC staff also finds that the I&C ITAAC and associated discussion in SDAA Part 8, Tables 2.4-1, 2.4-2, 2.4-3, 2.5-1, 2.5-2, 2.5-3, 2.6-1, 2.6-2, and 2.6-3, are acceptable.

14.3.6 Electrical Systems—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.6.1 Introduction

This section reviews ITAAC and design descriptions applicable to electrical systems. The following SDAA Part 8 tables contain the ITAAC applicable to this review area:

- Table 2.4-1, Number 2 and Number 9
- Table 3.14-1, Number 2

14.3.6.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.6.3 Regulatory Basis

In addition to the regulations listed in Section 14.3.1.3 of this SER, the following NRC regulation contains the relevant requirements for this review:

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," as it relates to the applicant establishing a program for qualifying electrical equipment important to safety located in a harsh environment

SRP Section 14.3.6, "Electrical Systems—Inspections, Tests, Analyses, and Acceptance Criteria," provides acceptance criteria and additional guidance for this review area.

14.3.6.4 Technical Evaluation

The staff reviewed the information in SDAA Part 8 related to the electrical power system to ensure, in part, that SDAA Part 8 contains the top-level, most safety-significant design, testing, and performance requirements for SSCs important to safety, consistent with the guidance in SRP Section 14.3. The staff also reviewed the information for conformance with RG 1.68, Appendix A, Section A-1. The ITAAC review documented in this SER section is limited to the ITAAC and the discussion of the ITAAC in SDAA Part 8. The staff reviewed whether meeting the ITAAC verifies that the SDAA Part 8 design commitments are met when the plant is built.

14.3.6.4.1 *Design Descriptions and ITAAC for Electrical Systems*

The staff reviewed the NuScale SDAA to determine whether the applicant established appropriate design commitments for the electrical power system and that they are verified by ITAAC. The applicant-proposed design descriptions and associated ITAAC for the electrical systems include design aspects related to equipment qualification for seismic and harsh environments, as discussed below.

14.3.6.4.1.1 Equipment Qualification for Seismic and Harsh Environment

Consistent with SRP Section 14.3.6, the ITAAC for equipment qualification for seismic and harsh environments should verify that the seismic design requirement of GDC 2, "Design Bases for Protection against Natural Phenomena," and the environmental qualification requirements of 10 CFR 50.49 are met. Specifically, the design description should determine that Class 1E (i.e., safety-related) equipment is seismic Category I and electrical equipment located in a harsh environment is qualified to withstand the harsh environment and perform its function. The staff evaluates the seismic design requirement of GDC 2 in Section 14.3.2 of this SER.

The staff reviewed the design descriptions in SDAA Part 8, Sections 2.4 and 3.14, which address the most safety-significant features for equipment qualification. SDAA Part 8, Sections 2.4 and 3.14, describe the module-specific and common equipment that would be subject to equipment qualification. The staff determined that design descriptions and ITAAC relating to module-specific and common electrical equipment located in a harsh environment adequately describe the top-level, most safety-significant design features that are based on and are consistent with the material.

Section 3.11 of this SER contains the staff's evaluation of SDAA Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," which describes the environmental qualification requirements for electrical and mechanical equipment. In addition, the staff discusses the applicant's approach for conformance to 10 CFR 50.49 pertaining to the environmental qualification of electrical equipment located in a harsh environment and identifies equipment that is within the scope of 10 CFR 50.49.

The staff reviewed ITAAC Numbers 2 and 9 in SDAA Part 8, Table 2.4-1, and ITAAC Number 2 in SDAA Part 8, Table 3.14-1, which verify that the Class 1E equipment located in a harsh environment is qualified and meets the environmental qualification requirements of 10 CFR 50.49. The staff finds that these ITAAC are necessary, sufficient, and meet the requirements of 10 CFR 52.47(b)(1).

14.3.6.5 Combined License Information Items

SDAA Section 14.3 contains no COL information items related to the electrical power system.

14.3.6.6 Conclusion

The staff has reviewed all the relevant ITAAC information applicable to the electrical systems and evaluated its sufficiency based on whether it demonstrates that the as-constructed plant complies with 10 CFR 50.49 and whether it conforms to relevant NRC guidance in SRP Section 14.3.6. The staff finds that the NuScale ITAAC for electrical systems demonstrates that the as-constructed plant complies with 10 CFR 50.49 and satisfies SRP Section 14.3.6. Therefore, the staff finds that the relevant ITAAC satisfy 10 CFR 52.47(b)(1). The staff concludes that the SDAA design descriptions contain the top-level, most safety-significant design features for the electrical system, consistent with SRP Section 14.3.6 and SECY-19-0034.

14.3.7 Plant Systems—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.7.1 Introduction

This section reviews the ITAAC in NuScale US460 plant systems presented in SDAA Part 8. The following SDAA Part 8 tables contain the ITAAC applicable to this review area:

- Table 2.3-1, "Containment Evacuation System ITAAC," Numbers 1–2
- Table 3.1-1, "Control Room Habitability System ITAAC," Numbers 1, 4, and 5
- Table 3.2-1, "Normal Control Room Heating Ventilation and Air Conditioning ITAAC," Numbers 1–3
- Table 3.3-1, "Reactor Building Heating Ventilation and Air Conditioning System ITAAC," Numbers 1–3
- Table 3.4-1, "Fuel Handling Equipment System ITAAC," Numbers 1–5
- Table 3.5-1, "Fuel Storage System ITAAC," Number 2
- Table 3.6-1, "Ultimate Heat Sink Piping System ITAAC
- Table 3.7-1, "Fire Protection System ITAAC," Numbers 1–4
- Table 3.10-1, "Overhead Heavy-Load Handling System ITAAC," Numbers 1–4
- Table 3.11-1, "Reactor Building ITAAC," Numbers 1–3
- Table 3.13-1, "Control Building ITAAC," Numbers 1-3

14.3.7.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.7.3 Regulatory Basis

See Section 14.3.1.3 of this SER. SRP Section 14.3.7, "Plant Systems—Inspections, Tests, Analyses, and Acceptance Criteria," provides acceptance criteria and additional guidance for this review area.

14.3.7.4 Technical Evaluation

Based on the staff's review of the information in SDAA Part 8 and the NuScale US460 FSAR, the staff draws the following overall conclusions regarding the plant systems information in the SDAA application. Consistent with SRP Section 14.3.7 and SECY-19-0034, the design descriptions found in SDAA Part 8 and ITAAC adequately describe the top-level design features and performance characteristics that are significant to safety. The staff reviewed the design description and system ITAAC to confirm completeness and consistency with the system design-basis, as described in various FSAR sections, and concludes the design description and ITAAC are based on and consistent with those in the FSAR. Tests, analyses, and acceptance criteria associated with each design commitment, when taken together, are sufficient to provide reasonable assurance that the final as-built system fulfills NRC requirements. These ITAAC, along with the corresponding discussions in the FSAR, generally conform to the standardized SDAA ITAAC, design commitments, and associated discussions in the standardized ITAAC guidance.

The requirements of 10 CFR 52.47(b)(1) are met, in part, by identifying ITAAC to verify the top-level design features of the plant systems in the SDAA.

The staff's review of the plant systems' ITAAC is presented below.

14.3.7.4.1 *Internal Flood Protection for Onsite Equipment Failures (SDAA Part 2, Section 3.4.1)*

The ITAAC associated with internal flooding barriers in the RXB and CRB are found in SDAA Part 8, Table 3.11-1, ITAAC Number 2, and Table 3.13-1, ITAAC Number 2, respectively.

The staff reviewed the proposed ITAAC and finds that they are acceptable because they will confirm that the as-built plant systems have the design characteristics stated in the design description and thus verify the flood protection features assumed in the plant's internal flood analysis. Therefore, these ITAAC are consistent with the guidance found in the SRP and meet the requirements of 10 CFR 52.47(b)(1).

14.3.7.4.2 *Internally Generated Missiles (Outside Containment) (SDAA Part 2, Section 3.5.1.1)*

In SDAA Part 2, Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," the applicant reviewed the RXB and CRB to determine what missile could be generated based on the plant equipment and processes. Based on its review, the applicant determined that, due to the plant and system design, there are no credible missiles that could affect SSCs important to safety. Upon reviewing SDAA Part 2, Section 3.5.1, the staff agrees with the applicant's assessment; therefore, the staff finds that no ITAAC are necessary to address the missiles evaluated in SDAA Part 2, Section 3.5.1.1. Turbine generator missiles are evaluated in SDAA Part 2, Section 3.5.1.3, "Turbine Missiles," and addressed in Section 14.3.7.4.4 of this SER.

14.3.7.4.3 *Internally Generated Missiles (Inside Containment) (SDAA Part 2, Section 3.5.1.2)*

As described in SDAA Part 2, Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)," the NPMs use a steel containment that encapsulates the reactor pressure vessel. The applicant stated, in SDAA Part 2, Section 3.5.1.2, that there is no rotating equipment inside containment and all pressurized components are ASME BPV Code Class 1 or 2 and therefore not credible missile sources. In its review in Section 3.5.1.2 of this SER, the staff concluded that there are no credible missiles inside containment. Therefore, the staff finds that no ITAAC are necessary to address such missiles.

14.3.7.4.4 *Turbine Missiles (SDAA Part 2, Section 3.5.1.3)*

SDAA Part 2, Section 3.5.1.3, describes an approach that includes building wall barriers, system redundancy, and defense-in-depth features to protect essential SSCs from turbine missiles. These essential SSCs are in the RXB. SDAA Part 2, Section 3.5.1.3, also states that "[e]ssential SSC within the RXB are protected from turbine missile penetration by the RXB exterior wall." The staff agrees with the assessment that the combined effect of these features provides reasonable assurance of protection to the essential SSCs in the RXB.

SDAA Part 8, Table 3.11-1, ITAAC Number 6, verifies RXB structural integrity under design-basis loads. The staff finds that these ITAAC are sufficient to verify that the RXB has been designed and constructed to withstand turbine missile loads without loss of overall structural integrity. Section 14.3.2 of this SER evaluates these ITAAC.

14.3.7.4.5 *Missiles Generated by Tornados and Extreme Winds (SDAA Part 2, Section 3.5.1.4)*

SDAA Part 8, Tables 3.11-1 and 3.13-1, address verification that the RXB and CRB have been designed and constructed to withstand the effects of natural phenomena, including missiles from hurricanes, tornados, and extreme winds. SDAA Part 8, Table 3.11-1, ITAAC Number 6, verifies RXB structural integrity under design-basis loads. SDAA Part 2, Tier 1, Table 3.13-1, ITAAC Number 4, verifies the CRB structural integrity under design loads. Therefore, the staff finds that these ITAAC address verification that the RXB and CRB have been designed and constructed to withstand missiles from hurricanes, tornados, and extreme winds. Section 14.3.2 of this SER evaluates these ITAAC.

14.3.7.4.6 *Structures, Systems, and Components to Be Protected from External Missiles (SDAA Part 2, Section 3.5.2)*

In SDAA Part 2, Section 3.5.2, "Structures, Systems, and Components to Be Protected from External Missiles," the applicant stated that all safety-related and risk significant SSCs that must be protected from external missiles are located inside the seismic Category I RXB and seismic Category I portions of the CRB. In its review of the information in SDAA Part 8, the staff found that ITAAC Number 6, Table 3.11-1, verifies RXB structural integrity under design-basis loads, and ITAAC Number 4, Table 3.13-1, verifies structural integrity under design loads in the CRB. Therefore, the staff finds that these ITAAC address verification that the RXB and CRB have been designed and constructed to withstand the effects of natural phenomena, including missiles from hurricanes, tornados, and extreme winds. Section 14.3.2 of this SER evaluates these ITAAC.

14.3.7.4.7 *Plant Design for Protection against Postulated Piping Failure in Fluid Systems (SDAA Part 2, Section 3.6.1)*

SDAA Part 8, Section 2.1, "Nuclear Power Module," identifies a design commitment to ensure safety-related SSCs are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems. ITAAC Number 4 in SDAA Part 8, Table 2.1-1, requires an inspection and analysis of the as-built high- and moderate-energy piping systems and protective features for the safety-related SSCs to ensure that they are installed in accordance with the as-built pipe break hazard analysis report and that safety-related SSCs are protected against, or are qualified to withstand, the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems. The staff evaluates this ITAAC in Section 14.3.3.4.2.2 of this SER.

14.3.7.4.8 *Reactor Coolant Pressure Boundary Leakage Detection (SDAA Part 2, Section 5.2.5)*

SDAA Part 8, Table 2.3-1, includes ITAAC Numbers 1 and 2 for RCS leakage detection. The ITAAC require tests to verify the design of the RCS leakage detection systems. These tests include (1) verifying that the containment evacuation system (CES) detects a level increase in the CES sample tank, which correlates to a detection of an unidentified RCS leakage rate of 3.79 liters per minute (lpm) (1 gallon per minute (gpm)) within 1 hour, and (2) verifying the CES inlet pressure instrumentation detects a pressure increase, which correlates to a detection of an unidentified RCS leakage rate of 3.79 lpm (1 gpm) within 1 hour. This is consistent with the guidance in RG 1.45 and SRP Section 5.2.5, “Reactor Coolant Pressure Boundary Leakage Detection.”

The staff reviewed the proposed ITAAC and finds that they are consistent with NRC guidance and meet the requirements of 10 CFR 52.47(b)(1).

14.3.7.4.9 *New and Spent Fuel Storage (SDAA Part 2, Section 9.1.2)*

The staff reviewed SDAA Part 8, Section 3.5, “Fuel Storage System,” which contains the specific ITAAC for the fuel storage system. It describes the top-level features of the fuel storage system design; specifies that the fuel storage racks will maintain the k-effective (k_{eff}) in accordance with the limits in 10 CFR 50.68, “Criticality accident requirements”; and, in Table 3.5-1, specifies the ITAAC for the fuel storage racks.

The ITAAC related to criticality safety of new and spent fuel storage and handling in SDAA Part 8, Table 3.5-1, ITAAC Number 2, includes a design commitment that the fuel storage racks will meet the portion of 10 CFR 50.68(b)(4) applicable when soluble boron is credited. An inspection of the as-built fuel storage racks, their configuration in the spent fuel pool (SFP), and the associated documentation will ensure that the as-built configuration conforms to the design values and their tolerances used in the approved criticality analysis. Furthermore, this ITAAC is consistent with the standardized ITAAC guidance. For these reasons, this ITAAC is acceptable for verifying criticality safety of new and spent fuel storage and meets 10 CFR 52.47(b)(1).

The SFP is part of the ultimate heat sink (UHS); Section 14.3.7.4.13 of this SER evaluates drain-down prevention.

14.3.7.4.10 *Spent Fuel Pool Cooling and Cleanup System (SDAA Part 2, Section 9.1.3).*

The SFP cooling and cleanup system is not safety related and is not credited for mitigation of any design-basis events. When the PCWS is unavailable to perform its active function, the pool is cooled by passive means using the volume of water in the combined reactor pool, refueling pool, and SFP. ITAAC Number 1 in SDAA Part 8, Table 3.6-1, for the UHS addresses verification that sufficient cooling water is available for design-basis events. Section 14.3.7.4.13 of this SER discusses this ITAAC.

14.3.7.4.11 *Fuel Handling Equipment (SDAA Part 2, Section 9.1.4)*

SDAA Part 8, Section 3.4, provides a general overview of the fuel handling equipment (FHE) system and the associated ITAAC. The FHE system ITAAC are provided to meet the requirements of 10 CFR 52.47(b)(1) by ensuring that the as-built system complies with the approved system design described in the SDAA FSAR. SDAA Part 8, Table 3.4-1, ITAAC Numbers 1–5, present the FHE system ITAAC.

The staff reviewed the proposed ITAAC and finds them acceptable because they will verify that the FHE has been constructed in accordance with ASME NOG-1, “Rules for Construction of Overhead and Gantry Cranes,” or ASME NUM-1, “Rules for Construction of Cranes, Monorails, and Hoists (with Bridge or Trolley or Hoist of the Underhung Type)” code and will have sufficient load-carrying capability and limits on travel to ensure that it has been constructed and will be operated in conformity with the SDAA.

The plant should be designed with appropriate radiation protection design features during potential accident conditions, in accordance with GDC 61, “Fuel Storage and Handling and Radioactivity Control.” In SDAA Part 8, Table 3.4-1, the design commitment and ITAAC Number 4 require that fuel-handling machine travel be limited so that the machine maintains at least 3 meters (10 feet) of water above the top of the fuel assembly when lifted to its maximum height, with the pool level at the lower limit of the normal operating low water level. This ITAAC will ensure that personnel are not overexposed from a raised spent fuel assembly, and a design feature is provided for maintaining a dose of less than 2.5 millirem per hour radiation exposure to operators on the refueling platform, in accordance with the American National Standards Institute/American Nuclear Society-57.1-1992, “Design Requirements for Light Water Reactor Fuel Handling Systems.” This is in accordance with GDC 61 and is acceptable.

Based on the above evaluation, the staff finds that the ITAAC are consistent with the guidance found in the SRP and meet the requirements of 10 CFR 52.47(b)(1).

14.3.7.4.12 *Overhead Heavy-Load Handling Systems (SDAA Part 2, Section 9.1.5)*

SDAA Part 8, Section 3.10, provides a general overview of the reactor building crane (RBC) and the associated ITAAC. SDAA Part 8, Table 3.10-1, ITAAC Numbers 1–3, include the RBC ITAAC.

SDAA Part 8 information should include the features and functions that could have a significant effect on the safety of a nuclear plant or that are important in preventing or mitigating accidents. A drop of the NPM, a spent fuel cask, or other components of similar size could affect plant safety. Therefore, design features that reduce the risk, or analyses that provide assurance of plant safety in the event of a dropped load, are of safety importance. The staff considers single-failure-proof design criteria for the overhead heavy-load handling systems equipment to be a significant design feature to include in SDAA Part 8. SDAA Part 8, Table 3.10-1, provides ITAAC Number 1 for verification that the RBC main hoist and lower block assembly, sister hook, articulating traveling jib crane, dry dock jib crane, two auxiliary hoists, and wet hoist, respectively, contain single-failure-proof design features. ITAAC Number 2 provides for a load test of at least 125 percent of the hoist rated capacity for the listed hoists. Also, ITAAC Number 3 provides for nondestructive examinations of welds on the load-carrying path for these hoists. The staff finds these ITAAC acceptable because they are consistent with the provisions of ASME NOG-1 for a Type I crane or ASME NUM-1 for a Type IA crane.

Based on the above evaluation, the staff finds that the overhead heavy-load handling systems ITAAC are consistent with SRP guidance and meet the requirements of 10 CFR 52.47(b)(1).

14.3.7.4.13 *Demineralized Water System (SDAA Part 2, Section 9.2.3)*

The demineralized water system (DWS) is a non-safety-related system and is not required for mitigation of any design-basis event. While DWS operation is not required or credited in any design-basis event, in its review of the DWS, the staff noticed that, because the DWS isolation valves limit or prevent boron dilution of the reactor coolant, the DWS isolation valves perform a safety-related function. However, for the NuScale design, demineralized water isolation valves are included as part of the CVCS. Design and operation of the DWS isolation valve is covered by the ITAAC in SDAA Part 8, Section 2.2, “Chemical and Volume Control System.” SDAA Part 8, Table 2.2-1, ITAAC Numbers 1 and 2, verify proper operation of the demineralized water isolation valves. Section 14.3.3 of this SER evaluates these ITAAC. Other than the function identified above, the system is not safety-related or risk significant, and the applicant did not credit it for providing a safety-significant function; therefore, the staff concluded that no additional ITAAC are necessary.

14.3.7.4.14 *Ultimate Heat Sink (SDAA Part 2, Section 9.2.5)*

SDAA Part 8, Table 3.6-1, provides ITAAC for UHS piping and connections. The SFP, refueling pool, reactor pool, and dry dock piping and connections are located to prevent the drain down of the SFP water level below the minimum safety water level.

SDAA Part 8, Table 3.6-2, specifies the ITAAC for the UHS. ITAAC Number 1 contains a design commitment that the SFP, refueling pool, reactor pool, and dry dock piping and connections are located to prevent drain down of the SFP and reactor pool water below the minimum safety water level.

The staff reviewed the proposed ITAAC and finds that an inspection will be performed as part of the ITAAC that will confirm that the as-built plant systems meet the design commitment regarding the prevention of drain down of the SFP. For this reason, this ITAAC is acceptable for SFP drain down. The staff finds that the ITAAC is consistent with the SRP guidance and meets the requirements in 10 CFR 52.47(b)(1).

14.3.7.4.15 *Equipment and Floor Drain Systems (SDAA Part 2, Section 9.3.3)*

In SDAA Part 8, Table 3.9-3, “Radiation Monitoring—Shared System Automatic Actions,” ITAAC Number 6 verifies that, upon initiation of a high-radiation signal, the balance-of-plant drain system automatically aligns or actuates the identified components to the positions identified in SDAA Part 8, Table 3.9-3. SER Section 14.3.8 evaluates these ITAAC.

14.3.7.4.16 *Fire Protection System (SDAA Part 2, Section 9.5.1)*

The fire protection system (FPS) performs the following non-safety-related system functions that are verified by ITAAC:

- The FPS supports the RXB by providing fire prevention, detection, and suppression.
- The FPS supports the RWB by providing fire prevention, detection, and suppression.

- The FPS supports the CRB by providing fire prevention, detection, and suppression.

In SDAA Part 8, Table 3.7-1, the applicant identified and described the following ITAAC related to the fire protection program:

- ITAAC Number 1: verifies that two separate firewater storage tanks provide a dedicated volume of water for firefighting.
- ITAAC Number 2: verifies that the FPS has a sufficient number of fire pumps to provide the design flow requirements to satisfy the flow demand for the largest sprinkler or deluge system, plus an additional 1,900 lpm (500 gpm) for fire hoses, assuming failure of the largest fire pump or loss of offsite power.
- ITAAC Number 3: verifies that safe shutdown can be achieved assuming that all equipment in any one fire area (except for the MCR and under-the-bioshield) is rendered inoperable by fire damage and that reentry into the fire area for repairs and operator actions is not possible. An alternative shutdown capability that is physically and electrically independent of the MCR exists. Additionally, smoke, hot gases, or fire suppressant cannot migrate from the affected fire area into other fire areas to the extent that they could adversely affect safe shutdown capabilities, including operator actions.
- ITAAC Number 4: verifies that a plant fire hazards analysis considers potential fire hazards and ensures the fire protection features in each fire area are suitable for the hazards.

In SDAA Part 8, Table 3.11-1, ITAAC Number 1 verifies that fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the RXB fire area of origin.

In SDAA Part 8, Table 3.13-1, ITAAC Number 1 verifies that fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the CRB fire area of origin.

The staff finds that the ITAAC are sufficient to demonstrate that the FPS can perform the non-safety-related functions identified above. Based on a graded approach commensurate with the safety significance of the FPS, the staff reviewed the proposed ITAAC and finds that they are consistent with the SRP guidance and meet the regulations contained in 10 CFR 52.47(b)(1).

14.3.7.4.17 *Main Steam Supply System (SDAA Part 2, Section 10.3)*

SDAA Part 8, Table 2.4-3, presents the ITAAC for the SSCs of the main steam system (MSS) as ITAAC Numbers 1, 2, 3, 6, 7, and 9.

The staff's review for the MSS information included descriptive information; safety-related functions; mechanical, I&C, and electric power design features; and environmental qualification, as well as system and equipment performance requirements. The staff reviews ITAAC Numbers 1, 3, 6, and 7 in Section 14.3.3 of this SER and ITAAC Numbers 2 and 9 in Section 14.3.6 of this SER.

The staff finds that the ITAAC presented in the above-listed sections are consistent with the guidance found in the SRP and meet the regulations contained in 10 CFR 52.47(b)(1).

14.3.7.4.18 *Condensate and Feedwater System (SDAA Part 2, Section 10.4.6)*

There are no ITAAC for the entire condensate and feedwater system shown in SDAA Part 8; however, in SDAA Part 8, Section 2.4, the applicant proposed ITAAC for the following condensate and feedwater system equipment: the feedwater supply check valves, the feedwater isolation valve, and the feedwater regulating valve. SDAA Part 8, Table 2.4-2, provides ITAAC Number 6 for testing and accepting these valves. The staff reviews this ITAAC in Section 14.3.3 of this SER. The staff finds that it provides reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the NuScale Power Plant US460 standard design has been built and will be operated in accordance with the applicable portions of the SDAA, the AEA, and the NRC's rules and regulations, as required by 10 CFR 52.47(b)(1).

14.3.7.4.19 *Control Room Habitability System (SDAA Part 2, Section 6.4)*

The staff reviewed the following ITAAC requirements in SDAA Part 8, Table 3.1-1, ITAAC Numbers 1, 4, and 5:

- ITAAC Number 1 on control room envelope (CRE) air exfiltration test: Tracer gas testing will be performed to verify the CRE leakage rate assumed in the radiation dose analysis is not exceeded.

- ITAAC Number 4 on CRE heat sink temperature: Analysis will be performed to show the CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a design-basis accident.

- ITAAC Number 5 on CRHS positive pressure: A test will be performed to verify that the CRHS maintains a positive pressure in the MCR relative to adjacent areas while in design-basis accident alignment.

The staff finds that these ITAAC are sufficient to demonstrate that the CRHS can provide clean breathing air to the control room, maintain a positive control room pressure, and maintain the temperature of the CRE within an acceptable range, as described in FSAR Section 6.4. The staff reviewed these proposed ITAAC and finds that they are consistent with SRP Section 14.3.7. Therefore, the ITAAC are acceptable for complying with the requirements of 10 CFR 52.47(b)(1).

Section 14.3.3 of this SER evaluates ITAAC Numbers 2 and 3 in SDAA Part 2, Table 3.1-1.

14.3.7.4.20 *Normal Control Room Heating, Ventilation, and Air Conditioning System (SDAA Part 2, Section 9.4.1)*

The staff reviewed the following ITAAC in SDAA Part 8, Table 3.2-1:

- ITAAC Number 1: tests that the control room heating, ventilation, and air conditioning (HVAC) system (CRVS) air-operated CRE isolation dampers perform their function to fail to the closed position on loss of motive power under design-basis conditions.

- ITAAC Number 2: tests and verifies that the CRVS maintains a positive pressure in the CRB relative to the outside environment.

- ITAAC Number 3: verifies that the hydrogen concentration levels in the CRB battery rooms are below 1 percent by volume. This is consistent with IEEE Std. 484-2002, “IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications,” as revised by RG 1.128, Revision 2, “Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants,” issued February 2007, which states, “the ventilation system shall limit hydrogen accumulation to one percent of the total volume of the battery area.”

The staff finds that the ITAAC conform to the guidance for ITAAC verifications in RG 1.206, as applied to the CRVS and, therefore, finds the ITAAC acceptable for complying with the requirements of 10 CFR 52.47(b)(1).

14.3.7.4.21 *Reactor Building Heating, Ventilation, and Air Conditioning System (SDAA Part 2, Section 9.4.2)*

The staff reviewed the applicant’s proposed ITAAC for the reactor building heating, ventilation, and air conditioning system (RBVS) in SDAA Part 8, Table 3.3-1:

- ITAAC Number 1: tests to verify that the RBVS maintains a negative pressure in the RXB relative to the outside environment.

- ITAAC Number 2: tests to verify that the RBVS maintains a negative pressure in the RWB relative to the outside environment.

- ITAAC Number 3: tests to verify that the RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below 1 percent by volume.

The staff finds the acceptance criteria for these three ITAAC conform to the guidance for ITAAC verifications in RG 1.206 as applied to the RBVS. The staff also reviewed the radiation protection aspects of ITAAC Numbers 1 and 2. SDAA Part 8, Section 3.3, “Reactor Building Heating Ventilation and Air Conditioning System,” provides design commitments and ITAAC specifying that the RXB and RWB ventilation systems will maintain the buildings at a negative pressure relative to the outside air to control airborne activity so that releases of airborne radioactivity from the buildings are minimized. The staff evaluated the information provided by the applicant and finds that the design commitments and ITAAC Numbers 1 and 2 in SDAA Part 8, Table 3.3-1, to be in accordance with SRP Section 14.3.8, “Radiation Protection—Inspections, Tests, Analyses, and Inspection Criteria,” in that the applicant provided ITAAC associated with controlling the release of radioactive material to the public.

Therefore, the staff finds the ITAAC requirements acceptable for complying with the requirements of 10 CFR 52.47(b)(1).

14.3.7.4.22 *Radioactive Waste Building Ventilation System (SDAA Part 2, Section 9.4.3)*

SDAA Part 8, Section 3.3, Table 3.3-1, includes ITAAC Number 2 that addresses verification of the capability of the RBVS to maintain a negative pressure in the RWB relative to the outside environment. The staff finds this to be acceptable for the RXB HVAC system as discussed above.

14.3.7.4.23 *Systems Not Requiring ITAAC*

In SDAA Part 2, Section 9.2.1, the applicant indicated that the NuScale US460 Power Plant design does not have a service water system. Therefore, there are no proposed ITAAC for this system, and the staff finds that no ITAAC are necessary.

The staff reviewed the following systems and found that they are not safety-related and do not perform any safety-related, risk significant, or safety-significant functions. Therefore, the staff finds that no ITAAC are necessary for these systems:

- reactor component cooling water system (SDAA Part 2, Section 9.2.2)
- potable and sanitary water systems (SDAA Part 2, Section 9.2.4)
- condensate storage facilities (SDAA Part 2, Section 9.2.6)
- site cooling water system (SDAA Part 2, Section 9.2.7)
- chilled water system (SDAA Part 2, Section 9.2.8)
- utility water system (SDAA Part 2, Section 9.2.9)
- compressed air systems (SDAA Part 2, Section 9.3.1)
- turbine building ventilation system (SDAA Part 2, Section 9.4.4)
- turbine generator (SDAA Part 2, Section 10.2)
- air cooled condenser (SDAA Part 2, Section 10.4.1)
- condenser air removal system (SDAA Part 2, Section 10.4.2)
- turbine gland sealing system (SDAA Part 2, Section 10.4.3)
- turbine bypass system (SDAA Part 2, Section 10.4.4)
- auxiliary boiler system (SDAA Part 2, Section 10.4.7)

14.3.7.5 Combined License Information Items

No COL information items are listed in FSAR Table 1.8-2 for this area of review.

14.3.7.6 Conclusion

The staff concludes that, if the ITAAC for the matters reviewed in this section are performed and the acceptance criteria met, there is reasonable assurance the relevant portions of the NuScale standard design nuclear power plant has been constructed and will be operated in accordance with the design, the AEA, and NRC rules and regulations in compliance with 10 CFR 52.47(b)(1). The staff also concludes that the applicant has included sufficient top-level design information, consistent with SECY-19-0034, and that SDAA Part 2 is consistent with the SDAA Part 8 information.

14.3.8 Radiation Protection—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.8.1 Introduction

This section reviews ITAAC applicable to radiation protection. The following SDAA Part 8 tables contain the ITAAC applicable to this review area:

- Table 2.7-1, “Radiation Monitoring—Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 1–4
- Table 3.3-1, “Reactor Building HVAC System Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 1–2
- Table 3.4-1, “Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 4–5
- Table 3.9-1, “Radiation Monitoring – Shared Systems Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 1–11
- Table 3.11-1, “Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria,” Number 4
- Table 3.12-1, “Radioactive Waste Building Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 1–3
- Table 3.14-1, “Equipment Qualification—Shared Equipment Inspections, Tests, Analyses, and Acceptance Criteria,” Number 3

14.3.8.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.8.3 Regulatory Basis

In addition to the regulations listed in Section 14.3.1.3 of this SER, the following NRC regulations contain the relevant requirements for this review:

- GDC 19, “Control Room,” as it relates to the requirement, in part, that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 sievert (5 rem) total effective dose equivalent for the duration of the accident
- GDC 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to the radiation monitors used to initiate mitigating actions to prevent a release of radioactive materials into the environment
- GDC 61, as it relates to the requirement that occupational radiation protection aspects of fuel storage, fuel handling, radioactive waste, and other systems that may contain radioactivity be designed such that they ensure adequate safety during normal and postulated accident conditions, with suitable shielding and appropriate containment and filtering systems

- GDC 63, “Monitoring Fuel and Waste Storage,” as it relates to the requirement, in part, that appropriate systems be provided for the fuel storage and radioactive waste systems and associated handling areas to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels.

- GDC 64, “Monitoring Radioactivity Releases,” as it relates to the requirement that the containment atmosphere, spaces containing components for recirculation of loss-of-coolant-accident fluids, effluent discharge paths, and the plant environs be monitored for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents

- 10 CFR 20.1101, “Radiation protection programs,” as it relates to the requirement that the licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as reasonably achievable (ALARA)

- 10 CFR 20.1201, “Occupational dose limits for adults,” as it relates to the requirement, in part, that with the exception of planned special exposures, the annual occupational dose limit for adults is equal to a total effective dose equivalent of 0.05 sievert (5 rem), or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue, other than the lens of the eye, being equal to 0.5 sievert (50 rem)

- 10 CFR 20.1406, “Minimization of contamination,” as it relates to applicants for SDAs describing in the application how the facility design will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste

- 10 CFR 20.1501, “General,” as it relates to the requirement, in part, that licensees make surveys that are reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, the concentrations or quantities of radioactive material, and the potential radiological hazards

- 10 CFR 20.1701, “Use of process or other engineering controls,” as it relates to the requirement that the applicant use, to the extent practical, process or other engineering controls to control the concentration of radioactive material in air

- 10 CFR 50.34(f)(2)(xvii), as it relates to the requirement, in part, that instrumentation be provided that can measure, record, and read out in the MCR containment radiation intensity (high level)

SRP Sections 14.3.7 and 14.3.8 provide acceptance criteria and additional guidance for this review area.

14.3.8.4 Technical Evaluation

The scope of the radiation protection top-level design and ITAAC review includes the following:

- radiation shielding provided by structures and components

- radiation monitoring systems

- ventilation systems (as they relate to radiation protection design features)
- design features for radiation protection

14.3.8.4.1 *Radiation Shielding*

SRP Section 14.3.8 indicates that the criteria in the ITAAC should ensure that the radiation shielding design (as provided by the plant structures or by permanent or temporary shielding included in the design) is adequate so that the maximum radiation levels in plant areas are commensurate with the areas' access requirements (and the requirements of 10 CFR Part 20, "Standards for Protection Against Radiation"). SRP Section 14.3.8 also specifies that the review should ensure that the application clearly describes the SSCs that provide a significant radiation protection function, including the key performance characteristics and safety functions of SSCs based on their safety significance.

As such, FSAR Section 12.3.2, "Shielding," describes some of the design considerations for radiation shielding, such as stating that concrete will be used for a significant portion of plant shielding. FSAR Section 12.3.2.2, "Design Considerations," states that the selection of shielding materials considers the ambient environment and potential degradation mechanisms. Concrete is used for a significant portion of plant shielding. In addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. FSAR Table 12.3-5, "Reactor Building Shield Wall Geometry," provides the concrete equivalent thickness for some of the walls, floors, ceilings, and other radiation barriers in the RXB. In addition to concrete, the RXB includes borated polyethylene shielding on the bioshield faceplate for neutron shielding. FSAR Table 12.3-6, "Radioactive Waste Building Shield Wall Geometry," provides the concrete equivalent thickness for some of the walls, floors, ceilings, and other radiation barriers in the RWB. The shielding for these barriers must provide equivalent radiation shielding to the thicknesses provided in the tables, as described in FSAR Chapter 12 and Chapter 12 of this SER. FSAR Table 12.3-7, "Radioactive Waste Building Radiation Shield Doors," lists the shielded doors located in the RWB. No shielded doors are specified for the RXB; they are modeled as openings in shielding calculations.

SDAA Part 8, Section 3.11, states that the RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding. SDAA Part 8, Table 3.11-1, contains the ITAAC for the RXB. Specifically, ITAAC Number 4 in SDAA Part 8, Table 3.11-1, verifies that the radiation attenuation capability of the RXB radiation shielding barriers is greater than or equal to the required attenuation capability of the approved design.

SDAA Part 8, Section 3.12, states that the RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding. SDAA Part 8, Table 3.12-1, contains the ITAAC for the RWB. Specifically, ITAAC Number 1 in SDAA Part 8, Table 3.12-1, verifies the radiation shielding capability of the RWB radiation shielding barriers is greater than or equal to the required attenuation capability of the approved design. In addition, ITAAC Number 2 in SDAA Part 8, Table 3.12-1, verifies that radiation attenuating doors for normal operation and for post-accident radiation shielding have a radiation attenuation capability that meets or exceeds that of the shielding provided in FSAR Table 12.3-7.

In addition, SDAA Part 8, Tables 3.11-2 and 3.12-2, provide a cross reference between the ITAAC and the FSAR information. This information specifies that the radiation shielding is provided to meet normal operation and post-accident radiation zone requirements and to ensure compliance with all relevant requirements, including 10 CFR 50.49; GDC 4, “Environmental and Dynamic Effects Design Bases”; Principal Design Criterion 19; GDC 61; 10 CFR 50.34(f)(2)(vii); and equipment survivability requirements for the compartment walls, ceilings, and floors, or other barriers that provide shielding. It also clarifies that an ITAAC inspection is performed of the RXB and RWB radiation barriers to verify materials and thicknesses. Finally, it indicates that attenuation capabilities are determined based on materials and thicknesses, and a report will conclude that attenuation capabilities are greater than or equal to the approved design.

The staff reviewed the SDAA information on radiation shielding barriers and radiation attenuation doors discussed above. The staff determined that the proposed approach is consistent with SECY-19-0034 in that it will allow applicants and licensees to make changes to the shielding barriers prior to and during construction. This reduces the potential for licensees to need to submit a license amendment for changes that are not safety-significant. The ITAAC can then be completed by showing that the radiation attenuation capability is equivalent to what is provided in the FSAR information at the time of ITAAC completion. Since the ITAAC verifies that the radiation attenuation capability is the same as specified in FSAR at the time of ITAAC completion, the staff finds the ITAAC for the shielding barriers and doors and supporting FSAR information to be acceptable.

14.3.8.4.2 *Under-the-Bioshield Radiation Monitors*

This section discusses ITAAC related to the under-the-bioshield radiation level display in the MCR. The staff reviewed SDAA Part 8, Section 2.5, “Module Protection Systems and Safety Display and Indication System,” and ITAAC Number 13 in Table 2.5-1. The design commitment for this ITAAC states, “The PAM Type B and Type C displays are indicated on the SDIS displays in the MCR,” and the ITAAC acceptance criterion is, “The PAM Type B and Type C displays listed in FSAR Table 7.1-7 are retrieved and displayed on the SDIS displays in the MCR.” Since the under-the-bioshield monitors are post-accident monitoring (PAM) Type B and Type C variables, this ITAAC verifies that the under-the-bioshield-area radiation monitor is displayed on the SDIS in the MCR. The staff evaluated this information and concluded that the SDAA includes an appropriate ITAAC for the under-the-bioshield radiation monitors. It is consistent with SRP Section 14.3.8 to include ITAAC that provide assurance that the radiation monitors respond and appropriately actuate components to mitigate an unexpected release of radioactive material. As a result, the staff finds these ITAAC to be acceptable. Section 14.3.5 of this SER also discusses this ITAAC.

14.3.8.4.3 *Radioactive Waste Systems and Radiation Effluent Monitoring*

The areas of review for radioactive waste systems include design objectives, design criteria, identification of all expected releases of radioactive effluents, methods of treatment, methods used in calculating effluent source terms and releases of radioactive materials in the environment, and operational programs in controlling and monitoring effluent releases and for assessing associated doses to members of the public. The radioactive waste systems include the liquid radioactive waste system, gaseous radioactive waste system, and solid radioactive waste system. These systems deal with the management of radioactive wastes, as liquid, wet, and dry solids, produced during normal operation and anticipated operational occurrences. SER Sections 11.2, 11.3, and 11.4, respectively, provide the staff’s review of these systems. In addition, the reviews include an evaluation of the process and effluent radiological monitoring instrumentation and sampling systems (PERMISS), which are used to monitor liquid and gaseous process streams and effluents and solid wastes generated by these systems. The

PERMISS include subsystems used to collect process and effluent samples during normal operation, anticipated operational occurrences, and post-accident conditions. Section 11.5 of this SER contains the staff's review of the PERMISS.

SDAA Part 8, Section 2.7, contains the design commitments and ITAAC related to the PERMISS for the automatic actions of various systems based on radiation monitoring that are module specific. These design commitments and ITAAC require the CES, CVCS, auxiliary boiler system (ABS), and MSS monitors to automatically respond to high-radiation signals and perform the necessary actions. The staff's review determined that the design commitments and ITAAC are acceptable because the ITAAC test the functions of the CES, CVCS, ABS, and MSS monitors, as described in SDAA Part 8, Table 2.7-3, "Radiation Monitoring—Module-Specific Automatic Actions," to initiate the desired actions on high-radiation signals to demonstrate the monitors' ability to mitigate radioactive releases, as required by the design commitments.

SDAA Part 8, Section 3.9, contains the design commitments and ITAAC related to the PERMISS for the automatic actions of various systems, based on radiation monitoring, that are shared among the 6 NPMs. These design commitments and ITAAC require that the radiation monitors associated with the normal CRVS, CRHS, RBVS, gaseous radioactive waste system, containment flooding and drain system, balance-of-plant drain system, liquid radioactive waste system, ABS, DWS, radioactive waste drain system, and site cooling water system automatically respond to high-radiation signals and perform the necessary actions. The staff's review determined that the design commitments and ITAAC are acceptable because the ITAAC test the functions of the systems as described in SDAA Part 8, Table 3.9-1, to initiate the desired actions on high-radiation signals to demonstrate the monitors' ability to mitigate radioactive releases, as required by the design commitments.

In addition to the ITAAC documented above, the staff reviewed information related to CES monitoring in relation to the ITAAC in SDAA Part 8, Section 2.3, "Containment Evacuation System Inspections, Tests, Analyses, and Acceptance Criteria." The staff found that the ITAAC related to the test for the RCS pressure boundary leakage did not include a test for the CES radiation monitor. FSAR Section 5.2.5.1, "Leakage Detection and Monitoring," identifies three methods for identifying RCS pressure boundary leakage: (1) CNV pressure monitoring, (2) CES sample tank level, (3) CES vacuum pump discharge process radiation monitor. In addition, the staff reviewed the information in the TS as it relates to TS 3.4.7 for RCS leakage detection. This TS relates to the test for RCS pressure boundary leakage ITAAC because the ITAAC verifies that a NuScale plant is capable of detecting the leakage described in the TS. The staff observed that the pressure and level methods included two channels provided for each of these methods. In addition, the conditions described by the TS require actions to verify amounts of RCS leakage when one or more of the channel indicators is inoperable. When one of the leakage detection methods has all channels inoperable, these methods must be restored. Therefore, there are multiple pressure and level channels available to detect RCS leakage in the TS and ITAAC and the radiation monitor for detecting RCS leakage in the TS. Since the radiation monitor for detecting RCS leakage is credited in the TS and discussed in the FSAR and since the ITAAC adequately address the other methods for detecting leakage, the staff determined that including an additional ITAAC for the CES radiation monitor was unnecessary.

Based on the discussion above, the staff finds that the information provided in SDAA Sections 2.7, 3.9, and 3.12 is complete and consistent with the plant design basis as described in FSAR Section 11.2, "Liquid Waste Management System"; Section 11.3, "Gaseous Waste Management System"; Section 11.4, "Solid Waste Management System"; and Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System.” Further, the staff finds that the ITAAC for the PERMISS are acceptable and comply with the requirements of 10 CFR 52.47(b)(1).

SDAA Part 8, Section 3.14, Table 3.12-1, ITAAC Number 3, contains an ITAAC ensuring that the below-grade portions of the RWB and the above-grade portions used for storage or processing of radioactive waste will be designed as RW-IIa in accordance with RG 1.143, Revision 2. As discussed in FSAR Chapters 3 and 11, this is consistent with the RWB design; therefore, it is appropriate to include an ITAAC for the RWB that verifies that the as-built RWB maintains its designated structural integrity under the design-basis loads. In addition, SDAA Part 8, Section 3.14, indicates that the RW-IIa components and piping used for processing gaseous radioactive waste listed in SDAA Part 8, Table 3.14-3, are constructed to the standards of RW-IIa. SDAA Part 8, Table 3.14-3, lists the degasifiers, degasifier condensers, degasifier vacuum pumps, guard beds, decay beds and valves associated with the guard and decay beds as being designed to RW-IIa. ITAAC Number 3 in SDAA Part 8, Table 3.14-1, requires a report demonstrating that the as-built RW-IIa components associated with processing gaseous radioactive waste (i.e., the degasifiers, degasifier condensers, degasifier vacuum pumps, guard beds, decay beds and valves associated with the guard and decay beds) meet the RW-IIa design criteria. SDAA Part 8, Table 14.3-2, provides more detail regarding the basis and scope of the ITAAC. It specifies that the scope of the ITAAC is RW-IIa components associated with processing gaseous radioactive waste.

The staff evaluated the information provided and determined that it was acceptable to only include ITAAC for the specified components and piping because, in the event of a structural failure of radioactive waste components, these gaseous radwaste system components are the radioactive waste components that the staff determined were most likely to result in a significant radiological release to the public and potential uncontrolled occupational dose. The staff determined these components were the most radiologically significant because (1) these components were classified as RW-IIa (due to their high radionuclide content), and (2) failure of these components would be most likely to result in an uncontained release.

The staff also considered the need for ITAAC for other radioactive waste system components and piping. The staff determined that, because of the lower radionuclide content of RW-IIc components, ITAAC for those components were not necessary. The staff determined that, while some components like the spent resin storage tanks (RW-IIa) and phase separator tanks (RW-IIb) contained higher quantities of radioactive material, the potential for an uncontrolled release from those components is low because these components contained slurry or liquid waste and were located underground in the RWB, in their own individual cubicles, which are stainless-steel lined up to a cubicle wall height equivalent to the full tank volume. Therefore, even if these components failed, the staff determined that radioactive material would be contained mostly within the cubicle where it could be appropriately handled by radiation protection personnel. As a result, while the FSAR specifies that all the radioactive waste SSCs are designed in accordance with RG 1.143, the staff determined that the only items requiring ITAAC were those associated with the potential for significant gaseous radioactive waste releases, as described above. As a result, the staff finds these ITAAC to be acceptable.

14.3.8.5 Combined License Information Items

No COL information items are associated with this section.

14.3.8.6 Conclusion

The applicant provided SDAA design information and ITAAC for radiation protection SSCs, which it credited for demonstrating that a plant incorporating the NuScale SDAA satisfies the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 52. The staff concludes that if the inspections, tests, and analyses for the matters reviewed in this section are performed and the acceptance criteria met, there is reasonable assurance the relevant portions of the NuScale standard design nuclear power plant have been constructed and will be operated in accordance with the design, the AEA, and NRC rules and regulations in compliance with 10 CFR 52.47(b)(1).

14.3.9 Human Factors Engineering—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.9.1 Introduction

This section reviews ITAAC applicable to human factors engineering (HFE). The following SDAA Part 8 tables contain the ITAAC applicable to this review area:

- Table 3.15-1, “Human Factors Engineering ITAAC,” ITAAC No. 03.15.01
- Table 3.15-1, “Human Factors Engineering ITAAC,” ITAAC No. 03.15.02

14.3.9.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.9.3 Regulatory Basis

See Section 14.3.1.3 of this SER. SRP Section 14.3.9, “Human Factors Engineering—Inspections, Tests, Analyses, and Acceptance Criteria,” provides acceptance criteria and additional guidance for this review area.

14.3.9.4 Technical Evaluation

The staff used the guidance in SRP Section 14.3.9 to review SDAA Part 8, Section 3.15, “Human Factors Engineering,” and the information related to the HFE design process.

SDAA Part 8, Section 3.15, includes two ITAAC for HFE: ITAAC 03.15.01 and ITAAC 03.15.02. The applicant’s HFE ITAAC are similar to the format and content of the HFE ITAAC in the draft standardized ITAAC discussed in RG 1.206 (ML16097A123); however, the HFE ITAAC have been modified to address unique aspects of the NuScale application. ITAAC 03.15.01, which corresponds to standardized ITAAC No. H02, includes the applicant’s design implementation (DI) process, which is used to verify that the final MCR is consistent with the verified and validated design resulting from the overall HFE design process.

ITAAC 03.15.02, which corresponds to standardized ITAAC No. H01, is for the completion of an integrated system validation (ISV) test in accordance with the human factors verification and validation (V&V) implementation plan (IP).

ITAAC 03.15.01 verifies that the as-built MCR human-system interfaces (HSIs) are consistent with the HSI resulting from the applicant’s HFE design process. Specifically, the ITAAC requires that the as-built HSI be consistent with the design verified and validated by the ISV as reconciled by the DI IP. The staff reviews the DI IP in Chapter 18 of this SER. The DI IP describes human factors activities that ensure that changes to the NuScale HSI design that occur after ISV and before startup will be assessed to ensure that there are no unintended effects on human performance. These activities help to ensure that the conclusions drawn regarding operator performance based on ISV tests will remain valid as the design continues to evolve. The staff finds that the ITAAC system description in SDAA Part 8, Section 3.15.1, adequately describes the top-level objectives for the applicant's HFE program design process for the control room design and HSI. The staff evaluates the applicant's HFE program in accordance with the review criteria of SRP Chapter 18, "Human Factors Engineering," and NUREG-0711, Revision 3, "Human Factors Engineering Program Review Model," issued November 2012. The staff evaluates the applicant's HFE program in Chapter 18 of this SER, based on technical information contained in IPs and result summary reports.

The applicant submitted IPs for six elements of the HFE program model: Operating Experience Review (OER), Function Requirements Analysis and Function Allocation (FRA/FA), Task Analysis (TA), Human-System Interface Design (HSI design), V&V, and DI. IPs describe the applicant's proposed methodology for conducting an HFE element. The NRC staff reviews the applicant's methodology to verify that it meets the review criteria. Acceptance criteria in SRP Section 14.3.9 state that if an IP, rather than a completed HFE element, was accepted as part of the DC process, then ITAAC should address the completion of the HFE program element. During the audit process, NuScale explained how HFE ITAAC No. 3.15.01 ensures the completion of the OER, FRA/FA, TA, HSI design, and V&V elements. NuScale stated that the HFE program for the SDA concludes with DI and that, as detailed in TR-130414-NP, Revision 0, "Human Factors Engineering Program Management Plan," issued December 2022 (HFE PMP), DI cannot occur without the completion of the other HFE elements. Therefore, ITAAC 03.15.01 for DI ensures completion of the other elements.

The staff reviewed the HFE PMP and the DI IP. Section 1.1 of the DI IP states that DI, once complete, ensures that the final or as-built HFE design conforms to the verified and validated design, which is the design that resulted from the applicant's HFE design process. The HFE PMP describes the applicant's HFE design process. The HFE PMP describes aspects of each HFE technical program element and shows how the elements interface with each other, giving examples of how the design process iterates to achieve improved results. For example, treatment of important human actions is a direct input into TA; important human actions are analyzed during TA to ensure each action is feasible and reliable. Figure A-2, "HFE Program Process," shows the iterative nature of the NuScale HFE design process. For example, iterations of the HSI design element occur because of inputs from staffing and qualification and V&V activities; the HSI design matures from the initial HSI design towards a final design that is tested during ISV. HFE PMP, Section 6.10, states that the DI process ensures that any design changes that occur after V&V are evaluated to determine the impact on the completed HFE elements. The staff concludes that the DI activity depends on completion of the NuScale HFE design process; therefore, the staff finds that ITAAC 03.15.01 is acceptable to ensure the completion of OER, FRA/FA, TA, HSI design, V&V, and DI and that a separate ITAAC for each of these HFE elements is not necessary. The SDAA includes ITAAC 03.15.02 for the ISV test as a separate HFE ITAAC because the applicant uses the ISV test to demonstrate that the MCR design incorporates HFE principles that reduce the potential for operator error. The staff will also use this as evidence that the HFE activities described in the IPs are complete. Additionally, in FSAR Section 18.1, the applicant committed to submitting result summary reports for OER, FRA/FA, TA, HSI design, and V&V before fuel load. The NRC staff can audit these result summary reports during the COL stage or as part of verifying the closure of ITAAC 03.15.01. The staff finds this to be an acceptable means of ensuring that HFE elements are complete.

Accordingly, the staff finds the HFE ITAAC to be an acceptable means of confirming that the final as-built control room is consistent with the design validated during the ISV test and that any deviations from the validated design will be assessed, and if needed resolved, according to an acceptable process described in the DI IP.

Review procedures in SRP Section 14.3.9, Revision 0, issued March 2007, direct the staff to ensure the standard ITAAC entries in SRP Section 14.3, Appendix D, “ITAAC Entries—Examples,” are included for each plant system that has alarms, controls, or displays. Appendix D to SRP Section 14.3 includes ITAAC entries for alarms, controls, or displays in the MCR and the remote shutdown station (RSS). In addition, the draft standardized ITAAC include entries for related ITAAC. Therefore, the staff also reviewed the ITAAC in SDAA Part 8, Table 2.5-7, “Module Protection System and Safety Display and Indication System ITAAC,” and FSAR Chapter 14. The staff compared the applicant’s ITAAC to the draft standardized ITAAC and found that NuScale did include ITAAC for displays, controls, and alarms in the MCR, which are reviewed in Section 14.3.5 of the SER. However, the applicant did not include ITAAC for the RSS.

The US460 design does not include an RSS; the capability to remotely shut down the reactor exists at the MPS cabinets, as stated in FSAR Chapter 7, Section 7.1.1.2.3, “Alternate Operator Workstation Controls and Monitoring,” which indicates that operators can achieve safe shutdown of the reactors from outside the MCR in the MPS I&C equipment rooms. ITAAC 03.07.03 in SDAA Part 8, Table 3.7-2, “Fire Protection System ITAAC,” is for a complete safe shutdown analysis to verify the alternative shutdown capability from the I&C equipment rooms.

The staff finds it acceptable that the applicant has excluded ITAAC or displays, alarms, and controls in the RSS because there is no RSS, and the application includes ITAAC to verify the remote shutdown capability of the MPS.

14.3.9.5 Combined License Information Items

There are no COL information items associated with this section.

14.3.9.6 Conclusion

The staff concludes that SDAA Part 8 satisfactorily summarizes the top-level HFE program design process objectives that are significant to safety and used to develop the HFE design and that it is consistent with SDAA Part 2, Chapter 18, “Human Factors Engineering.” Therefore, the design information associated with SDAA Part 8, Section 3.15, is acceptable.

Furthermore, the staff concludes that the HFE ITAAC adequately verify the SDAA Part 8 HFE design. Therefore, within the review scope of this section, the staff concludes that the NuScale HFE ITAAC are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria are met, a facility that incorporates the standard NuScale design has been constructed and will be operated in conformity with the applicable portions of the SDAA, the AEA, and the NRC’s rules and regulations.

14.3.10 Emergency Planning—Inspections, Tests, Analyses, and Acceptance Criteria

The applicant did not provide site-specific emergency planning ITAAC for the standard design and specified in COL Item 14.3-1, shown in this report's Table 14.3.10-1, that a license applicant that references the NuScale US460 standard design will provide the site-specific emergency planning ITAAC. Section 13.3.4.5 of this SER evaluates the emergency planning ITAAC.

Table 14.3.10-1 NuScale US460 SDAA COL Information Items

Item No.	Description	FSAR Section
14.3-1	An applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for emergency planning.	14.3.10

14.3.11 Containment Systems—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.11.1 Introduction

This section reviews ITAAC and design descriptions applicable to containment and associated systems. The NuScale CNTS ITAAC are listed in the following SDAA Part 8 table:

- Table 2.1-1, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.01.xx)," Numbers 7–9 and 12

14.3.11.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.11.3 Regulatory Basis

See Section 14.3.1.3 of this SER. SRP Section 14.3.11 provides acceptance criteria and additional guidance for this review area.

14.3.11.4 Technical Evaluation

The staff reviewed the system- and nonsystem-based ITAAC in accordance with SRP Section 14.3.11, particularly the applicable review procedures, as well as the guidance in RG 1.206, Section C.II.1. The staff examined the ITAAC to ensure that they can be completed by the organization holding the COL. The staff examined the phrasing and format of the ITAAC to determine whether they were consistent (i.e., the design commitment; the inspection, test, or analysis; and the acceptance criteria are parallel and in agreement). In addition, the staff determined that the SDAA Part 8 ITAAC items were derived from the SDAA Part 2 information. The staff reviewed the information and finds that it is consistent with the NuScale design and the associated ITAAC.

14.3.11.4.1 *Containment Systems ITAAC*

The staff used the following SRP sections identified in SRP Section 14.3.11 that have a potential impact on the ITAAC sections related to CNTS:

- SRP Section 14.3 (general guidance on ITAAC)
- SRP Section 14.3.2 (the ability of SSCs to withstand various natural phenomena)
- SRP Section 14.3.3 (piping design)
- SRP Section 14.3.5 (I&C)
- SRP Section 14.3.6 (electrical systems and components)
- SRP Chapter 19, "Severe Accidents" (design of the features and functions of SSCs that should be addressed based on severe accident, probabilistic risk assessment, and shutdown safety evaluations)

The staff assessed the CNTS ITAAC items associated with the following SDAA Part 2 sections in accordance with the applicable procedures and guidance in SRP Sections 14.3 and 14.3.11:

- Section 6.2.4, "Containment Isolation System"
- Section 6.2.6, "Containment Leakage Testing"

14.3.11.4.2 *Containment Isolation System ITAAC*

The CNTS provides for the isolation of process systems that penetrate the CNV. The purpose of containment isolation is to permit the normal or post-accident passage of fluids through the containment boundary, while protecting against the release to the environment of fission products that may be present in the containment atmosphere and fluids because of postulated accidents.

SDAA Part 8, Section 2.1, specifies ITAAC for containment isolation. SDAA Part 8, Section 2.1, includes design commitments requiring that CIV closure times limit potential releases of radioactivity and that the length of piping between the containment penetration and the associated outboard CIVs be minimized. Tables in SDAA Part 8, Section 2.1, define the required closure times and piping lengths. SDAA Part 8, Table 2.1-1, includes ITAAC Number 8 to verify CIV closure times and ITAAC Number 9 to verify the length of piping between each penetration and its associated outboard CIV.

SDAA Part 8, Section 2.1.1, "Design Description," describes the containment pressure boundary as a top-level design feature by "providing a barrier to contain mass, energy, and fission product release." The staff reviewed the information and finds that it is consistent with SRP Section 14.3 because the containment boundary, which includes the containment isolation function, is a top-level design feature based on the safety significance of containment as identified in safety analyses and defense-in-depth considerations.

The staff reviewed the proposed ITAAC requirements specified in SDAA Part 8, Section 2.1, Table 2.1-1, ITAAC Numbers 8 and 9, and finds the ITAAC to be consistent with the staff guidance contained in SRP Section 14.3.11 and the standardized ITAAC (ML16096A132) because the valve closure times limit potential releases of radioactivity and the CIVs outside containment are located as close to containment as practical. The staff finds that the proposed ITAAC are acceptable and meet the requirements in 10 CFR 52.47(b)(1) because the ITAAC are consistent with staff guidance.

14.3.11.4.3 *Containment Leakage Testing ITAAC*

The SDAA Part 8 design description states that the containment is an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. This containment design description is acceptable because it meets the criteria for accommodating the pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the design leakage rate, in accordance with GDC 50, “Containment Design-Basis.” This design description is consistent with SDAA Part 2, Section 6.2.6.

The containment leakage rate testing is designed to verify the leak-tight integrity of the CNV by showing that leakage will not exceed the allowable leakage rate specified in the TS. The preoperational and periodic containment leakage testing capability for CNV openings (Type B) and CNV piping penetrations (Type C) is designed to meet the leakage acceptance criteria of 10 CFR Part 50, Appendix J.

The applicant has requested an exemption from the GDC 52, “Capability for Containment Leakage Rate Testing,” requirement to design the containment for integrated leak rate testing. The applicant has requested an exemption from the 10 CFR Part 50, Appendix J, requirements for preoperational and periodic Type A integrated leak rate testing. The staff has reviewed this exemption request and determined that it meets the requirements for exemptions as described in Section 6.2.6 of this SER.

The CNV serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. The containment leakage testing program performs the following safety-related functions that are verified by ITAAC: Type B tests are intended to detect and measure local leaks for reactor containment penetrations. Type C tests are intended to measure CIV leakage rates.

The staff reviewed the proposed ITAAC Number 7 in SDAA Part 8, Table 2.1-1, which lists the following test and acceptance criteria:

- A leakage test will be performed of the pressure-containing or leakage-limiting boundaries, and CIVs.

- The leakage rate for local leak rate tests (Type B and Type C) for pressure-containing or leakage-limiting boundaries and CIVs [meet] the requirements of 10 CFR Part 50, Appendix J.

The staff finds that the applicant has adequately identified ITAAC consistent with the requirements for Type B and Type C testing, consistent with the guidance in SRP Section 14.3.11.

The staff has also reviewed ITAAC Number 12 in SDAA Part 8, Table 2.1-1, which lists the following test and acceptance criteria and was proposed by NuScale to support the exemption request:

- A preservice design pressure leakage test of the CNV will be performed.
- No water leakage is observed at CNV bolted flange connections.

This ITAAC is intended to confirm that the design of the bolted flanges (Type B penetrations) results in no leakage. This ITAAC is acceptable, as the preservice design pressure test resulting in zero leakage at the bolted flanges demonstrates that the bolted flange design is leak tight.

14.3.11.5 Combined License Information Items

Table 14.3.11-1 lists COL information item numbers and descriptions related to this area of review from SDAA Part 2, Section 6.2.6. COL item 6.2-1 is evaluated in Section 6.2.6 of this SER.

Table 14.3.11-1 NuScale US460 SDAA COL Information Items

Item No.	Description	FSAR Section
6.2-1	An applicant that references the NuScale Power Plant US460 standard design will verify that the final design of the containment vessel meets the design-basis requirement to maintain flange contact pressure at accident temperature, concurrent with peak accident pressure.	6.2.6

14.3.11.6 Conclusion

The staff concludes that if the inspections, tests, and analyses for the CNTS are performed and the acceptance criteria met, there is reasonable assurance the NuScale standard design nuclear power plant has been constructed and will be operated in accordance with the applicable portions of the SDAA, the AEA, and NRC rules and regulations, in compliance with 10 CFR 52.47(b)(1). The staff also concludes that the applicant has included sufficient top-level design information consistent with SECY-19-0034.

14.3.12 Physical Security Hardware—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.12.1 Introduction

This section identifies ITAAC design descriptions applicable to physical security systems (PSS) in the SDAA Part 8, Revision 1. The following table contains the ITAAC applicable to this review area:

- Table 3.16-1, “Physical Security System ITAAC,” Numbers 1–13

14.3.12.2 Summary of Application

The SDAA Part 2 sections cited below, and the referenced technical report, contain the applicant’s descriptions of the PSS and physical security ITAAC (SDAA Part 8) for the standard design and describe how they meet regulatory requirements.

SDAA Part 2: SDAA Part 2, Section 13.6.1, "Physical Security," states that "[t]he NuScale Power Plant physical security design provides the capabilities to detect, assess, impede, and delay threats up to and including the design-basis threat, and to provide defense-in-depth through the integration of systems, technologies, and equipment. The design of PSS within the nuclear island and structures is described in technical report TR-118318, 'NuScale Design of Physical Security Systems,' (Reference 13.6-1), which is incorporated by reference to this Final Safety Analysis Report."

SDAA Part 8: SDAA Part 8, Section 3.0, "Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Based Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design Descriptions and ITAAC," and Table 3.0-1, "Shared Systems Subject to Inspections, Tests, Analyses, and Acceptance Criteria," identify the systems that support multiple NPMs and are verified by ITAAC.

SDAA Part 8, Section 3.16.1, "Inspections, Tests, Analyses, and Acceptance Criteria Design Description," describes the standard design commitments for the PSS that implement security response functions.

Technical Reports: By letter dated November 23, 2022, the applicant submitted to the NRC TR-118318, Revision 0, which describes the security considerations in the standard design. By letter dated August 28, 2023, the applicant submitted to the NRC TR-118318, Revision 1. This technical report describes the design bases for the PSS designs, including plant layout and building configurations, results of evaluations, and identified vital equipment and areas for the standard design. The scope of the PSS described in the SDAA is limited to the PSS related to the nuclear island and structures that are within the scope of the standard design. TR-118318, Revision 1, contains safeguards information, security-related information, and proprietary information; therefore, it is protected in accordance with 10 CFR 73.21, "Protection of Safeguards Information: Performance requirements," and 10 CFR 2.390, "Public inspections, exemptions, requests for withholding."

14.3.12.3 Regulatory Basis

In addition to the regulations listed in Section 14.3.1.3 of this SER, the following NRC regulations contain the relevant requirements for this review:

- 10 CFR 52.54, "Issuance of standard design certification," section (a)(5) requires that the proposed ITAAC are necessary and sufficient, within the scope of the standard design, to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in accordance with the design certification, the provisions of the AEA, and the Commission's regulations.

- 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," requires that managerial and administrative controls, programs, and processes for security are addressed by the COL applicant and are not within the scope of the standard design. Section 52.79(a)(28) requires COL applicants to provide plans for preoperational testing and initial operations. Section 52.79(a)(35) requires a physical security plan describing how the applicant will meet the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials," and listing tests, inspections, audits, and other means to be used to demonstrate compliance with the requirements of 10 CFR Part 73.

- 10 CFR 52.139, “Standards for review of applications,” requires applications filed under this subpart to be reviewed for compliance with the standards set out in 10 CFR Part 73.

- 10 CFR Part 73, “Physical Protection of Plants and Materials,” includes performance and prescriptive requirements that, when adequately met and implemented, provide protection against acts of radiological sabotage, prevent the theft or diversion of special nuclear material, and protect safeguards information.

- 10 CFR 73.55, “Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage,” subsection 73.55(b) requires COL applicants to describe PSS and the security organization whose objective will be to provide high assurance[5] that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to public health and safety.

- 10 CFR 73.1(a)(1) describes the design-basis threat for radiological sabotage. The provisions within 10 CFR 73.54, “Protection of digital computer and communication systems and networks”; 10 CFR 73.55; 10 CFR 73.56, “Personnel access authorization requirements for nuclear power plants”; 10 CFR 73.58, “Safety/security interface requirements for nuclear power reactors”; and Appendix B, “General Criteria for Security Personnel,” and Appendix C, “Licensee Safeguards Contingency Plans,” to 10 CFR Part 73, establish performance and prescriptive requirements that apply to the design of the PSS, operational security requirements, management processes, and programs.

SRP Section 14.3.12, “Physical Security Hardware—Inspections, Tests, Analyses, and Acceptance Criteria,” provides acceptance criteria and additional guidance for this review area.

14.3.12.4 Technical Evaluation

The staff reviewed the design descriptions described in top-level physical security features and performance characteristics to determine whether they satisfy regulatory requirements. SDAA Part 2, Section 14.3.2, states that “the selection of the top-level design features is based on the safety significance of SSC, their importance in various safety analyses, and their functions for defense-in-depth considerations.” The staff’s review also included the review of ITAAC required to ensure the reliability, availability, and performance of the PSS.

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[5] The general performance objective of 10 CFR 73.55(b)(1) is to provide “high assurance that activities involving special nuclear material are not inimical to the common defense and security and do not constitute an unreasonable risk to the public health and safety.” In SRM-SECY-16-0073, “Staff Requirements—SECY 16 0073—Options and Recommendations for the Force-on-Force Inspection Program in Response to SRM-SECY-14-0088,” dated October 5, 2016, the Commission stated that “the concept of ‘high assurance’ of adequate protection found in our security regulations is equivalent to ‘reasonable assurance’ when it comes to determining what level of regulation is appropriate.” Throughout this publication, the term “high assurance” is used in alignment with Commission policy statements that high assurance is equivalent to reasonable assurance of adequate protection.

The PSS described in the standard design (and those specific to a COL application) must be reliable and available to ensure their performance and to meet their intended security functions. The design and technical bases for the PSS are described in the SDAA Part 2, Section 13.6, "Security," which incorporates by reference TR-118318, Revision 1. These documents provide the system designs and performance requirements that support the identified ITAAC design commitments for verification.

14.3.12.4.1 *Design Commitments, Inspections, Tests, Analyses, and Acceptance Criteria*

SDAA Part 2, Section 13.6.1, and TR-118318, Revision 1, describe the design of the PSS that is relied on to implement security response functions (i.e., detection, assessment, communications, security response—delays, interdictions, and neutralization). The ITAAC described in SDAA Part 8, Table 3.16-1, include those related to vital equipment locations, physical barriers, bullet-resistant structures, physical and access controls and security measures for vital areas, intrusion detection and assessment systems and subsystems and components, location of the central alarm station (CAS), and communications that meet the requirements of 10 CFR Part 73. These ITAAC verify the following design commitments for PSS in the scope of the standard design:

(1) Vital equipment will be located only within a vital area.

(2) Access to vital equipment will require passage through at least two physical barriers.

(3) The external walls, doors, ceilings, and floors in the MCR and the CAS will be bullet resistant.

(4) An access control system will be installed and designed for use by individuals who are authorized access to vital areas within the nuclear island and structures without escort.

(5) Unoccupied vital areas within the nuclear island and structures will be designed with locking devices and intrusion detection devices that annunciate in the CAS.

(6) The CAS will be located inside the protected area and will be designed so that the interior is not visible from the perimeter of the protected area.

(7) Security alarm devices in the RXB and CRB, including transmission lines to annunciators, will be tamper indicating and self-checking, and alarm annunciation indicates the type of alarm and its location.

(8) Intrusion detection and assessment systems in the RXB and CRB will be designed to provide visual display and audible annunciation of alarms in the CAS.

(9) Intrusion detection systems' recording equipment will record security alarm annunciations within the nuclear island and structures, including each alarm, false alarm, alarm check, and tamper indication, and the type of alarm, location, alarm circuit, date, and time.

(10) Emergency exits through the vital area boundaries within the nuclear island and structures will be alarmed with intrusion detection devices and will be secured by locking devices that allow prompt egress during an emergency.

(11) The CAS will have a landline telephone service with the control room and local law enforcement authorities.

(12) The CAS will be capable of continuous communication with on-duty security force personnel.

(13) Nonportable communications equipment in the CAS will remain operable from an independent power source in the event of the loss of normal power.

SDAA Part 2, Section 13.6, states that the applicant will address site-specific ITAAC as described in SDAA Part 2, Section 14.3.

SDAA Part 2, Section 14.3.1, COL Information Item 14.3-2, states that "[a]n applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for structures, systems, and components within their scope."

The staff made the following findings:

Physical security ITAAC described in SDAA Part 8, Table 13.6-1, comply with 10 CFR 52.54(a)(5).

The design requirement and physical security ITAAC for illumination of the security isolation zones and exterior areas within the protected areas are not within the scope of the standard design and, therefore, will be addressed by the license applicant.

The applicant adequately identified other PSS features, such as protected area barriers; isolation zones; protected area intrusion detection; engineered access controls for personnel, vehicles, and material; and personnel identification systems that are outside the scope of the standard design and that will be addressed by the license applicant.

14.3.12.4.2 *Verification Program and Processes*

SDAA Part 2, Section 14.0, "Verification Programs," describes verification programs for the standard design, including ITAAC, that ensure that the as-built facility configuration and operation comply with the approved plant design and applicable regulations.

SDAA Part 8, Table 3.0-1, identifies PSS as shared systems supporting the NPMs. Specifically, it states that the PSS is one system that supports six modules. Section 3.0 states "[s]atisfactory completion of a shared ITAAC for the lead module shall constitute satisfactory completion of the shared ITAAC for associated modules. The ITAAC in Sections 3.1 through 3.16 shall only be completed once in conjunction with the ITAAC in Chapter 2 for the lead NPM." The applicant indicated that the physical security ITAAC identified in SDAA Part 8, Table 3.16-1, ITAAC Numbers 1 through 13, are not NPM specific; instead, they verify engineered SSCs that provide security functions throughout the RXB and CRB and are verified by ITAAC for the lead NPM.

The staff finds the following:

The test program, as described in SDAA Part 2, Sections 14.2 and 14.3, which the license applicant must establish, if adequately implemented, will demonstrate through testing that credited engineered SSCs will perform their intended security functions.

The applicant established the requirements for a license applicant referencing the standard design to verify the installation, construction, and performance of the PSS through ITAAC.

ITAAC verification of common (shared) PSS that support all NPMs before the first NPM fuel load is acceptable; however, the staff notes that it is the license applicant's responsibility to meet 10 CFR 52.103(g), which states that "[t]he licensee shall not operate the facility until the Commission makes a finding that the acceptance criteria in the license are met, except for those acceptance criteria that the Commission found were met under § 52.97(a)(2). If the license is for a modular design, each reactor module may require a separate finding as construction proceeds."

14.3.12.4.3 *Verification Methods for Physical Security ITAAC*

SDAA Part 8, Section 1.2.5, "Implementation of Inspections, Tests, Analyses, and Acceptance Criteria," indicates that the verification (inspections, tests, and analyses) may be performed by more than a single individual or group, implemented through discrete activities separated by time, performed at any time before fuel load (including before the issuance of the COL for those ITAAC that do not require as-built equipment), and performed at locations other than the construction site. Additionally, the applicant indicated that inspections, tests, and analyses may be performed as part of other activities, such as construction inspections or preoperational testing, and that the inspections, tests, and analyses do not need to be performed as separate or discrete activities.

SDAA Part 2, Section 14.2, "Initial Plant Test Program," and SDAA Part 8, Section 3.16.2, "Inspections, Tests, Analyses, and Acceptance Criteria," discuss performance methodologies for physical security ITAAC in more detail.

14.3.12.4.3.1 Inspections, Tests, and Analyses for Vital Equipment and Vital Areas

The acceptance criteria identified for the physical security ITAAC related to the vital areas are the successful inspections and tests that verify locking, intrusion detection, and alarms in accordance with the requirements of 10 CFR 73.55(e)(9)(i) through (iii) and 10 CFR 73.55(e)(8)(iii).

ITAAC Number 1 — An inspection is performed of vital equipment to verify that the equipment is located within a vital area.

The methods described in SDAA Part 8, Table 3.16-2, "Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information," include inspections to locate vital equipment and verify that access to each component meets the stated objective.

ITAAC Number 2 — An inspection is performed of vital equipment location to verify that access to vital equipment requires passage through at least two physical barriers.

The methods described in SDAA Part 8, Table 3.16-2, include inspections performed of the as-built vital equipment locations to verify that access to vital equipment within the nuclear island and structures requires passage through at least two physical barriers. The list of vital equipment in

TR-118318, Revision 1, is needed for verification of physical security ITAAC Number 2.

ITAAC Number 3	A type test, analysis, or a combination of type test and analysis are performed of the bullet-resisting barriers used in the external walls, doors, ceilings and floors in the MCR and CAS. The methods described in SDAA Part 8, Table 3.16-2, include the statement that a qualification will demonstrate that the barriers are bullet-resistant to Underwriters Laboratories Ballistic Standard 752, “The Standard of Safety for Bullet-Resisting Equipment,” Level 4, or National Institute of Justice Standard 0108.01, “Ballistic Resistant Protective Materials,” Type III.
ITAAC Number 4	A test demonstrates that the access control system provides authorized access to vital areas, within the nuclear island and structures, only to those individuals with authorization for unescorted access. SDAA Part 2, Section 14.2, Table 14.2-66, “Security Access Control,” describes the test abstract for physical security ITAAC Number 4 for verifying the access control system with a numbered photo-identification badge system.
ITAAC Number 5	A test, inspection, or a combination of test and inspection demonstrates that unoccupied vital areas, within the nuclear island and structures, are locked and alarmed and intrusion is detected and annunciated in the CAS. SDAA Part 2, Section 14.2, Table 14.2-67, “Security Detection and Alarm,” describes the test abstract for physical security ITAAC Number 5 for locked and alarmed access into vital areas. The verification methods include testing the unauthorized opening of each vital area access door to verify that an intrusion alarm is generated, verifying that alarms are detected by the alarm annunciator computers and displays in the CAS, verifying audible and visual alarm annunciation in the CAS, and verifying recording of alarm information.
ITAAC Number 6	An inspection is performed of the CAS to verify that it is located inside the protected area and the interior is not visible from the perimeter of the protected area. The methods described in SDAA Part 8, Table 3.16-2, include inspections of the as-built CAS to verify that it is located inside the protected area and the interior is not visible from the protected area perimeter.
ITAAC Number 10	A test, inspection, or a combination of test and inspection demonstrates that emergency exits through the vital area boundaries, within the nuclear island and structures, are alarmed with intrusion detection devices and secured by locking devices that allow prompt egress during an emergency.

SDAA Part 2, Section 14.2, Table 14.2-67, describes the test abstract for physical security ITAAC Number 10 for emergency exits through the vital area boundaries. The verification methods include verification of emergency exits from the vital areas within the nuclear island and structures have installed locking devices, which will allow emergency egress, and installed alarms that will notify the CAS operator that the door has been opened.

The staff finds the following:

The applicant provided adequate descriptions of the objectives, prerequisites, methods, and acceptance criteria that support the identified ITAAC related to the vital equipment and vital areas and emergency exit controls for the vital areas in SDAA Part 2, Section 14.2, and SDAA Part 8, Section 3.16.2.

14.3.12.4.3.2 Inspections, Tests, and Analyses for Alarms, System Supervision, Assessment, and Records

The acceptance criteria identified for the physical security ITAAC related to the alarms, system supervision, assessment, and records verify that the criteria are in accordance with the requirements of 10 CFR 73.55(i)(3)(iv) through 10 CFR 73.55(i)(3)(v) for ITAAC Number 7 and in accordance with the requirements of 10 CFR 73.55(i)(3)(i) through 10 CFR 73.55(i)(3)(iii), 10 CFR 73.55(i)(4)(ii)(H), and 10 CFR 73.55(i)(2) for ITAAC Numbers 8 and 9.

ITAAC Number 7 A test demonstrates that security alarm devices in the RXB and CRB, including transmission lines to annunciators, are tamper indicating and self-checking; an automatic indication is provided when failure of the alarm system or a component thereof occurs or when the system is on standby power; and the alarm annunciation indicates the type of alarm and location.

SDAA Part 2, Table 14.2-67, describes the test abstract for ITAAC Number 7. The test method includes (1) inserting a signal real or simulated tamper signal, (2) inserting a signal real or simulated of a component failure for all alarm devices and transmission lines in the RXB and CRB, and (3) placing all security alarm devices in the RXB and CRB on standby power to verify the test objective.

ITAAC Number 8 A test demonstrates that the intrusion detection and assessment system provide visual display and audible annunciation of alarms in the CAS.

SDAA Part 2, Table 14.2-67, describes the test abstract for ITAAC Number 8. The test method includes putting all intrusion detection equipment described in TR-118318, Revision 1, into an alarm state to verify the test objective.

ITAAC Number 9 A test demonstrates that the intrusion detection and assessment systems' recording equipment is capable of recording each security alarm annunciation within the nuclear island and structures, including each alarm, false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, and time.

SDAA Part 2, Table 14.2-67, describes the test abstract for ITAAC Number 9. The test method includes placing all intrusion detection equipment in the RXB and CRB in a false alarm, alarm check, and tamper indication alarm condition, as applicable to the equipment to verify the test objective.

The staff finds the following:

The applicant provided adequate descriptions to support the identified ITAAC related to security alarm, system supervision, assessment, and intrusion detection system records in SDAA Part 2, Section 14.2.

These ITAAC do not cover the secondary alarm station because the license applicant is responsible for providing a secondary alarm station that is equal and redundant to the CAS (COL Item 13.6-3).

14.3.12.4.3.3 Inspections, Tests, and Analyses for Security Communications

The acceptance criteria identified for the physical security ITAAC related to security communications are in accordance with the requirements of 10 CFR 73.55(j)(3), 10 CFR 73.55(j)(4)(i) through (4)(ii), and 10 CFR 73.55(j)(5).

SDAA Part 2, Table 14.2-61, "Communication System," describes the following preoperational inspections and tests that demonstrate the system's physical security functions for ITAAC Numbers 11, 12, and 13:

ITAAC Number 11 A preoperational test, inspection, or a combination of test and inspection demonstrates that the CAS is equipped with conventional landline telephone service with the MCR and with local law enforcement authorities.

SDAA Part 2, Table 14.2-61, "Communication System," describes the test abstract for ITAAC 11. The test method includes testing the conventional (landline) service from the CAS to the MCR and local law enforcement authorities to verify the test objective.

ITAAC Number 12 A preoperational test, inspection, or a combination of test and inspection demonstrates that the CAS is capable of continuous communication with on-duty security force personnel.

SDAA Part 2, Table 14.2-61, describes the test abstract for ITAAC 12. The test method includes testing communications with the plant radio system in areas described in the physical protection program boundaries and areas described in the contingency response event areas to verify the test objective.

ITAAC Number 13 A preoperational test, inspection, or a combination of test and inspection demonstrates that nonportable communications equipment in the CAS remains operable (without disruption) from an independent power source in the event of loss of normal power.

SDAA Part 2, Table 14.2-61, describes the test abstract for ITAAC 13. The test method includes removing normal power from the CAS nonportable communication devices to verify the test objective.

The staff finds the following:

The applicant provided adequate descriptions of the objectives, verification methods, and acceptance criteria that support the identified physical security ITAAC related to security communications in SDAA Part 2, Section 14.2.

14.3.12.5 Combined License Information Items

SDAA Part 2, Table 1.8-1, lists the COL information item number and description related to Section 14.3.12:

Item No.	Description of COL Information Item	Section
COL Item 14.3-2	An applicant that references the NuScale Power Plant US460 standard design will provide the site-specific selection methodology and inspections, tests, analyses, and acceptance criteria for structures, systems, and components within their scope.	14.3

14.3.12.6 Conclusion

The NRC staff has determined that these PSS ITAAC include the necessary elements that, when effectively implemented, will provide protection against the design-basis threat of radiological sabotage as described in 10 CFR 73.1(a)(1). The responsibility to effectively implement these plans remains with the license applicant.

Specifically, the applicant (1) proposed and adequately described attributes for physical security ITAAC verification, (2) identified an appropriate and reasonable set of test methods (inspections, tests, or analyses) and acceptance criteria for the standard design that comply with 10 CFR 52.54(a)(5), (3) provided adequate descriptions of elements of the test abstracts and inspections and analyses for verifying PSS (i.e., objectives, prerequisites, test methods, data requirements, and acceptance criteria), (4) identified appropriate descriptions for tests, inspections, and analyses that establish the framework for developing the detailed procedures for the conduct of the ITAAC, and (5) provided adequate descriptions of requirements (i.e., COL Information Item 14.3-2) that indicate that a license applicant referencing the standard design will describe the ITAAC for PSS that are outside the standard design.

The staff concludes that the applicant has met 10 CFR 52.54(a)(5), which requires that the standard design proposed inspections, tests, analyses, and acceptance criteria are necessary and sufficient, within the scope of the standard design, to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in accordance with the design, the provisions of the AEA, and the Commission's regulations. The staff concludes that the applicant has provided sufficient information in the test program for the physical security test abstracts to satisfy 10 CFR 52.79(a)(28).

14.3.13 External Flooding Protection—Inspections, Tests, Analyses, and Acceptance Criteria

14.3.13.1 Introduction

The staff reviewed the ITAAC and its design descriptions for protecting the seismic Category I RXB and the seismic Category I CRB against external flooding in SDAA Part 8, Section 3.11, and Section 3.13, "Control Building", and notes that the following ITAAC tables contain the ITAAC applicable to this review area:

- SDAA Part 8, Table 3.11-1, "Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria," Number 03

- SDAA Part 8, Table 3.13-1, "Control Building Inspections, Tests, Analyses, and Acceptance Criteria," Number 03

14.3.13.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.13.3 Regulatory Basis

See Section 14.3.1.3 of this SER. SRP Section 14.3.2; SRP Section 2.0, "Site Characteristics and Site Parameters"; and SRP Section 2.4.2, "Floods," provide acceptance criteria and additional guidance for this review area.

14.3.13.4 Technical Evaluation

The staff reviewed the external flooding-related information in FSAR Section 2.4.2, "Flooding," Section 3.4.2.1, "Probable Maximum Floods," and Table 2.0-1, "Site Parameters," and finds that the maximum flood elevation (including wind-induced wave run-up and other effects) is 1 foot below baseline plant elevation. In addition, the staff reviewed ITAAC Number 03 in SDAA Part 8, Table 3.11-1, and ITAAC Number 03 in SDAA Part 8, Table 3.13-1, and finds that ITAAC inspections are performed to verify that the as-built RXB and CRB floor elevations at ground entrances are located above the maximum external flood elevations to protect the RXB from external flooding. Therefore, the staff concludes that the ITAAC and its design descriptions for protecting the seismic Category I RXB and the seismic Category I CRB against external flooding adequately describe the top-level design features and performance characteristics that are significant to safety because these features and characteristics appropriately require that the seismic Category I RXB and the seismic Category I CRB are protected from external flooding, as discussed below.

For the RXB, SDAA Part 8, Section 3.11.1, "Inspections, Tests, and Acceptance Criteria Design Description," states that the RXB supports the following systems by housing and providing structural support:

- NPM
- CVCS

- UHS
- MPS
- NMS

For the CRB, SDAA Part 8, Section 3.13.1, “Inspections, Tests, and Acceptance Criteria Design Description,” states that the CRB supports the MPS by housing and providing structural support and the CRB supports the normal CRVS by providing a portion of the CRE.

SDAA Part 8 design commitments for the RXB and CRB require that these seismic Category I structures be protected from external flooding to prevent flooding of safety-related SSCs within the structure. The ITAAC associated with these design commitments require inspections of the as-built RXB and CRB structures to ensure that the floor elevations at the ground entrances are higher than the maximum external flood elevation.

ITAAC Number 03 in SDAA Part 8, Table 3.11-1, and ITAAC Number 03 in SDAA Part 8, Table 3.13-1, along with their corresponding design commitments, conform to the standardized ITAAC and design commitments in the SDAA. The staff finds that the ITAAC design descriptions in SDAA Part 8 require that the safety-related SSCs of the seismic Category I RXB and the seismic Category I CRB are adequately protected from external flooding, and the ITAAC are sufficient to demonstrate this protection.

Based on the above review, the staff finds that these ITAAC comply with 10 CFR 52.47(b)(1), and that the external flooding protection ITAAC and its design descriptions in SDAA Part 8, Sections 3.11 and 3.13, are acceptable.

14.3.13.5 Combined License Information Items

There are no COL information items listed in SDAA Part 8, Sections 3.11 and 3.13, for this area of review.

14.3.13.6 Conclusion

The NRC staff finds that the SDAA Part 8 ITAAC for external flooding protection satisfy the requirements in 10 CFR 52.47(b)(1) and that the SDAA Part 8 design descriptions conform to NRC guidance. The staff also finds that the description of how to complete these ITAAC in SDAA Part 8, Tables 3.11-2 and 3.13-2, is acceptable.