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

14.1 Introduction

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC or Commission) staff (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 (NuScale) (hereafter referred to as the applicant) Design Certification Application (DCA), Part 2, “Final Safety Analysis Report (FSAR),” Revision 3 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19241A431). The Phase 2 SER for this chapter (ADAMS Accession No. ML18201A154) identified open items (OIs) all of which the applicant addressed through information submitted on the docket. Rather than discuss the individual OIs in this SER chapter, the staff has directly evaluated the adequacy of the information submitted on the docket to address them, which is included in the current version of the DCA. All of the OIs have been satisfactorily closed. With the exception of certain confirmatory items that have not been incorporated into DCA, Revision 3, the staff’s regulatory findings documented in this SER are based on the latest version of the application on the docket.

In this chapter, the NRC staff uses the term “nonsafety-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), Part 50, “Domestic Licensing of Production and Utilization Facilities,” Section 50.2, “Definitions.” However, among the “nonsafety-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, and others that are not considered “important to safety.”

14.2 Initial Test Program - Design Certification and New License Applicants

14.2.1 Generic Guidelines for Initial Test Programs

14.2.1.1 Introduction

The applicant for an operating license (OL) under Title 10 of the *Code Federal Regulations* (CFR), Part 50, “Domestic Licensing of Production and Utilization Facilities,” 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 structures, systems, and component (SSCs) 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 the health and safety of the public.

Initial startup testing consists of equipment performance tests completed during and after fuel loading. These performance tests are normally completed during 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 DCA.

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 the performance of 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," dated 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 [Light Water Reactor] Edition," Section 14.2, "Initial Test Program," Revision 3, dated 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 (iPWR) designs. On May 11, 2011, the Commission approved the staff's approach and provided additional direction (ADAMS Accession No. ML111320551). In response, the NRC staff subsequently developed a design specific review standard (DSRS) for the NuScale design. The 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 for review of the proposed NuScale ITP.

Section 14.2 of the DSRS notes that there is no requirement for a design certification (DC) applicant to provide an ITP submittal under 10 CFR Part 52, Subpart B, "Standard Design Certifications." 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 the Standard Review Plan Section 14.2 and RG 1.68.

14.2.1.2   Summary of Application

DCA Part 2, Tier 1: No Tier 1 information is provided in the NuScale DCA Part 2 for this program.

DCA Part 2, Tier 2: The applicant provided a Tier 2 program description in DCA Part 2, Tier 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. Preoperational testing is conducted following completion of construction testing but prior to fuel load. Completion of preoperational testing is necessary to ensure the overall plant is ready for fuel loading and startup testing of a NuScale Power Module (NPM).*

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

Technical Specifications: There are no generic technical specifications (GTS) 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 design certification that proposes nuclear reactor designs which differ significantly from light-water reactor 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 either analysis, appropriate test programs, experience, or a combination thereof.
- 10 CFR 52.47(c)(2), which requires that an application for certification for a nuclear power reactor design that differs significantly from the light-water reactor designs described in paragraph (c)(1) of Section 52.47 must provide an essentially complete nuclear power reactor design and 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 DCA Part 2, Tier 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 the RG 1.68 and DSRS Section 14.2. In DCA Part 2, Tier 2, Section 14.2, the applicant described the NuScale ITP, which consists of preoperational and initial startup tests.

For each phase of the ITP, a design certification applicant may define organizational responsibilities, provide 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, utilization of reactor operating and testing experience, use of 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 DCA Part 2, Tier 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 provided 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 has 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 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 DCA Part 2, Tier 2, Chapter 15; (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

Section 14.2 of the DSRS states that the COL applicant is responsible for providing a detailed description of management organizations and staff responsibilities, authorities, and qualifications. As such, in DCA Part 2, Tier 2, Section 14.2.2, "Organization and Staffing," the applicant provided COL Item 14.2-1, which states, "A COL applicant that references the NuScale Power Plant design certification will describe the site-specific organizations that manage, supervise, or execute the Initial Test Program, 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 COL holder will implement an 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; coordination with organizations involved in the test program; participation of plant operating and technical staff; and review, evaluation, and approval of test results.

In DCA Part 2, Tier 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 DCA 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 DCA 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, DCA Part 2, Tier 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.

DCA Part 2, Tier 2, Section 14.2.3.2, "Graded Approach to Testing," outlines the graded approach to testing, consistent with the requirements of General Design Criterion 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. It requires, in part, that SSCs important to safety shall 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 DCA Part 2, Tier 2 Section 17.4, "Reliability Assurance Program." As noted in the test abstracts in DCA Part 2, Tier 2 Section 14.2.12, "Individual Test Descriptions," the testable functions contain a safety and risk categorization.

The guidance in RG 1.68 and DSRS Section 14.2 describe certain tests that should be included in the ITP, such as first-of-a-kind (FOAK) tests, which are new, unique, or special tests used to verify design features that the NRC has not previously reviewed. As such, DCA Part 2, Tier 2, Section 14.2.3.3, "Testing of First-of-a-Kind Design Features," highlights the four FOAK tests, and refers to Table 14.2-110, "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 DCA Part 2 Tier 2, Section 14.2.3, "Test Procedures," are acceptable for the design certification 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 COL holder.

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. Section 14.2 of the DSRS 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 if performance of the tests in the proposed manner will accomplish the objectives of certain plant tests.

In DCA Part 2, Tier 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, DCA

Part 2, Tier 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 design certification because the NuScale 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 COL applicant. Thus, the staff concludes that those RGs do not apply to the NuScale design certification.

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 DC 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. Section 14.2 of the DSRS 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 DCA Part 2, Tier 2, Section 14.2.8, “Utilization of Reactor Operating and Testing Experience in Test Program Development,” the staff noted that 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 DCA Part 2, Tier 2, Section 14.2.4, “Conduct of the Test Program,” the staff noted that the COL applicant will be responsible for providing test specifications and test procedures for preoperational and startup tests for review by the NRC 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 a COL applicant that references the NuScale Power Plant design certification 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 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 detailed plant-specific design information review by the COL holder, and thus, the staff concludes that it is acceptable to defer the review of the use of operating and testing experience to the COL 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. Section 14.2 of the DSRS 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 DCA Part 2, Tier 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 COL applicant that references the NuScale Power Plant design certification 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 COL 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 COL 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.

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

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 DCA Part 2, Tier 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 test program schedule and sequence, specifically, 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 DCA Part 2, Tier 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 DCA states that the COL 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 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 the COL 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 COL 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

The individual test abstracts are provided in Tables 14.2-1 through 14.2-108 of DCA Part 2, Tier 2, Section 14.2. Each abstract identifies each test by title and gives the test objectives, prerequisites, test methods, and acceptance criteria. These test abstracts will be utilized 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:

- 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.

- Test abstracts associated with SSCs identified to be of low risk and low safety significance with no safety impact during initial plant start-up were given a more limited depth review.

- Test abstracts that will be developed by the COL applicant and are not reviewed at the DC stage. These test abstracts are indicated in Table 14.2-2 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 DC Stage

Abstract	Test Title
Table 14.2-1[2]	Spent Fuel Pool Cooling System Test #1
Table 14.2-2[2]	Pool Cleanup System Test #2
Table 14.2-3[2]	Reactor Pool Cooling System Test #3
Table 14.2-4	Pool Surge Control System Test #4

Commented [A1]: Confirmatory Item – Do not delete this comment until staff has verified the proposed changes described in letter dated 9/19/19 (ADAMS Accession No. ML19262G576) have been incorporated into the latest version of the DCA.

[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 Standard Review Plan, Introduction Part 2, the NRC staff limited the depth of the review for these abstracts.

Abstract	Test Title
Table 14.2-5	Ultimate Heat Sink #5
Table 14.2-6	Pool Leak Detection System Test #6
Table 14.2-7[2]	Reactor Component Cooling Water System Test #7
Table 14.2-8[2]	Chilled Water System Test #8
Table 14.2-9	Auxiliary Boiler System Test #9
Table 14.2-10[2]	Circulating Water System Test #10
Table 14.2-11[2]	Site Cooling Water System Test #11
Table 14.2-13[2]	Utility Water System Test #13
Table 14.2-14[2]	Demineralized Water System Test #14
Table 14.2-15[2]	Nitrogen Distribution System Test #15
Table 14.2-16[2]	Service Air System Test #16
Table 14.2-17[2]	Instrument Air System Test #17
Table 14.2-18	Control Room Habitability System Test #18
Table 14.2-19	Normal Control Room HVAC [Heating, Ventilation, and Air Conditioning] System Test #19
Table 14.2-20	Reactor Building HVAC System Test #20
Table 14.2-21[2]	Radioactive Waste Building HVAC System Test #21
Table 14.2-22[2]	Turbine Building HVAC System Test #22
Table 14.2-23[2]	Radioactive Waste Drain System Test #23
Table 14.2-24	Balance-of-Plant Drains Test #24
Table 14.2-25	Fire Protection System Test #25
Table 14.2-26[2]	Fire Detection System Test #26
Table 14.2-27[2]	Main Steam System Test #27
Table 14.2-28[2]	Feedwater System Test #28
Table 14.2-29[2]	Feedwater Treatment System Test #29
Table 14.2-30[2]	Condensate Polishing System Test #30
Table 14.2-31[2]	Feedwater Heater Vents and Drains System Test #31
Table 14.2-32[2]	Condenser Air Removal System Test #32
Table 14.2-33	Turbine Generator Test #33
Table 14.2-34[2]	Turbine Oil Storage System Test #34
Table 14.2-35	Liquid Radioactive Waste System Test #35
Table 14.2-36	Gaseous Radioactive Waste System Test #36
Table 14.2-37[2]	Solid Radioactive Waste System Test #37
Table 14.2-38	Chemical and Volume Control System Test #38
Table 14.2-39[2]	Boron Addition System Test #39
Table 14.2-40[2]	Module Heatup System Test #40
Table 14.2-41	Containment Evacuation System Test #41
Table 14.2-42	Containment Flooding and Drain System Test #42
Table 14.2-43	Containment System Test #43
Table 14.2-44	Control Rod Drive System Flow-Induced Vibration Test #44
Table 14.2-45	Reactor Vessel Internals Flow-Induced Vibration Test #45
Table 14.2-46	Reactor Coolant System Test #46
Table 14.2-47	Emergency Core Cooling System Test #47
Table 14.2-48	Decay Heat Removal System Test #48
Table 14.2-49[2]	In-core Instrumentation System Test #49

Commented [A2]: Confirmatory Item – Do not delete this comment until staff has verified the proposed changes described in RAI 9719 response dated 10/23/19 (ADAMS Accession No. ML19296D805) have been incorporated into the latest version of the DCA.

Abstract	Test Title
Table 14.2-50	Module Assembly Equipment Test #50
Table 14.2-51	Fuel Handling Equipment System Test #51
Table 14.2-52	Reactor Building Cranes Test #52
Table 14.2-53[2]	Process Sampling System Test #53
Table 14.2-54[2]	13.8kV [kilovolt] and Switchyard System Test #54
Table 14.2-55[2]	Medium Voltage AC [alternating current] Electrical Distribution System Test #55
Table 14.2-56[2]	Low Voltage AC Electrical Distribution System Test #56
Table 14.2-57[2]	Highly Reliable DC [direct current] Power System Test #57
Table 14.2-58[2]	Normal DC Power System Test #58
Table 14.2-59[2]	Backup Power Supply System Test #59
Table 14.2-60	Plant Lighting System Test #60
Table 14.2-61[2]	Module Control System Test #61
Table 14.2-62[2]	Plant Control System Test #62
Table 14.2-63	Module Protection System Test #63
Table 14.2-64[2]	Plant Protection System Test #64
Table 14.2-65[2]	Neutron Monitoring System Test #65
Table 14.2-66	Safety Display and Indication System Test #66
Table 14.2-67[2]	Fixed-Area Radiation Monitoring System Test #67
Table 14.2-68	Communication System Test #68
Table 14.2-70	Hot Functional Testing Test #70
Table 14.2-71[2]	Module Assembly Equipment Bolting Test #71
Table 14.2-72	Steam Generator Flow-Induced Vibration Test #72
Table 14.2-73	Security Access Control Test #73
Table 14.2-74	Security Detection and Alarm Test #74
Table 14.2-75[2]	Initial Fuel Loading Precritical Test #75
Table 14.2-76	Initial Fuel Load Test (Test #76)
Table 14.2-77	Reactor Coolant System Flow Measurement Test (Test #77)
Table 14.2-78	NuScale Power Module Temperature Test (Test #78)
Table 14.2-79	Primary and Secondary System Chemistry Test (Test #79)
Table 14.2-80	Control Rod Drive System – Manual Operation, Rod Speed, and Rod Position Indication Test (Test #80)
Table 14.2-81	Control Rod Assembly Drop Time Test (Test #81)
Table 14.2-81a	Control Rod Assembly Ambient Temperature Full-Height Drop Time Test #81A
Table 14.2-82	Pressurizer Spray Bypass Flow Test (Test #82)
Table 14.2-83	Initial Criticality Test (Test #83)
Table 14.2-84	Post-Critical Reactivity Computer Checkout Test (Test #84)
Table 14.2-85[2]	Low-Power Test Sequence Test #85
Table 14.2-86	Determination of Zero-Power Physics Testing Range Test (Test #86)
Table 14.2-87	All Rods Out Boron Endpoint Determination Test (Test #87)
Table 14.2-88	Isothermal Temperature Coefficient Measurement Test (Test #88)
Table 14.2-89	Bank Worth Measurement Test (Test #89)
Table 14.2-90[2]	Power Ascension Test #90
Table 14.2-91	Core Power Distribution Map Test (Test #91)

Commented [A3]: Confirmatory Item – Do not delete this comment until staff has verified the proposed changes described in letter dated 9/19/19 (ADAMS Accession No. ML19262G576) have been incorporated into the latest version of the DCA.

Abstract	Test Title
Table 14.2-92	Neutron Monitoring System Power Range Flux Calibration Test (Test #92)
Table 14.2-93	Reactor Coolant System Temperature Instrument Calibration Test (Test #93)
Table 14.2-94	Reactor Coolant System Flow Calibration Test (Test #94)
Table 14.2-95	Radiation Shield Survey Test (Test #95)
Table 14.2-96	Reactor Building Ventilation System Capability (Test #96)
Table 14.2-97	Thermal Expansion Test (Test #97)
Table 14.2-98	Control Rod Assembly Misalignment (Test #98)
Table 14.2-99	Steam Generator Level Control Test (Test #99)
Table 14.2-100	Ramp Change in Load Demand (Test #100)
Table 14.2-101	Step Change in Load Demand Test (Test #101)
Table 14.2-102	Loss of Feedwater Heater Test (Test #102)
Table 14.2-103	100 Percent Load Rejection Test (Test #103)
Table 14.2-104	Reactor Trip from 100 Percent Power Test (Test #104)
Table 14.2-105	Island Mode Test for NuScale Power Module #1 (Test #105)
Table 14.2-106	Island Mode Test for Multiple NuScale Power Modules (Test #106)
Table 14.2-107	Remote Shutdown Workstation Test #107
Table 14.2-108	NuScale Power Module Vibration Test (Test #108)

Commented [A4]: Confirmatory Item – Do not delete this comment until staff has verified the proposed changes described in letter dated 10/10/19 (ADAMS Accession No. ML19283E530) have been incorporated into the latest version of the DCA.

The staff confirmed that each of the test abstracts identified above from NuScale DCA Part 2, Tier 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 DC review.

Although the NRC is approving only the test abstracts listed in the table above, the staff notes that additional test abstracts that are not being reviewed by the NRC at this stage are included in DCA Part 2, Tier 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 COL applicant. Only those operational requirements completely reviewed and approved in the design certification rulemaking will be subject to the provisions of Section VIII.C of the Commission's design certification rule. Table 14.2-2 lists the tests that were not evaluated by the staff at the DC 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 initial test program, which must be addressed at the COL stage.

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

Abstract	Test Title
Table 14.2-12[3]	Potable Water System Test #12

[3] COL Item 14.2-5 states "a COL Applicant that references the NuScale Power Plant design certification will provide a test abstract for the potable water system pre-operational testing.

Abstract	Test Title
Table 14.2-69[4]	Seismic Monitoring System Test #69

14.2.1.5 Combined License Information Items

Table 1.8-2 lists COL information item numbers and descriptions related to Chapter 14.2, from DCA Part 2, Tier 2.

Table 14.2-4: NuScale Combined License Information Items for Section 14.2

Item No.	Description	DCD Tier 2 Section
COL Item 14.2-1	A COL Applicant that references the NuScale Power Plant design certification 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	A COL Applicant that references the NuScale Power Plant design certification is responsible for the development of the Startup Administration Manual that will contain the administrative procedures and requirements that control the activities associated with the Initial Test Program. The COL applicant will provide a milestone for completing the Startup Administrative Manual and making it available for NRC inspection.	14.2
COL Item 14.2-3	A COL Applicant that references the NuScale Power Plant design certification will identify the specific operator training to be conducted during low-power testing related to the resolution of TMI Action Plan Item I.G.1, as described in NUREG-0660, NUREG-0694, and NUREG-0737.	14.2
COL Item 14.2-4	A COL Applicant that references the NuScale Power Plant design certification will provide a schedule for the Initial Test Program.	14.2
COL Item 14.2-5	A COL Applicant that references the NuScale Power Plant design certification will provide a test abstract for the potable water system pre-operational testing.	14.2
COL Item 14.2-6	A COL Applicant that references the NuScale Power Plant design certification will provide a test abstract for the SMS pre-operational testing.	14.2
COL Item 14.2-7	A COL Applicant that references the NuScale Power Plant design certification will select the plant configuration to perform the Island Mode Test (number of NPMs in service).	14.2

[4] COL Item 14.2-6 states “a COL Applicant that references the NuScale Power Plant design certification will provide a test abstract for the SMS pre-operational testing.

14.2.1.6 Conclusion

The staff completed its review of the NuScale ITP at the DC stage in accordance with the requirements of 10 CFR 30.53, 10 CFR 50.43, 10 CFR 52.47, 10 CFR 50.34, 10 CFR 52.79, 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 DC 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

DCA Part 2, Tier 2, Section 14.3, "Certified Design Material and Inspections, Tests, Analyses, and Acceptance Criteria," discusses the development of Tier 1. DCA Part 2, Tier 2, Section 14.3.2, "Tier 1 Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria First Principles," describes the criteria used to identify the scope of Tier 1 design descriptions and the scope of the inspections, tests, analyses, and acceptance criteria (ITAAC). The staff notes that significant portions of DCA Tier 2, Section 14.3.2, are extracted from a Nuclear Energy Institute white paper that the NRC has not reviewed or approved. This document was submitted on June 14, 2017, and is titled, "First Principles for Use in Developing Design Certification Tier 1 Information and Inspections, Tests, Analyses, and Acceptance Criteria [ITAAC]" (ADAMS Accession No. ML17235A591). It proposes "first principles" that could be used to determine the scope of Tier 1 design descriptions and ITAAC. The staff excludes DCA Part 2, Tier 2, Section 14.3.2, from its review of this DCA and does not take a position on the "first principles" described in that section. DCA Part 2, Tier 2, Section 14.3.2 will not be incorporated by reference into a design certification rule for the NuScale design.

DCA Part 2, Tier 2, Section 14.0, "Initial Test Program and Inspections, Tests, Analyses and Acceptance Criteria," states "[t]he initial test program addresses structures, systems, and components and design features for both the nuclear portion of the facility and the balance-of-plant." The ITAAC are presented in Tier 1. The DCA Part 2, Tier 2, Section 14.2, describes the initial test program (ITP) that is performed during the initial startup of the NuScale plant. The ITP includes test activities that commence with the completion of construction and installation and end with the completion of reactor power ascension testing. The staff's review of the ITP is in Section 14.2 of this SER.

14.3.1 Selection Criteria for DCA Part 2, Tier 1

14.3.1.1 Introduction

Section 14.3 of this SER describes the staff's evaluation of the DCA Part 2, Tier 1 information for the NuScale design. This section also addresses the technical adequacy and completeness of the inspections, tests, analyses, and acceptance criteria (ITAAC) in DCA Part 2, Tier 1. The staff issued requests for additional information (RAIs) to NuScale to resolve the staff's questions on the information in the DCA submittal. In response to the RAIs, NuScale revised DCA Part 2

to clarify specific information. In this SER section, the staff focuses on how the revised DCA complies with 10 CFR 52.47(b)(1) and conforms with the applicable NRC guidance, rather than discussing each RAI and NuScale response.

The staff reviewed DCA Part 2, Tier 1 for the type of information and the level of detail discussed in NUREG-0800, Section 14.3, “Inspections, Tests, Analyses, and Acceptance Criteria,” issued March 2007, and SECY-19-0034, “Improving Design Certification Content,” dated April 8, 2019 (ADAMS Accession No. ML19080A032). As reflected in SECY-19-0034, the following general principles apply to the review of Tier 1:

1. Tier 1 should include “the top-level design features and performance characteristics” that are “the most significant to safety.”

2. Tier 1 descriptions should typically be at a qualitative and functional level of detail.

3. The level of detail is governed by a graded approach based on safety significance.

4. Tier 1 should not include detail that could necessitate NRC approval for departures from the certified design that have minimal safety significance. Nonetheless, Tier 1 should still reflect the specific safety-significant features of the design and not just include general statements that apply to classes of reactors.

5. The acceptance criteria in ITAAC should generally be “objective and unambiguous.” This can be accomplished if the acceptance criteria clearly state the functional requirement and Tier 2 describes detailed methodologies and criteria for verifying that the functional requirement has been met.

6. Numeric values in Tier 1 should be minimized. Numeric values could be used for basic design descriptions (e.g., numbers of modules, pumps, or diesel generators) or where a deviation from the value clearly indicates a failure to meet fundamental design criteria. Otherwise, specific numeric values should be only in Tier 2.

7. The use of codes and standards in Tier 1 should be minimized, as discussed in SRP Section 14.3. If a code is referenced in Tier 1, the specific edition, date, etc. should be specified in Tier 2 rather than Tier 1 to provide flexibility.

These principles are largely taken from SRP Section 14.3, but principles 2, 4, and 6 in the list above are from SECY-19-0034. Principles 2, 4, and 6 are intended to lead to a more judicious selection of information to be included in Tier 1. With respect to numeric values, SECY-19-0034 acknowledges that applicants might wish to include numeric values in Tier 1 beyond what the staff would accept as a minimum. In such cases, the staff stated that it will entertain alternatives to the staff’s revised position on numeric values. For example, structural dimensions might be retained in Tier 1 as a design goal, but construction deviations from these values would be allowed if a reconciliation analysis shows that the as-built structure still accomplishes its required functions. Other examples include the use of bounding values or appropriate tolerances.

The NuScale Tier 1 information includes the following:

- definitions and general provisions;
- design descriptions;
- ITAAC;
- significant site parameters; and
- significant interface requirements.

The applicant intends to have this Tier 1 information certified in a design certification (DC) rulemaking pursuant to Subpart B of 10 CFR Part 52, “Standard Design Certifications.” The Tier 1 design descriptions are binding requirements for the life of a facility referencing the certified design unless an exemption request is submitted and approved.

The Tier 1 design descriptions, interface requirements, and site parameters are derived from Tier 2 information. The staff’s review of how the underlying Tier 2 information satisfies the NRC’s regulations is documented throughout this SER, and these conclusions also apply to the same information included in Tier 1. Thus, for the Tier 1 design descriptions, interface requirements, and site parameters, the additional staff review is limited to addressing whether Tier 1 includes appropriate information from Tier 2.

The purpose of the ITAAC portion of the Tier 1 information is to verify that a facility referencing the DC has been constructed and will be operated in accordance with the certified design, the Atomic Energy Act of 1954, as amended (AEA), and applicable regulations. The principal performance characteristics and safety functions of the structures, systems, and components (SSCs) are verified by the appropriate ITAAC.

14.3.1.2 Summary of Application

DCA Part 2, Tier 1: The Tier 1 information is summarized below.

<u>Definitions and General Provisions</u>: The definitions and general provisions are provided in DCA Part 2, Tier 1, Sections 1.1, “Definitions,” and 1.2, “General Provisions.”

<u>Design Descriptions</u>: Design descriptions are provided in each subsection of DCA Part 2, Tier 1, Section 2.0, “Unit Specific Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria,” and Section 3.0, “Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria.” The unit-specific descriptions in Tier 1, Section 2 apply to each NuScale module, while Tier 1, Section 3 addresses SSCs that support multiple NuScale modules. Top-level design information in DCA Part 2, Tier 1 is extracted from the more detailed design information presented in DCA Part 2, Tier 2. The design description consists of the system description and design commitments. The design features in the design commitments are verified by ITAAC.

<u>ITAAC</u>: The ITAAC are provided in Sections 2.0 and 3.0 of DCA Part 2, Tier 1.

Significant Interface Requirements: The significant interface requirement is described in DCA Part 2, Tier 1, Section 4.0, “Interface Requirements,” and is associated with site-specific structures.

Significant Site Parameters: The significant site parameters are provided in DCA Part 2, Tier 1, Section 5.0, “Site Parameters.”

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 14.3, “Certified Design Material and Inspections, Tests, Analyses, and Acceptance Criteria,” discusses the development of Tier 1. DCA Part 2, Tier 2, Section 14.3.2, “Tier 1 Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria First Principles,” describes the criteria used to identify the scope of Tier 1 design descriptions and the scope of the ITAAC. DCA Part 2, Tier 2, Table 14.3-1, “Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference,” describes how the module-specific ITAAC will be performed. DCA Part 2, Tier 2, Table 14.3-2, “Shared/Common Structures, Systems, and Components and Non-Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference,” describes how the shared/common system and non-SSC ITAAC will be performed. DCA Part 2, Tier 2, Section 14.3.3 describes the organization of Tier 1 information. DCA Part 2, Tier 2 Section 14.3.4 provides information on the definitions, general provisions, acronyms, abbreviations, and figures used in Tier 1. DCA Part 2, Tier 2 Sections 14.3.5 and 14.3.6 distinguish between those design descriptions and ITAAC that apply to a specific unit or module versus those that are shared by multiple modules. DCA Part 2, Tier 2 Sections 14.3.7 and 14.3.8 discuss interface requirements and site parameters, respectively.

ITAAC: The applicant provided ITAAC tables for each of the systems listed in DCA Part 2, Tier 1, Chapters 2 and 3, that had Tier 1 design descriptions. The ITAAC are provided in Sections 2.0 and 3.0 of the DCA Part 2, Tier 1. Tier 1, Chapter 2 provides the ITAAC tables for SSCs that support a single NuScale Power Module and should be completed for each module in a multi-unit plant. Tier 1, Chapter 3 provides the ITAAC tables for SSCs that are shared by multiple NuScale Power Modules. The first column of the table proposes design commitments extracted from the Tier 1 design description that must be verified. The second and third columns identify proposed methods of verifications and acceptance criteria that demonstrate that design commitments are met.

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

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

14.3.1.3   Regulatory Basis

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

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

- 10 CFR 52.47(a)(26), which requires that a DCA provide justification that compliance with the interface requirements of 10 CFR 52.47(a)(25) is verifiable through inspections, tests, or analyses. The method to be used for verification of interface requirements must be included as part of the proposed ITAAC required by 10 CFR 52.47(b)(1).

The staff reviewed DCA Part 2, Tier 1, for the type of information and the level of detail discussed in NUREG-0800, Section 14.3 and SECY-19-0034. For specific technical areas, the staff used the acceptance criteria and additional guidance in SRP Sections 14.3.2 – 14.3.4, 14.3.6 – 14.3.9, 14.3.11, and 14.3.12, and in Section 14.3.5 of the design-specific review standard (DSRS) for the NuScale small modular reactor design, as discussed below in SER Sections 14.3.2 – 14.3.9, and 14.3.11 – 14.3.13.

In reviewing the ITAAC, the staff also considered the guidance in 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 (RG) 1.206, "Combined License Applications for Nuclear Power Plants – Light Water Reactor Edition," issued June 2007, provides COL applicants referencing a certified design with guidance on the development of site-specific ITAAC and the use of ITAAC contained in a certified design. The DCA Part 2, Tier 1 information provides the principal design bases and design characteristics that are proposed for certification by the 10 CFR Part 52 rulemaking process.

The following documents provide additional criteria or guidance in support of the SRP acceptance criteria to meet the above requirements:

- RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 3, issued March 2007.

- In letters dated April 8, 2016, and June 21, 2016, the staff transmitted to NuScale a set of Standardized DCA ITAAC (ADAMS Accession Nos. ML16096A121 and ML16160A179) that could be used in its DCA.

14.3.1.4 Technical Evaluation

14.3.1.4.1 DCA Part 2, Tier 2

DCA Part 2, Tier 2, Section 14.3.1 provides an introduction and lists two COL information items, which are discussed in Section 14.3.1.5. DCA Part 2, Tier 2, Sections 14.3.3 through 14.3.6 describe the content and organization of Tier 1 information and includes information on the definitions, general provisions, acronyms, abbreviations, and figures used in Tier 1. The staff reviewed the information in these sections and finds it consistent with guidance in SRP Section 14.3 and is acceptable. The staff notes that the applicant organized the Tier 1 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 14.3 of this report are not an evaluation of their corresponding sections in DCA Part 2, Tier 2.

14.3.1.4.2 DCA Part 2, Tier 1

14.3.1.4.2.1 Definitions, General Provisions, Design Descriptions, and ITAAC

In accordance with SRP Section 14.3, DCA Part 2, Tier 1, information should identify the principal performance characteristics and safety functions of the standard design. 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 required by 10 CFR 52.47(b)(1), the proposed ITAAC must be necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses (ITA) are performed and the acceptance criteria (AC) met, a facility that incorporates the DC has been constructed and will operate in accordance with the DC, the provisions of the AEA, and the NRC's rules and regulations.

For the ITAAC to be "sufficient" as required by 10 CFR 52.47: (1) the ITA must clearly identify those activities necessary to demonstrate that the AC are met; (2) the AC must state clear design or performance objectives demonstrating that the Tier 1 design commitments are satisfied; (3) the ITA and AC must be consistent with each other and the Tier 1 design commitment; (4) the ITAAC must be capable of being performed and satisfied prior to fuel load; and (5) the ITAAC, as a whole, must provide reasonable assurance that, if the ITAAC are satisfied, the facility has been constructed and will be operated in accordance with the DC, 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 DCA Part 2, Tier 1, and the staff's review of whether the Tier 1 descriptions have the type of information and the level of detail discussed in NUREG-0800, Section 14.3 and SECY-19-0034.

The staff conducted a review of DCA Part 2, Tier 1 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 Tier 1 definitions, general provisions, and design descriptions are acceptable. Furthermore, the staff concludes that (1) the ITA and AC are consistent with each other and with the design commitment; (2) the ITA are clearly stated; and (3) the AC are clear and objective.

Commented [A5]: Confirmatory Item – Do not delete this comment until staff has verified the proposed changes described in RAI 9681 response dated 10/10/19 (ADAMS Accession No. ML19283E530) have been incorporated into the latest version of the DCA

As evaluated in Section 17.4, "Reliability Assurance Program," of this SER, the applicant fully described the design reliability assurance program (D-RAP). In accordance with SECY-18-0093, "Recommended Change to Verification of the Design Reliability Assurance Program" (ADAMS Accession No. ML18192B471), and its associated Staff Requirements Memorandum (SRM) dated August 7, 2019 (ADAMS Accession No. ML19219A944), a DC applicant is not required to propose ITAAC for the D-RAP.

14.3.1.4.2.2 Site Parameters

The staff evaluates Tier 1 site parameters in Chapter 2 of this SER.

14.3.1.4.2.3 Interface Requirements

DCA Part 2, Tier 1 Section 4.1, "Site-Specific Structures," states "[f]ailure of any of the site-specific structures not within the scope of the NuScale Power Plant certified design will not cause any of the Seismic Category I structures within the scope of the NuScale Power Plant-certified design to fail." The staff's evaluation of this interface requirement and the basis for the staff's finding that they meet 10 CFR 52.47(a)(25) can be found in Section 3.7.2, "Seismic System Analysis," of this SER. ITAAC Number 7 in DCA Part 2, Tier 1, Table 3.11-2 and ITAAC Number 5 in DCA Part 2, Tier 1, Table 3.13-1 verify that as-built non-Seismic-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). These ITAAC are evaluated in Section 14.3.2 of this SER. Based on the acceptability of these ITAAC, the staff finds that the provisions of 10 CFR 52.47(a)(26) have been met.

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 DCA Part 2, Tier 2, Table 1.8-2, "Combined License Information Items." COL Item 14.3-1 is evaluated in Section 13.3.5 of this SER. Regarding COL Item 14.3-2 the staff agrees that it is the COL applicant's responsibility to provide the site-specific selection methodology and ITAAC for site-specific SSCs.

Table 14.3.1-1 NuScale Combined License Information Items for Section 14.3.1

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

14.3.1.6 Conclusion

The staff concludes that the requirements of 10 CFR 52.47(a)(26) are met because acceptable ITAAC have been provided to verify the Tier 1 interface requirement.

The staff concludes that the form and clarity of the Tier 1 definitions, general provisions, and design descriptions are acceptable. Furthermore, the staff concludes that (1) the ITA and AC are consistent with each other and with the design commitment; (2) the ITA are clearly stated; and (3) the AC 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 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 design certification has been constructed and will operate in accordance with the design certification, the AEA, and the NRC's rules and regulations.

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

14.3.2.1 Introduction

This section reviews ITAAC and Tier 1 design descriptions applicable to structural and systems engineering. The following DCA Part 2, Tier 1 tables contain the ITAAC applicable to this review area:

- Table 3.11-2, "Reactor Building ITAAC," Numbers 6 and 7**.**
- Table 3.12-2, "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 RXB, radioactive waste building (RWB), and 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 Table 3.11-2 and ITAAC Number 5 in Table 3.13-1 is to verify that as-built non-Seismic-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 an SSE.

14.3.2.2 Summary of Application

See Section 14.3.1.2 of this SER.

14.3.2.3 Regulatory Basis

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

14.3.2.4 Technical Evaluation

The staff reviewed the ITAAC information in DCA Part 2, Tier 1 and Tier 2 described in Section 14.3.2.1 of this SER. The staff reviewed the structural design descriptions for reactor, radioactive waste, and control building structures in DCA Part 2, Tier 1, Sections 3.6, 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 Tier 1.

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 a DCA include the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will operate in accordance with the DC, the AEA, and NRC regulations. The as-built structures need to be reconciled under the actual design basis loads to demonstrate that their structural integrity is maintained, that the in-structure responses are enveloped by those in the approved design, and that a potential adverse interaction between as-built non-Seismic-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 a SSE. A design summary report should document that the Seismic Category I structures meet acceptance criteria specified in Sections 3.7 and Section 3.8 of the UFSAR. The staff's review of the ITAAC against these criteria is documented in the following subsections.

14.3.2.4.2 ITAAC for Structural Integrity of Safety-Related Structures

The staff's review focus is on whether ITAAC confirm that post-construction, the design parameters used in the design certification are not exceeded and that the deviations between the as-built structure and the certified design are reconciled to demonstrate that the deviations from the certified design made during construction are within the design code limits. 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 certification has been constructed and will be operated as required.

The staff reviewed the structural integrity ITAAC for the RXB, RWB, and CRB structures identified in DCA, Part 2, Tier 1, as 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, respectively. These ITAAC state that, based on inspection, reconciliation analysis will be performed of the as-built RXB, RWB, and CRB structures. The ITAAC acceptance criteria states 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 response for the as-built buildings are enveloped by those in the approved design. The applicant in DCA, Part 2, Tier 2, Section 3.8.4.5.1 provided further details on the design summary report:

> *A Design Summary Report is prepared that documents the results of a reconciliation analysis of the cumulative effect of changes between the approved design and the actual design basis loads and as-built structural components to demonstrate that (1) the computed demand continues to be within the capacity of the structural component and (2) the as-built in-structure seismic response is enveloped by the in-structure seismic response in the approved design.*
>
> *The Design Summary Report documents that the Seismic Category I structures meet the acceptance criteria specified in Section 3.7 and Section 3.8.*

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Deviations from the design are tracked as required by 10 CFR Part 50, Appendix B, and are evaluated consistent with the methods and procedures of Section 3.7 and Section 3.8. Deviations include changes outside applicable tolerances in load, dimension, and configuration between the approved design and as-built structure. 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 results of these evaluations will be documented in the Design Summary Report.

The staff also reviewed the corresponding discussions of these ITAAC in DCA Part 2, Tier 2, Table 14.3-2 and finds that the Tier 1 information is based on and is consistent with the Tier 2 information. Based on the applicant's description of the design summary report for as-built structures, the staff concludes that the ITAAC for the RXB, RWB, and CRB 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 SSCs/non-Seismic-Category-I SSCs

The applicant provided ITAAC to verify that as-built non-Seismic-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 an SSE. These ITAAC are identified in DCA, Part 2, Tier 1, as ITAAC Number 7 in Table 3.11-2, and ITAAC Number 5 in Table 3.13-1 for the RXB and the CRB, respectively. These ITAAC along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-2, conform to the Standardized DCA ITAAC design commitments and associated Tier 2 discussion in the staff's April 8, 2016, letter. Also, the staff finds that the Tier 1 design descriptions and ITAAC are based on and consistent with the Tier 2 material. Therefore, the staff finds the ITAAC Number 7 in Table 3.11-2 for the RXB and ITAAC Number 5 in Table 3.13-1 for the CRB are sufficient to verify that as-built Seismic Category I structures are protected from adverse seismic interaction with non-Seismic-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 certified RXB, RWB, and CRB structures.

14.3.2.6 Conclusion

The staff reviewed design descriptions for the RXB, RWB, and CRB structures in DCA Part 2, Tier 1 Sections 3.6, 3.11, 3.12, and 3.13, and finds that the design descriptions are consistent with the enclosure to SECY-19-0034 covering the level of structural information in Tier 1. The staff also reviewed ITAAC Numbers 6 and 7 in Table 3.11-2, ITAAC Number 3 in Table 3.12-2, ITAAC Numbers 4 and 5 in Table 3.13-1, and the ITAAC background discussion in DCD Tier 2, Table 14.3.2 to ensure that the final as-built condition of these structures is reconciled to conform to the approved design basis for the structural design. The staff concludes that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the NuScale design certification has been constructed and will be operated in

accordance with the design certification, 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 Tier 1 design descriptions applicable to piping and components. The following DCA Part 2, Tier 1, tables contain the ITAAC applicable to this review area:

- Table 2.1-4, “NuScale Power Module ITAAC,” Numbers 1–6.
- Table 2.1-4, “NuScale Power Module ITAAC,” Numbers 12–15.
- Table 2.1-4, “NuScale Power Module ITAAC,” Numbers 18–21 and 26.
- Table 2.2-3, “Chemical and Volume Control System ITAAC,” Numbers 1, 2, 3, and 5.
- Table 2.8-2, “Equipment Qualification ITAAC,” Numbers 1, 3, 6, and 7.
- Table 3.1-2, “Control Room Habitability System ITAAC,” Numbers 2 and 3.
- Table 3.5-1, “Fuel Storage System ITAAC,” Number 1.
- Table 3.6-2, “Ultimate Heat Sink Piping System ITAAC,” Number 1 and 2.
- Table 3.11-2, “Reactor Building ITAAC,” Number 8.
- Table 3.14-2, “Equipment Qualification—Shared Equipment ITAAC,” 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. They are 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’s Tier 1 and Tier 2 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 is discussed 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 DCA Part 2 associated with this review area is consistent with the guidance in SRP Section 14.3.3. The staff has also reviewed the contents of DCA Part 2, Tier 1 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 American Society of Mechanical Engineers (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 Reactor Vessel Materials

SER Section 5.3.1 discusses the staff's review of reactor vessel materials. In DCA Part 2, Tier 1, Table 2.1-4, ITAAC Number 6, for Charpy testing of reactor pressure vessel beltline material, and ITAAC Number 12, for reactor pressure vessel surveillance capsule holders, follow the standardized ITAAC guidance provided to NuScale. As these ITAAC and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1 satisfy the guidance and contain wording previously found acceptable by the staff, these ITAAC and the associated Tier 2 discussion are found to be acceptable.

14.3.3.4.2 Generic Piping Design

14.3.3.4.2.1 Piping Stress Analysis

SER Section 3.12 discusses the staff's review of the piping stress analysis. ITAAC Number 1 in DCA Part 2, Tier 1, Tables 2.1-4, 2.2-3, and 3.6-2 require 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 the DCA Part 2, Tier 2, Table 14.3-1 discussion for ITAAC Number 1 in DCA Part 2, Tier 1, Table 2.1-4, 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 DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2 for compliance with ASME BPV Code, Section III, are consistent with the standardized ITAAC guidance provided to NuScale. 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 DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2 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.

14.3.3.4.2.2 Pipe Break Analysis

The staff's review of pipe break analysis is discussed in Section 3.6.2 of this SER.

The ITAAC for protection of safety-related SSCs from dynamic and environmental effects (DCA Part 2, Tier 1, Table 2.1-4, Number 4) is in the section for the NuScale Power Module 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 3.6-3 and an additional ITAAC (DCA Part 2, Tier 1, Table 3.11-2, Number 8). The staff has confirmed that the full scope of the plant area for the certified design where pipe breaks may be postulated is covered by ITAAC.

The applicant’s proposed ITAAC, located in ITAAC tables for the NuScale Power Module and the reactor building, and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2 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 associated Tier 2 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.

14.3.3.4.2.3   Leak-Before-Break Evaluation

The staff’s review of LBB analysis is discussed in Section 3.6.3 of this SER. During the course of this review, no ITAAC issues were identified. The applicant’s proposed ITAAC for LBB analysis; specifically, ITAAC Number 5 in DCA Part 2, Tier 1, Table 2.1-4, and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1 is consistent with the standardized ITAAC guidance. Therefore, based on the technical review conducted in SER Section 3.6.3, which determines the technical adequacy of the applicant’s LBB analysis methodologies, and the applicant’s use of proposed ITAAC and associated Tier 2 discussion consistent with the standardized ITAAC guidance, the staff finds this ITAAC and associated discussion acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for LBB analysis.

14.3.3.4.2.4   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 the DCA Part 2, Tier 2, Table 14.3-1 discussion for ITAAC Number 1 in DCA Part 2, Tier 1 Table 2.1-4, 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 per 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.

The ITAAC Number 1 in DCA Part 2, Tier 1, Tables 2.1-4, 2.2-3, and 3.6-2 verify 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 Tier 2 discussion are acceptable, as discussed in SER Section 14.3.3.4.2.1. ITAAC Numbers 2 and 3 in DCA Part 2, Tier 1, Table 2.1-4; ITAAC Number 2 in DCA Part 2, Tier 1, Table 2.2-3; ITAAC Number 1 in DCA Part 2, Tier 1,

Table 3.5-1; and ITAAC Number 2 in DCA Part 2, Tier 1, Table 3.6-2 verify that the ASME BPV Code Class 1, 2, 3, NF, 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, NF, and CS components and interconnecting piping. The applicant's proposed ITAAC and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2 for as-built reconciliation are consistent with the standardized ITAAC guidance. As these ITAAC and associated Tier 2 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.

14.3.3.4.3 Verifications of Components and Systems

14.3.3.4.3.1 Piping and Component Safety Classification

The staff's review of piping and component safety classification is discussed in Section 3.2.2 of this SER. The safety classification of piping and components is a topic that is resolved during the design certification phase, with the exception of any site-specific elements, which will be reviewed during the review 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 piping and component safety classification in order to meet the requirements of 10 CFR 52.47(b)(1). The safety classifications assigned in the DCA 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.

14.3.3.4.3.2 Fabrication (Welding)

The topic of welding is covered in this section 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 14.3.3.4.2.4 of this report, there are no additional issues identified for this review area.

14.3.3.4.3.3 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 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 14.3.3.4.2.4 of this report, there are no additional issues identified for this review area.

14.3.3.4.3.4 Environmental and Seismic and Dynamic Qualification of Equipment

The staff's review of seismic and dynamic qualification of equipment is discussed in Section 3.10 of this SER. The staff's review of environmental qualification of mechanical and electrical equipment is discussed in Section 3.11 of this SER.

ITAAC Number 1 in DCA Part 2, Tier 1, Table 2.8-2, and ITAAC Number 1 in DCA Part 2, Tier 1, Table 3.14-2, 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 DCA Part 2, Tier 1, Table 2.8-1, "Module Specific Mechanical and Electrical/I&C Equipment," and DCA Part 2, Tier 1, Table 3.14-1, "Mechanical and Electrical/Instrumentation and Controls Shared Equipment." The staff confirmed that the Tier 1 tables contain the required seismic Category I SSCs. The ITAAC and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2 ensure that SSCs listed in the Tier 1 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 Tier 2 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 associated discussion acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for the seismic and dynamic qualification of SSCs.

The applicant proposed ITAAC Number 3 in DCA Part 2, Tier 1, Table 2.8-2, for the environmental qualification of nonmetallic parts, materials, and lubricants used in safety-related mechanical equipment. The proposed ITAAC and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1 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 Tier 2 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.

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14.3.3.4.3.5 Valve Qualification

The NRC staff's review of valve qualification is discussed in Section 3.9.6 of this SER.

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 Power Plant to be appropriate as a DCA Part 2, Tier 1, requirement that should not be modified without prior NRC review. The staff requested the applicant to specify the process for qualification of safety-related valves (ADAMS Accession No. ML17307A452). In its responses to RAI 9131, Question 14.03.03-6, dated December 27, 2017, and May 24, 2018 (ADAMS Accession Nos. ML17361A136 and ML18144A918), the applicant explained the intent of the term, "Qualification Report," as used in ITAAC Number 6 in DCA Part 2, Tier 1, Table 2.8-2; namely, that use of the term, "Qualification Report," refers to the ASME QME-1 Qualification Report, as defined in QR-4000 of ASME Standard QME-1-2007, "Qualification of Active Mechanical Equipment Used in Nuclear Power Plants," which requires adherence to the provisions of ASME QME-1-2007 in order to satisfy the report requirements. The staff has confirmed that DCA Part 2, Tier 1 and Tier 2 includes the clarification. The staff finds that the wording of ITAAC Number 6 in DCA Part 2, Tier 1, Table 2.8-2 and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1, 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 DCA Part 2, Tier 1, Table 2.8-2, and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1 is acceptable for meeting the requirements of 10 CFR 52.47(b)(1) for the functional qualification of safety-related valves.

The staff reviewed DCA Part 2, Tier 1, Section 2.8, "Equipment Qualification," to ensure it included all applicable safety-related valves and that ITAAC Number 6 in DCA Part 2, Tier 1, Table 2.8-2 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 DCA Part 2, Table 14.3-1 is consistent with the standardized ITAAC guidance. Because the applicant includes ITAAC and associated Tier 2 discussions, for the equipment qualification of all safety-related valves in all environments and their applicable fluid conditions, that is consistent with the standardized ITAAC guidance, the staff finds the proposed ITAAC and associated discussion sufficient for the equipment qualification of safety-related valves.

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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 DCA Part 2, Tier 1, Table 2.1-4), emergency core cooling system valves (ITAAC Number 14 in Table 2.1-4), decay heat removal system (DHRS) valves (ITAAC Number 15 in Table 2.1-4), CNTS check valves (ITAAC Number 21 in Table 2.1-4), chemical and volume control system (CVCS) air-operated valves (ITAAC Number 3 in DCA Part 2, Tier 1, Table 2.2-3), and control room habitability system valves (ITAAC Number 2 in DCA Part 2, Tier 1, Table 3.1-2). The proposed ITAAC and associated discussion in DCA Part 2, Tier 2 Table 14.3-1 and Table 14.3-2 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 Tier 2 discussion listed 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.

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 DCA Part 2, Tier 1, Table 2.1-4), emergency core cooling system reactor recirculation valves and reactor vent valves (ITAAC Number 19 in Table 2.1-4), DHRS hydraulic-operated valves (ITAAC Number 20 in Table 2.1-4), CVCS air-operated valves (ITAAC Number 5 in DCA Part 2, Tier 1, Table 2.2-3), and control room habitability system solenoid-operated valves (ITAAC Number 3 in DCA Part 2, Tier 1, Table 3.1-2). The proposed ITAAC and associated discussion in DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2, 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 Tier 2 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.

NuScale also proposed ITAAC Number 7 in DCA Part 2, Tier 1, Table 2.8-2, to demonstrate the set pressure, capacity, and overpressure design requirements for safety-related relief valves. The proposed ITAAC and associated discussion in DCA Part 2, Tier 2, Table 14.3-1, 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 Tier 2 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.

NuScale included ITAAC Number 26 in DCA Part 2, Tier 1, Table 2.1-4, to verify the installation of the ECCS, CIV, and DHRS actuation valves and their associated hydraulic lines. The staff has reviewed the proposed ITAAC and the associated discussion in DCA Part 2, Tier 2, Table 14.3-1, and finds it acceptable for meeting the requirements of 10 CFR 52.47(b)(1) 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 DCA Part 2, Tier 1, information in the NuScale DCA Part 2 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) by proposing ITAAC that are sufficient to provide reasonable assurance that the piping systems and components have been constructed and installed in accordance with the design certification and will be operated in conformity with the design certification, the provisions of the AEA, and the Commission’s rules and regulations. The staff also concludes that the applicant has included sufficient, top-level design information in Tier 1, 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 Tier 1 design descriptions applicable to reactor systems. The following DCA Part 2, Tier 1 table contains the ITAAC that is reviewed in this section:

- Table 2.8-2, “Equipment Qualification ITAAC,” Number 8**.**

Also, technical evaluation Subsection 14.3.4.4 references additional ITAAC for reactor systems that are evaluated elsewhere in this report. Reactor systems piping and mechanical ITAAC are addressed in Subsections 14.3.3.4.3.4 and 14.3.3.4.3.5 of this report. Component electrical ITAAC are addressed in Subsection 14.3.3.4.3.4 and automatic reactor trip functions, engineered safety functions, and manual switches ITAAC are addressed in Section 14.3.5.4 of this report. 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

The staff makes the following overall conclusions regarding the reactor systems information in Tier 1. Consistent with SRP Section 14.3.4 and SECY-19-0034, the Tier 1 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 DCA Part 2, Tier 2, sections and concludes that the Tier 1 design description and ITAAC are based on and consistent with Tier 2 material. The reactor systems ITAAC, along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2, generally conform to the standardized DCA ITAAC, design commitments, and associated Tier 2 discussions in the NRC's April 8, 2016, letter.

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 DCA.

The staff's review of the reactor systems' ITAAC are presented here, listed in order of the associated DCA Part 2, Tier 2, section.

14.3.4.4.1 Fuel Assembly Design (DCA Part 2, Tier 2, Section 4.2)

For this section, there are no proposed ITAAC designated in DCA Part 2, Tier 1, Section 2.9 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 report.

14.3.4.4.2 Control Rod Drive System (DCA Part 2, Tier 2, Section 4.6)

DCA Part 2, Tier 1, Tables 2.1-4 and 2.8-2 provide ITAAC for control rod drive system (CRDS) piping and components as defined in Tables 2.1-1, 2.1-2, 2.1-3, and 2.8-1. The as-built piping and mechanical components must comply with ASME Code, Section III, requirements, and electrical equipment must perform their operational function as described in the ITAAC. DCA Part 2, Tier 1, Table 2.5-7 provides module protection system (MPS) ITAAC for the CRDS related to automatic reactor trip functions, engineered safety functions, and manual switches as defined in Tables 2.5-1, 2.5-2, and 2.5-3. The staff reviewed the Tier 1 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. Each ITAAC identified above, and the associated Tier 1 design descriptions and Tier 2, Section 14.3 material, are evaluated in other subsections of Section 14.3 of this report as noted above in Section 14.3.4.1.

14.3.4.4.3 Overpressure Protection System (DCA Part 2, Tier 2, Section 5.2.2)

DCA Part 2, Tier 1, Tables 2.1-4 and 2.8-2 provide ITAAC for overpressure protection system mechanical and electrical equipment as defined in Tables 2.1-2, 2.1-3, and 2.8-1. The as-built mechanical equipment must comply with ASME Code, Section III, requirements, and electrical equipment must perform their operational function as described in the ITAAC. DCA Part 2, Tier 1, Table 2.5-7 provides MPS ITAAC relating to the overpressure protection system's automatic engineered safety functions and manual switches as defined in Tables 2.5-2 and 2.5-3. The staff reviewed the Tier 1 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. Each ITAAC identified above, and the associated Tier 1 design descriptions and Tier 2, Section 14.3 material, are evaluated in other subsections of Section 14.3 of this report as noted above in Section 14.3.4.1.

14.3.4.4.4 Decay Heat Removal System (DCA Part 2, Tier 2, Section 5.4.3)

DCA Part 2, Tier 1, Tables 2.1-4 and 2.8-2 provide ITAAC for decay heat removal system (DHRS) piping and mechanical and electrical equipment as defined in Tables 2.1-1, 2.1-2, 2.1-3, and 2.8-1. The as-built piping and mechanical equipment must comply with ASME Code, Section III, requirements, and electrical equipment must perform their operational function as described in the ITAAC. DCA Part 2, Tier 1, Table 2.5-7 provides MPS ITAAC for the DHRS related to the automatic reactor trip functions, engineered safety functions, and manual switches as defined in Tables 2.5-1, 2.5-2, and 2.5-3. These ITAAC, and the associated Tier 1 design descriptions and Tier 2, Section 14.3 material, are evaluated in other subsections of Section 14.3 of this report as noted above in Section 14.3.4.1.

ITAAC Number 8 in Table 2.8-2 verifies that the DHRS condensers have the capacity to transfer their design heat load. The staff used the acceptance criteria in SRP Sections 14.3 and 14.3.11, "Containment Systems—Inspections, Tests, Analyses, and Acceptance Criteria," and the ITAAC-related guidance in RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Revision 0, Sections C.1.14.3 and C.II.1.2.11, to evaluate the ITAAC. The staff finds the ITAAC acceptable because the DHRS heat removal capability is credited in Chapter 15 transient analyses for mitigation of non-loss-of-coolant accident design basis events (DBEs). 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.

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The staff finds that the DHRS-related ITAAC items mentioned above are necessary and sufficient to verify the DCA Part 2, Tier 1 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 ITAAC are performed and the acceptance criteria are met, the as-built top-level design parameters described in DCA Tier 1 would be in conformity with the certified 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 DCA Tier 1, and the Tier 1 information 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 (DCA Part 2, Tier 2, Section 5.4.4)

The reactor coolant system 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. DCA Part 2, Tier 1, Tables 2.1-4 and 2.8-2 provide ITAAC for piping and mechanical and electrical equipment as defined in Tables 2.1-1, 2.1-2, 2.1-3, and 2.8-1. The as-built piping and mechanical equipment must comply with ASME Code, Section III, requirements, and electrical equipment must perform their operational function as described in the ITAAC. The staff reviewed the Tier 1 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. Each ITAAC identified above, and the associated Tier 1 design descriptions and Tier 2, Section 14.3 material, are evaluated in other subsections of Section 14.3 of this report as noted above in Section 14.3.4.1.

14.3.4.4.6 Emergency Core Cooling System (DCA Part 2, Tier 2, Section 6.3)

DCA Part 2, Tier 1, Tables 2.1-4 and 2.8-2 provide ITAAC for emergency core cooling system (ECCS) piping and components as defined in Tables 2.1-2, 2.1-3, and 2.8-1. The as-built mechanical components must comply with ASME Code, Section III, requirements, and electrical equipment must perform their operational function as described in the ITAAC. DCA Part 2, Tier 1, Table 2.5-7 provides MPS ITAAC for the ECCS related to the automatic engineered safety functions and manual switches as defined in Tables 2.5-2 and 2.5-3. The staff reviewed the Tier 1 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. Each ITAAC identified above, and the associated Tier 1 design descriptions and Tier 2, Section 14.3 material, are evaluated in other subsections of Section 14.3 of this report as noted above in Section 14.3.4.1.

14.3.4.4.7 Chemical and Volume Control System (DCA Part 2, Tier 2, Section 9.3.4)

DCA Part 2, Tier 1, Tables 2.2-3 and 2.8-2, provide ITAAC for CVCS piping and components as defined in Tables 2.2-1, 2.2-2, and 2.8-1. The ASME Code Class 3 as-built piping and isolation valves connected to the reactor pressure vessel (RPV) must comply with ASME Code, Section III, requirements. The staff reviewed the Tier 1 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. Each ITAAC identified above, and the associated Tier 1 design descriptions and Tier 2, Section 14.3 material, are evaluated in other subsections of Section 14.3 of this report as noted above in Section 14.3.4.1.

14.3.4.5 Combined License Information Items

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

14.3.4.6 Conclusion

The staff reviewed the DCA Part 2, Tier 1, 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 systems SSCs are appropriately

described in Tier 1, and the Tier 1 information is acceptable. In addition, the staff finds that ITAAC Number 8 for the DHRS in DCA Part 2, Tier 1, Table 2.8-2 complies with 10 CFR 52.47(b)(1). The staff's conclusions regarding mechanical, electrical, and I&C aspects of reactor systems ITAAC are discussed in other subsections of Section 14.3 of this report 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 Tier 1 design descriptions applicable to instrumentation and controls (I&C). The NuScale I&C-related ITAAC are listed in the following DCA Part 2, Tier 1 tables:

- Table 2.5-7, "Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 1, 3, 4, 5, 7, 11, 13, 16, 17, 22, 23, 24, 25, and 26.

- Table 2.6-1, "Neutron Monitoring Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 1, 2, and 3.

- Table 2.8-2, "Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria," Numbers 4–5.

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

The staff prepared this SER section based on the information provided in DCA Part 2, Tier 1, Sections 2.5 and 2.6, and DCA Part 2, Tier 2, Section 14.3. Additionally, the staff reviewed the ITAAC in Tier 1, Section 2.8, Table 2.8-2, Numbers 4 and 5. The staff reviewed this Tier 1 information against the NuScale DSRS Section 14.3.5 acceptance criteria and SECY-19-0034.

Overall Evaluation of I&C-Related Information in Tier 1

Based on the review of the I&C-related information in DCA Part 2, Tier 2 Chapter 7, the staff concludes the following regarding I&C-related information in Tier 1:

- Consistent with NuScale DSRS Section 14.3.5 and SECY-19-0034, the Tier 1 design descriptions 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 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 assures 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 back-up 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 assure compliance with each NRC requirement identified in DSRS Chapter 7. 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 are discussed in greater detail below.

- The Tier 1 design descriptions and ITAAC are based on and consistent with the Tier 2 material.

The staff also evaluated whether Tier 1 design descriptions and ITAAC were needed from an I&C perspective for active systems. Based on the I&C design information provided in Tier 2 Chapter 7, the staff finds that no active systems are needed for reactor-coolant makeup or decay heat removal and therefore no Tier 1 design description or ITAAC is required from an I&C perspective.

14.3.5.4.1 Module Protection System and Safety Display and Indication System ITAAC

In DCA Part 2, Tier 1, Table 2.5-7, the applicant provided ITAAC verifying design features for the module protection system (MPS) and its associated components in the safety display and indication system (SDIS) provided in DCA Part 2, Tier 1, Section 2.5.1, "Design Description." Section 2.5.1 states:

> *The MPS is comprised of the reactor trip system (RTS) and the engineered safety features actuation system (ESFAS). The RTS is responsible for monitoring key variables and shutting down the reactor when specified limits are reached. The ESFAS is responsible for monitoring key variables and actuating the engineered safety features (ESF) such as the emergency core cooling system (ECCS) and the decay heat removal system (DHRS) when specified limits are reached.*

These ITAAC, along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-1, “Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference,” for ITAAC 02.05.01, 02.05.03, 02.05.04, 02.05.05, 02.05.07, 02.05.11, 02.05.13, 02.05.16, 02.05.17, 02.05.22, 02.05.23, 02.05.24, 02.05.25 and 02.05.26, generally conform to the Standardized DCA ITAAC, design commitments, and associated Tier 2 discussion in the staff’s April 8, 2016, letter on standardized ITAAC for a DCA (ADAMS Accession No. ML16096A121). The 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 DCA Part 2, Tier 1, Section 2.5.1.

The Tier 1 review of the remote shutdown station capabilities is discussed in Section 14.3.9, “Human Factors Engineering – Inspections, Tests, Analyses, and Acceptance Criteria,” of this SER.

14.3.5.4.2 Neutron Monitoring System ITAAC

The design description in DCA Part 2, Tier 1, Section 2.6, “Neutron Monitoring System,” states the following:

> *The NMS monitors the neutron flux level of the reactor core by detecting neutron leakage from the core. The NMS measures neutron flux as an indication of core power and provides safety-related inputs to the module protection system.*

There is no ITAAC to verify the capability of the as-built Neutron Monitoring System (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 staff finds that the ITAAC in Tier 1, Table 2.6-1, along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-1 for ITAAC 02.06.01 through 02.06.03, conform to the Standardized DCA ITAAC, design commitments, and associated Tier 2 discussion in the staff’s April 8, 2016, letter on standardized ITAAC for a DCA.

Based on the above, the NRC staff finds that the ITAAC for the NMS in DCA Part 2, Tier 1, 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 DCA Part 2, Tier 1, Section 2.8, “Equipment Qualification.” Section 2.8 states the following:

> *The Class 1E computer-based instrumentation and control systems listed in Table 2.8-1 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 2.8-1 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 DCA Part 2, Tier 1, Table 2.8-2, Numbers 4 and 5, along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-1 for ITAAC Numbers 02.08.04 and 02.08.05, conform to the Standardized DCA ITAAC, design commitments, and associated Tier 2 discussion in the staff's April 8, 2016, letter. 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, radio frequency interference, and electrical surge.

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Based on the above, the staff finds that the ITAAC for equipment qualification of the safety-related digital I&C in DCA Part 2, Tier 1, Table 2.8-2, Numbers 4 and 5, comply with 10 CFR 52.47(b)(1).

14.3.5.5 Combined License Information Items

There are no COL information items listed in DCA Part 2 Tier 2, Table 1.8-2, "Combined License Information Items," for this area of review.

14.3.5.6 Conclusion

The staff finds that the DCA Part 2, Tier 1, 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 Tier 1 design content. The staff also finds that the description of how to complete the I&C ITAAC in DCA Part 2, Tier 2, Table 14.3-1 is acceptable.

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

14.3.6.1 Introduction

This section reviews ITAAC and Tier 1 design descriptions applicable to electrical systems. The following DCA Part 2, Tier 1 tables contain the ITAAC applicable to this review area:

- Table 2.1-4, "NuScale Power Module ITAAC," Numbers 10 and 22.
- Table 2.8-2, "Equipment Qualification ITAAC," Numbers 2 and 9.
- Table 3.8-1, "Plant Lighting System ITAAC," Numbers 1–3.
- Table 3.14-2, "Equipment Qualification—Shared Equipment ITAAC," 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 DCA Part 2, Tier 1, and DCA Part 2, Tier 2, related to the electrical power system to ensure, in part, that DCA Part 2, Tier 1, 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 1.206, Sections C.II.1.2.6 and C.II.1-A. The ITAAC review documented in this SER section is limited to the ITAAC listed in Section 14.3.6.1 of this SER and the discussion of these ITAAC in DCA Part 2, Tier 2, Section 14.3, Tables 14.3-1 and 14.3-2. The staff reviewed whether meeting the ITAAC verifies that the DCA Part 2, Tier 1, 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 DCA to determine whether the applicant established appropriate Tier 1 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 (1) equipment qualification (EQ) for seismic and harsh environment, (2) containment electrical penetrations, and (3) lighting, 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 EQ for seismic and harsh environments should verify that the seismic design requirement of GDC 2 and the EQ requirements of 10 CFR 50.49 are met. Specifically, the design description should identify 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 14.3.3 of this SER.

The staff reviewed the DCA Part 2, Tier 1, Sections 2.8 and 3.14, design descriptions, which address the most safety-significant features for equipment qualification. Sections 2.8 and 3.14 describe the module-specific and common equipment that would be subject to equipment qualification. The staff determined that Tier 1 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 consistent with the Tier 2 material.

The staff reviewed DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2, which provides background information associated with the Tier 1 design commitments for ITAAC Numbers 02.08.02, 02.08.09, and 03.14.02, and a discussion of how to complete the ITAAC. The staff determined that this DCA Part 2, Tier 2 information is consistent with the NuScale design and

ITAAC in DCA Part 2, Tier 1, and provides appropriate information to implement the ITAAC successfully.

Section 3.11 of this SER contains the staff's evaluation of DCA Part 2, Tier 2, 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 DCA Part 2, Tier 1, Table 2.8-2, and ITAAC Number 2 in DCA Part 2, Tier 1, Table 3.14-2, which verify that the Class 1E equipment located in a harsh environment is qualified and meet the EQ requirements of 10 CFR 50.49. These ITAAC, along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2, conform to the Standardized DCA ITAAC, design commitments, and associated Tier 2 discussion in the staff's April 8, 2016, letter. Therefore, the staff finds that these ITAAC are necessary, sufficient, and meet the requirements of 10 CFR 52.47(b)(1).

14.3.6.4.1.2 Containment Electrical Penetrations

Consistent with SRP Section 14.3.6, the ITAAC for containment electrical penetrations should verify that all the penetrations (both Class 1E and non-Class 1E circuits) are protected against postulated fault currents (that is, currents greater than the continuous current rating) so that the electrical faults do not breach the containment.

Section 8.3.1 of this SER contains the staff's evaluation of DCA Part 2, Tier 2, Section 8.3.1.2.5, "Containment Electrical Penetration Assemblies," which describes the electrical design requirements for electrical penetration assemblies.

The staff reviewed the DCA Part 2, Tier 1, Section 3.1, design descriptions, which address the most safety-significant features of the NuScale Power Module (NPM). This section describes the systems contained within the boundary of the NPM, the safety-related and nonsafety-related functions that are performed by the NPM and verified by the ITAAC. The staff determined that Tier 1 design descriptions and ITAAC relating to containment electrical penetrations adequately describe the top-level, most safety-significant design features that are based on and consistent with the Tier 2 material.

The staff reviewed DCA Part 2, Tier 2, Table 14.3-1, which provides background information associated with the Tier 1 design commitments for ITAAC Numbers 02.01.10 and 02.01.22, and a discussion of how to complete the ITAAC. The staff determined that this DCA Part 2, Tier 2, information is consistent with the NuScale design and ITAAC in DCA Part 2, Tier 1, and provides appropriate information to implement the ITAAC successfully.

The staff reviewed ITAAC Numbers 10 and 22 in DCA Part 2, Tier 1, Table 2.1-4, which verify that the containment electrical penetrations are protected against postulated fault currents. These ITAAC, along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-1, conform to the Standardized DCA ITAAC, design commitments, and associated Tier 2 discussion in the NRC's April 8, 2016, letter. Therefore, the staff finds that these ITAAC are necessary, sufficient, and meet the requirements of 10 CFR 52.47(b)(1).

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14.3.6.4.1.3 Lighting

Consistent with SRP Section 14.3.6, the ITAAC for lighting should verify the continuity of power sources for plant lighting systems to ensure that portions of the plant lighting remain available during accident scenarios and power failures. The basis for inclusion may be more related to defense in-depth, support function, operating experience, or PRA rather than "accomplishing a direct safety function."

The staff reviewed the DCA Part 2, Tier 1, Section 3.8, design descriptions, which address the most safety-significant features of the plant lighting system (PLS). The PLS is a nonsafety-related system that provides artificial illumination for the entire plant: buildings (interior and exterior), rooms, spaces, and all outdoor areas of the plant. The staff determined that Tier 1 design descriptions and ITAAC adequately describe the top-level, most safety-significant design features that are based on and consistent with the Tier 2 material.

The staff reviewed DCA Part 2, Tier 2, Table 14.3-2, which provides background information associated with the Tier 1 design commitments for ITAAC Numbers 03.08.01, 03.08.02, and 03.08.03, and a discussion of how to complete the ITAAC. The staff determined that this DCA Part 2, Tier 2, information is consistent with the NuScale design and ITAAC in DCA Part 2, Tier 1, and provides appropriate information to implement the ITAAC successfully.

Section 9.5.3 of this SER discusses and evaluates DCA Part 2, Tier 2, Section 9.5.3, "Lighting Systems," to determine whether the plant lighting levels are adequate during all plant operating conditions, and whether the lighting systems can operate without adversely impacting the operation, control, and maintenance of SSCs. The NuScale plant lighting system includes normal plant lighting, emergency plant lighting, and normal and emergency main control room lighting.

The staff reviewed ITAAC Numbers 1–3 in DCA Part 2, Tier 1, Table 3.8-1, which verify that portions of the plant lighting remain available during accident scenarios and power failures. These ITAAC, along with the corresponding discussions in DCA Part 2, Tier 2, Table 14.3-2, conform to the Standardized DCA ITAAC, design commitments, and associated Tier 2 discussion in the NRC's April 8, 2016 letter. Therefore, 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

DCA Tier 2, 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 with relevant NRC guidance in SRP Section 14.3.6. The staff finds that the NuScale Tier 1 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). Also, the staff concludes that the Tier 1 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. The staff also concludes that the information associated with electrical systems in DCA Part 2, Tier 2, Tables 14.3-1 and 14.3-2, are consistent with the NuScale design and ITAAC.

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

14.3.7.1 Introduction

This section reviews ITAAC and Tier 1 design descriptions related to most of the plant systems that are not part of the core reactor systems. The following DCA Part 2, Tier 1, tables contain the ITAAC applicable to this review area:

- Table 2.1-4, “NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 24–25.
- Table 2.3-1, “Containment Evacuation System ITAAC,” Numbers 1–2.
- Table 3.1-2, “Control Room Habitability System ITAAC,” Numbers 1, 4, and 5.
- Table 3.2-2, “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 1–6.
- Table 3.5-1, Fuel Storage System ITAAC,” Number 2.
- Table 3.6-2, “Ultimate Heat Sink Piping System ITAAC,” Number 3.
- Table 3.7-1, “Fire Protection System ITAAC,” Numbers 1–4.
- Table 3.10-1, “Reactor Building Crane ITAAC,” Numbers 1–7 and Numbers 9–10.
- Table 3.11-2, “Reactor Building ITAAC,” Numbers 1–2.
- Table 3.13-1, “Control Building ITAAC,” Numbers 1–2.

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 DCA Part 2, Tier 1 and Tier 2, the staff makes the following overall conclusions regarding the plant systems information in Tier 1. Consistent with SRP Section 14.3.7 and SECY-19-0034, the Tier 1 design descriptions and ITAAC adequately describe the top-level design features and performance characteristics that are significant to safety. The safety significance of the plant systems and how they are treated in Tier 1 is discussed in detail below. The staff reviewed the design description and system ITAAC to confirm completeness and consistency with the system design basis as described in various DCA Part 2, Tier 2, sections and conclude the Tier 1 design description and ITAAC are based on and consistent with Tier 2 material. 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 DCA Part 2, Tier 2, Table 14.3-1 and Table 14.3-2 generally conform to the standardized DCA ITAAC, design commitments, and associated Tier 2 discussions in the NRC's April 8, 2016, letter.

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 design certification application.

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

14.3.7.4.1 Internal Flood Protection for Onsite Equipment Failures (DCA Part 2, Tier 2, Section 3.4.1)

The ITAAC associated with internal flooding barriers in the reactor building (RXB) and control building (CRB) are found in DCA Part 2, Tier 1, Table 3.11-2, Number 2, and Table 3.13-1, Number 2, respectively. These ITAAC ensure barriers, including flood-resistant doors, curbs and sills, walls, watertight penetration seals, and National Electrical Manufacturers Association (NEMA) enclosures exist and are qualified in accordance with the internal flooding analysis to provide confinement so that the impact from an internal flood in the RXB or CRB is contained within the flooding area of origin.

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) (DCA Part 2, Tier 2, Section 3.5.1.1)

In DCA Part 2, Tier 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 plant and system design, there are no credible missiles that could affect SSCs important to safety. Upon reviewing Tier 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 DCA

Part 2, Tier 2, Section 3.5.1.1. Turbine generator missiles are evaluated in DCA Part 2, Tier 2, Section 3.5.1.3 and addressed below in SER Section 14.3.7.4.4.

14.3.7.4.3 Internally Generated Missiles (Inside Containment) (DCA Part 2, Tier 2, Section 3.5.1.2)

In DCA Part 2, Tier 2, Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)," it is stated that the NuScale power modules use a steel containment that encapsulates the reactor pressure vessel. The applicant also states that there is no rotating equipment inside containment and all pressurized components are ASME 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 (DCA Part 2, Tier 2, Section 3.5.1.3)

DCA Part 2, Tier 2, Section 3.5.1.3, "Turbine Missiles," describes an approach which includes building wall barriers, system redundancy, and defense-in-depth features to protect essential SSCs from turbine missiles. These essential SSCs are located in the RXB and CRB. DCA Part 2, Tier 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," and "[e]ssential SSC in the CRB are located below grade and are protected by the CRB exterior wall and grade-level slab." The staff agrees with the assessment that the combined effect of these provisions provides reasonable assurance of protection to the essential SSCs in the RXB and CRB.

DCA Tier 1, Table 3.11-2, ITAAC Number 6 verifies RXB structural integrity under design basis loads and DCA Tier 1, Table 3.13-1, ITAAC Number 4 verifies the CRB structural integrity under design loads at CRB elevation 120'-0" and below. The staff finds that these ITAAC are sufficient to verify that the RXB and CRB have been designed and constructed to withstand turbine missiles loads without loss of overall structural integrity. These ITAAC are evaluated in SER Section 14.3.2.

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14.3.7.4.5 Missiles Generated by Tornadoes and Extreme Winds (DCA Part 2, Tier 2, Section 3.5.1.4)

In its review of the information in Tier 1, the staff found that ITAAC in Table 3.11-2 and Table 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. DCA Tier 1, Table 3.11-2, ITAAC Number 6 verifies RXB structural integrity under design basis loads, which as indicated in DCA Tier 2, Table 14.3-2 includes missile impact loads. DCA Tier 1, Table 3.13-1, ITAAC Number 4 verifies the CRB structural integrity under design loads at CRB elevation 120'-0" and below. 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. These ITAAC are evaluated in SER Section 14.3.2.

14.3.7.4.6 Structures, Systems, and Components To Be Protected from External Missiles (DCA Part 2, Tier 2, Section 3.5.2)

In DCA Part 2, Tier 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 Tier 1, the staff found that ITAAC Number 6 in DCA Part 2, Tier 1, Table 3.11-2, verifies RXB structural integrity under design basis loads, and ITAAC Number 4 in DCA Tier 1, Table 3.13-1, verifies CRB structural integrity under design loads at CRB elevation 120'-0" and below. 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. These ITAAC are evaluated in SER Section 14.3.2.

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

DCA Part 2, Tier 1, 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 DCA Part 2, Tier 1, Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," 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 they are installed in accordance with the as-built Pipe Break Hazard Analysis Report and 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 SER Section 14.3.3.4.2.2, "Pipe Break Analysis."

14.3.7.4.8 Reactor Coolant Pressure Boundary Leakage Detection (DCA Part 2, Tier 2, Section 5.2.5)

DCA Part 2, Tier 1, Table 2.3-1, includes ITAAC Numbers 1 and 2 for reactor coolant system (RCS) leakage detection. The ITAAC require tests to verify the design of the RCS leakage detection systems. These tests include (1) verifying 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 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 1 gpm within 1 hour. This is consistent with the guidance in RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," and the SRP.

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The staff reviewed the proposed ITAAC and find 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 (DCA Part 2, Tier 2, Section 9.1.2)

The staff reviewed DCA Part 2, Tier 1, Section 3.5, "Fuel Storage System," which contains the specific ITAAC for the fuel storage system. It describes the high-level features of the fuel storage system design and 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.” DCA Part 2, Tier 1, 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 DCA Part 2, Tier 1 (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 an NRC letter to the applicant containing draft standard ITAAC to be considered for the DCA (ADAMS Accession No. ML16096A121). 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), and drain down prevention is evaluated in Section 14.3.7.4.13, “Ultimate Heat Sink,” of this SER.

14.3.7.4.10 Spent Fuel Pool Cooling and Cleanup System (DCA Part 2, Tier 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. The pool is cooled by passive means using the volume of water in the combined reactor pool and SFP. ITAAC Number 3 in DCA Part 2, Tier 1, Table 3.6-2 for the UHS addresses verification that sufficient cooling water is available for design-basis events. This ITAAC is discussed in section 14.3.7.4.13 of this report.

14.3.7.4.11 Fuel Handling Equipment (DCA Part 2, Tier 2, Section 9.1.4)

DCA Part 2, Tier 1, 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 DCA Part 2, Tier 1. DCA Part 2, Tier 1, Table 3.4-1, Numbers 1–6, present the FHE system ITAAC.

The staff reviewed the proposed ITAAC and finds them acceptable because they will verify that that FHE has been constructed in accordance with the ASME NOG-1 Code and will have sufficient load-carrying capability and limits on travel to assure that it has been constructed and will be operated in conformity with the design certification.

In addition, the majority of occupational radiation exposure typically occurs during refueling outages, with exposure to plant personnel from the movement of irradiated fuel and in-core components being a potentially significant contributor to this dose. Furthermore, the plant should be designed with appropriate radiation protection design features during potential accident conditions, in accordance with 10 CFR Part 50, Appendix A, GDC 61, “Fuel storage and handling and radioactivity control.” In DCA Part 2, Tier 1, Table 3.4-1, design commitment and ITAAC Number 5 require that the fuel-handling machine travel is limited so that the machine maintains at least 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 is a design feature provided for maintaining a dose of less than 2.5 millirem (mrem)/hour radiation

exposure to operators on the refueling platform in accordance with the American National Standards Institute (ANSI)/American Nuclear Society (ANS)-57.1-1992, “Design Requirements for Light Water Reactor Fuel Handling Systems,” and 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 (DCA Part 2, Tier 2, Section 9.1.5)

DCA Part 2, Tier 1, Section 3.10, provides a general overview of reactor building crane (RBC) and the associated ITAAC. DCA Part 2, Tier 1, Table 3.10-1, Numbers 1–7 and 9–10, include the RBC ITAAC.

Tier 1 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 (OHLHS) equipment to be a significant design feature to include in DCA Part 2, Tier 1. DCA Part 2, Tier 1, Table 3.10-1, provides ITAAC Numbers 1, 2, and 3 for verification that the RBC main hoist, two auxiliary hoists, and the wet hoist, respectively, contain single-failure-proof design features. ITAAC Numbers 4, 5, and 6, respectively, provide for a load test of at least 125 percent of the hoist rated capacity for the three listed hoists. Also, ITAAC Number 7 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.

In addition, the staff noted that DCA Part 2, Tier 1, Section 2.1, “NuScale Power Module,” includes information for load-carrying structural members attached to the containment vessel (CNV). Similarly, DCA Part 2, Tier 1, Section 3.10, includes information for the module lifting adapter (MLA), which is used together with the RBC to lift and transport the NPM during its refueling outage. The NuScale design credits these load-carrying structural components, in part, for an assumed low failure rate of the RBC when the RBC is used to transport the NPM. DCA Part 2, Tier 1, Sections 2.1 and 3.10, includes ITAAC requirements for the NPM lifting fixture that is welded to the CNV and the MLA, respectively. The staff finds these ITAAC acceptable because the tests will adequately confirm the design features in the OHLHS that are credited for an assumed low failure rate of the RBC when the RBC is used to transport the NPM.

DCA Part 2, Tier 1, Table 3.10-1, ITAAC Number 9 will provide for a load test on the MLA single and dual load-path elements, and ITAAC Number 10 will verify that the MLA contains single-failure-proof design features. The staff finds these ITAAC acceptable and complete because they are consistent with ASME NOG-1 provisions for a Type I crane. DCA Part 2, Tier 1, Table 2.1-4, ITAAC Number 24 will verify that the NPM lifting fixture supports its rated load, and ITAAC Number 25 will verify that it contains single-failure-proof design features. The staff finds these ITAAC acceptable because they are consistent with ANSI-N14.6, “Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 KG) or More.”

Based on the above evaluation, the staff finds that the OHLHS 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 (DCA Part 2, Tier 2, Section 9.2.3)

The demineralized water system (DWS) is a nonsafety-related system and is not required for mitigation of any DBE. While DWS operation is not required or credited in any DBE, in its review of the demineralized water system, 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 chemical and volume control system (CVCS). Design and operation of the demineralized water system isolation valve is covered by the ITAAC in DCA Part 2, Tier 1, Section 2.2, “Chemical and Volume Control System.” DCA, Part 2, Tier 1, Table 2.2-3, ITAAC Numbers 3 and 5 verify proper operation of the demineralized water isolation valves. These ITAAC are evaluated in Section 14.3.3 of this SER. 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, and therefore, the staff concluded that no additional ITAAC are necessary.

14.3.7.4.14 Ultimate Heat Sink (DCA Part 2, Tier 2, Section 9.2.5)

DCA Part 2, Tier 1, Table 3.6-2, provides ITAAC for UHS piping and connections. The ASME Code Class 3 as-built piping system makeup line to the SFP must comply with ASME Code, Section III, requirements. 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.

DCA Part 2, Tier 1, Table 3.6-2, specifies the ITAAC for the UHS. ITAAC Number 3 contains a design commitment that spent fuel pool, 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 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 meet the requirements in 10 CFR 52.47(b)(1).

14.3.7.4.15 Equipment and Floor Drain Systems (DCA Part 2, Tier 2, Section 9.3.3)

In DCA Part 2, Tier 1, Table 3.17-2, “Radiation Monitoring Inspections, Tests, Analyses, and Acceptance Criteria for NuScale Power Modules 1-6,” ITAAC Number 2, and DCA Part 2, Tier 1, Table 3.18-2, “Radiation Monitoring Inspections, Tests, Analysis, and Acceptance Criteria for NuScale Power Modules 7-12,” ITAAC Number 2 verify that, upon initiation of a high radiation signal, the balance-of-plant drainage system automatically aligns or actuates the identified components to the positions identified in DCA Part 2, Tier 1, Table 3.17-1, “Radiation Monitoring – Automatic Actions for NuScale Power Modules 1 – 6,” and DCA Part 2, Tier 1, Table 3.18-1, “Radiation Monitoring – Automatic Actions for NuScale Power Modules 7 – 12,” respectively. SER Section 14.3.8 evaluates these ITAAC.

14.3.7.4.16 Fire Protection System (DCA Part 2, Tier 2, Section 9.5.1)

In DCA Part 2, Tier 1, Table 3.7.1, "Fire Protection System Inspections, Tests, Analysis, and Acceptance Criteria," the applicant provided ITAAC verifying the design features for the fire protection system provided in DCA Part 2, Tier 1, Section 3.7.1, "Design Description." Section 3.7.1 states:

> *The FPS is comprised of the equipment and components that provide early fire detection and suppression to limit the spread of fires. The FPS is a nonsafety-related system that supports up to 12 NuScale Power Modules (NPMs). The FPS equipment is located throughout the plant site.*

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 500 gpm for fire hoses assuming failure of the largest fire pump or loss of off-site 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 DCA Part 2, Tier 1, Table 3.11-2, "Reactor Building Inspections, Tests, Analysis, and Acceptance Criteria," the applicant provided ITAAC Number 1 verifying that fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the reactor building fire area of origin.

In DCA Part 2, Tier 1, Table 3.13-1, "Control Building Inspections, Tests, Analysis, and Acceptance Criteria," the applicant provided ITAAC Number 1 verifying that fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the control building 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 in DCA Part 2, Tier 1, Sections 3.7.1, 3.11.1, and 3.13.1. 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 (DCA Part 2, Tier 2 Section 10.3)

The ITAAC for the portions of the safety-related SSCs of the Main Steam System (MSS) are presented as Numbers 1, 2, 3, 6, 7, and 9 in DCA Part 2, Tier 1, Table 2.8-2, "Equipment Qualification Inspections, Tests, Analyses, and Acceptance Criteria."

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The staff's review for the MSS Tier 1 information included review of descriptive information; safety-related functions; mechanical, I&C and electric power design features; environmental qualification; as well as system and equipment performance requirements. The staff's review of ITAAC Numbers 1, 3, 6, and 7 is in Section 14.3.3 of this SER. The staff's review of ITAAC Numbers 2 and 9 is 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 (DCA Part 2, Tier 2, Section 10.4.7)

There are no ITAAC for the entire Condensate and Feedwater System (CFWS) shown in Tier 1; however, in DCA, Part 2, Tier 1, Section 2.8, "Equipment Qualification," the applicant proposes ITAAC for the following CFWS equipment: the feedwater supply check valves, the FWIV, and the feedwater regulating valve (FWRV). DCA, Part 2, Tier 1, Table 2.8-2, "Equipment Qualification Inspection, Tests, Analyses, and Acceptance Criteria," provides ITAAC Number 6 for testing/accepting these valves. The staff's review of this ITAAC is in Section 14.3.3 of this SER. The staff finds that they provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification has been built and will be operated in accordance with the applicable portions of the design certification, the AEA, and the NRC's regulations as required by 10 CFR 52.47(b)(1).

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14.3.7.4.19 Control Room Habitability System (DCA Part 2, Tier 2, Section 6.4)

The staff reviewed the following ITAAC requirements in DCA, Tier 1, Tables 3.1-2, "Control Room Habitability System Inspection, Tests, analyses, and Acceptance Criteria," ITAAC Numbers 1, 4, and 5. ITAAC Number 2 and 3 in Table 3.1-2 are evaluated in Section 14.3.3 of this SER.

ITAAC 1 on Control room envelope (CRE) air exfiltration test: Tracer gas testing will be performed to verify the CRE leakage rate is not exceeded.

ITAAC 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 DBA.

ITAAC 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 DBA 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 DCA Part 2, Tier 1 Section 3.1. 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).

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

The staff reviewed the following ITAAC in DCA Part 2, Tier 1, Table 3.2-2:

- ITAAC Number 1—Test that the control room heating, ventilation, and air conditioning 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—Test and verify that the CRVS maintains a positive pressure in the CRB relative to the outside environment.

- ITAAC Number 3—Verify that the hydrogen concentration levels in the CRB battery rooms are below 1 percent by volume. This is consistent with Institute of Electrical and Electronics Engineers Standard 484-2002, as revised by RG 1.128, “Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants,” Revision 2, 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, “Combined License Applications for Nuclear Power Plants (LWR Edition),” dated June 2007, 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 (DCA Part 2, Tier 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 DCA Part 2, Tier 1, Table 3.3-1:

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

- ITAAC Number 2—Test to verify that the RBVS maintains a negative pressure in the radioactive waste building (RWB) relative to the outside environment.

- ITAAC Number 3—Test 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. DCA Part 2, Tier 1, 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 in Numbers 1 and

2 in DCA Part 2, Tier 1, Table 3.3-1, to be in accordance with SRP Section 14.3.8, in that the applicant provides 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 (DCA Part 2, Tier 2, Section 9.4.3)

DCA Part 2, Tier 1, Section 3.3, includes ITAAC Number 2 that addresses verification of the capability of the reactor building HVAC system to maintain a negative pressure in the RWB relative to the outside environment. The staff finds this to be acceptable for the reactor building HVAC system as discussed above.

14.3.7.4.23 Systems Not Requiring ITAAC

In DCA Part 2, Tier 2, the applicant indicated that the NuScale 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.

1. Reactor Component Cooling Water System (DCA Part 2, Tier 2, Section 9.2.2).
2. Potable and Sanitary Water Systems (DCA Part 2, Tier 2, Section 9.2.4).
3. Condensate Storage Facilities (DCA Part 2, Tier 2, Section 9.2.6).
4. Site Cooling Water System (DCA Part 2, Tier 2, Section 9.2.7).
5. Chilled Water System (DCA Part 2, Tier 2, Section 9.2.8).
6. Utility Water System (DCA Part 2, Tier 2, Section 9.2.9).
7. Compressed Air Systems (DCA Part 2, Tier 2, Section 9.3.1).
8. Turbine Building Ventilation System (DCA Part 2, Tier 2, Section 9.4.4).
9. Turbine Generator (DCA Part 2, Tier 2, Section 10.2).
10. Main Condenser (DCA Part 2, Tier 2, Section 10.4.1).
11. Condenser Air Removal System (DCA Part 2, Tier 2, Section 10.4.2).
12. Turbine Gland Sealing System (DCA Part 2, Tier 2, Section 10.4.3).
13. Turbine Bypass System (DCA Part 2, Tier 2, Section 10.4.4).
14. Circulating Water System (DCA Part 2, Tier 2, Section 10.4.5).

15. Auxiliary Boiler System (DCA Part 2, Tier 2, Section 10.4.10).

14.3.7.5 Combined License Information Items

There are no COL information items listed in DCA Part 2 Tier 2, Table 1.8-2, “Combined License Information Items,” 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 certification, 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 in Tier 1, consistent with SECY-19-0034, and that the DCA Part 2, Tier 2 is consistent with the Tier 1 information.

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

14.3.8.1 Introduction

This section reviews ITAAC and Tier 1 design descriptions applicable to radiation protection. The following DCA Part 2, Tier 1, tables contain the ITAAC applicable to this review area:

- Table 2.7-2, “Radiation Monitoring—Module-Specific ITAAC,” Numbers 1–2.
- Table 3.3-1, “Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 1–2.
- Table 3.4-1, “Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria,” Number 5.
- Table 3.9-2, “Radiation Monitoring—NuScale Power Modules 1–12 ITAAC,” Numbers 1–4, 7, 8, and 10.
- Table 3.11-2, “Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria,” Numbers 4–5.
- Table 3.12-2, “Radioactive Waste Building ITAAC,” Numbers 1–2.
- Table 3.14-2, “Equipment Qualification – Shared Equipment ITAAC,” Number 3.
- Table 3.17-2, Radiation Monitoring ITAAC for NuScale Power Modules 1–6,” Numbers 1–2.
- Table 3.18-2, “Radiation Monitoring ITAAC for NuScale Power Modules 7–12,” Numbers 1–2.

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:

- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criterion (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 5 rem total effective dose equivalent (TEDE) 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, “Fuel storage and handling and radioactivity control,” 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 TEDE of 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 50 rem.

- 10 CFR 20.1406, “Minimization of contamination,” as it relates to applicants for standard design certifications describing in the application how 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 shall 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 main control room containment radiation intensity (high level).

SRP Sections 14.3.7, “Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria,” and 14.3.8, “Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria,” provide acceptance criteria and additional guidance for this review area.

14.3.8.4 Technical Evaluation

The scope of the radiation protection Tier 1 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); and

- Design features for radiation protection.

For each of the ITAAC discussed below, the staff reviewed the information contained in DCA Part 2, Tier 2, Tables 14.3-1 and 14.3-2, and verified it contained information consistent with the ITAAC.

14.3.8.4.1 Radiation Shielding

SRP Section 14.3.8 indicates that the criteria in DCA Part 2, Tier 1, 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). SRP Section 14.3.8 also specifies that the review should ensure that DCA Part 2, Tier 1, clearly describes the systems, structures, and components (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, DCA Part 2, Tier 2, Section 12.3.2, “Shielding,” describes some of the design considerations for radiation shielding, such as stating that material used for a significant portion of plant shielding is to be concrete. DCA Part 2, Tier 2, Section 12.3.2.2, “Design Considerations,” states that the selection of shielding materials considers the ambient environment and potential degradation mechanisms. The material used for a significant portion of plant shielding is concrete. In addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. DCA Part 2, Tier 2, Section 12.3.2.4.3, “Reactor Building,” states that cubicle walls are concrete supported by carbon steel plates, called structural steel partition walls. DCA Part 2, Tier 2, Table 12.3-6, “Reactor Building Shield Wall Geometry,” provides the nominal thickness of concrete for some of the walls in the reactor building (RXB). DCA Part 2, Tier 2, Table 12.3-8, “Reactor Building Radiation Shield Doors,” lists the shielded doors located in the RXB. DCA Part 2 Tier 2, Table 12.3-7, “Radioactive Waste Building Shield Wall Geometry,” provides the nominal thickness of concrete for some of the walls in the radioactive waste building (RWB). DCA Part 2, Tier 2, Table 12.3-9, “Radioactive Waste Building Radiation Shield Doors,” lists the shielded doors located in the RWB.

DCA Part 2, Tier 1, Section 3.11, states that the RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding. DCA Part 2, Tier 1, Table 3.11-2, also contains the ITAAC for the RXB. Specifically, ITAAC Number 4 in Table 3.11-2 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. In addition, ITAAC Number 5 in Table 3.11-2 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 wall within which they are installed.

DCA Part 2, Tier 1, Section 3.12, states that the RWB includes radiation shielding barriers for normal operation and postaccident radiation shielding. Further, DCA Part 2, Tier 1, Table 3.12-2, contains the ITAAC for the RWB. Specifically, ITAAC Number 1 in Table 3.12-2 verifies radiation shielding and ITAAC Number 2 in Table 3.12-2 verifies the radiation attenuation doors; the RWB ITAAC are analogous to the ITAAC for the RXB.

In addition, Tier 2, Table 14.3-2 provides a cross reference between the ITAAC and the Tier 2 information. Table 14.3-2, Items 03.11.04 and 03.12.01 provide information regarding the ITAAC for the radiation shielding barriers. 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, Principal Design Criteria (PDC) 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. Table 14.3-2, Items 03.11.04 and 03.12.01 also clarify that an ITAAC inspection is performed of the RXB and RWB radiation barriers to verify wall materials and to verify that material thicknesses are as provided in DCA Part 2, Tier 2, Tables 12.3-6 and 12.3-7 and that a report will conclude that attenuation capabilities are greater than or equal to the approved design.

Commented [A17]: Confirmatory Item – Do not delete this comment until staff has verified the proposed changes described in RAI 9681 response dated 10/10/19 (ADAMS Accession No. ML19283E530) have been incorporated into the latest version of the DCA.

The staff reviewed the Tier 1 and Tier 2 information on radiation shielding barriers and radiation attenuation doors discussed above. In addition to concrete, the Tier 2 Table 12.3-6 and 12.3-7 information includes borated polyethylene shielding on the bioshield faceplate for neutron

shielding. Tier 2, Table 12.3-7 also specified that the equivalent of a 1-inch-thick steel plate is modeled to cover the liquid radioactive waste system (LRWS) ion exchange and granulated activated charcoal skid, and an equivalent 2-inch-thick plate of steel to cover the drum dryer skid cubicles. Table 12.3-6 specifies that the 20 inches of concrete and steel partition walls within the RXB consist of two one-half-inch steel plates with 19 inches of concrete in between them. The 20-inch concrete and steel composite slabs for floors and ceilings also consist of two one-half-inch steel plates with 19 inches of concrete in between them. The staff also noted that the ITAAC ensures that the radiation attenuation capability of the radiation barriers is equivalent to that provided by the materials and thicknesses specified in DCA Part 2, Tier 2, Tables 12.3-6 and 12.3-7, as revised at the time of the ITAAC completion, and the doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed. 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 using the Tier 2 change process 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 Tier 2 information at the time of ITAAC completion. Since the ITAAC verifies that the radiation attenuation capability is the same as specified in Tier 2, at the time of ITAAC completion, the staff finds the ITAAC for the shielding barriers and doors and supporting Tier 1 and Tier 2 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 main control room. The staff reviewed DCA Part 2, Tier 1, Section 2.5, “Module Protection Systems and Safety Display and Indication System,” and ITAAC Number 25 in Table 2.5-7, “Module Protection Systems and Safety Display and Indication System ITAAC.” 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 criteria is, “The PAM Type B and Type C displays listed in Table 2.5-5 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 safety display and information system (SDIS) in the main control room (MCR). The staff evaluated this information and concludes that the DCA 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. This ITAAC is also discussed in Section 14.3.5 of this SER.

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 LRWS, gaseous radioactive waste system (GRWS), and SRWS. 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.

DCA Part 2, Tier 1, 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 containment evacuation system (CES) and chemical and volume control system 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 tests the functions of the CES and chemical and volume control system monitors, as described in DCA Part 2, Tier 1, Table 2.7-1, "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.

DCA Part 2, Tier 1, 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 12 NPMs. These design commitments and ITAAC require that the control room HVAC system (CRVS), control room habitability system (CRHS), RBVS, GRWS, LRWS, auxiliary boiler system (ABS), and pool surge control system (PSCS) monitors 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 CRVS, CRHS, RBVS, GRWS, LRWS, ABS, and the PSCS monitors, as described in DCA Part 2, Tier 1, Table 3.9-1, "Radiation Monitoring – NuScale Power Modules 1-12 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.

DCA Part 2, Tier 1, Section 3.17, 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 NPMs 1–6. These design commitments and ITAAC require that the containment flooding and drains system (CFDS) and balance-of-plant drains system (BPDS) monitors 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 CFDS and BDPS monitors, as described in DCA Part 2, Tier 1, Table 3.17-1, "Radiation Monitoring – Automatic Actions for NuScale Power Modules 1-6," to initiate the desired actions on high-radiation signals to demonstrate the monitors' ability to mitigate radioactive releases, as required by the design commitments.

DCA Part 2, Tier 1, Section 3.18, 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 NPMs 7–12. These design commitments and ITAAC require that the CFDS and BPDS monitors 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 CFDA and BDPS monitors, as described in DCA Part 2, Tier 1, Table 3.18-1, "Radiation Monitoring – Automatic Actions for NuScale Power Modules 7-12," 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 above ITAAC documented in NuScale's DCA Part 2, Tier 1, the staff reviewed information related to CES monitoring in relation to the ITAAC in DCA Part 2, Tier 1, Section 2.3, "Containment Evacuation System." The staff identified that the ITAAC related to the test for the reactor coolant system (RCS) pressure boundary leakage did not include a test for the CES radiation monitor. In discussions with NuScale, they highlighted three methods the NuScale design uses to detect leakage: containment pressure, CES sample tank level, and radiation monitoring. The applicant stated that only two methods (containment pressure and CES sample tank level) are used to quantify RCS leakage. The third method, the CES radiation monitoring, is used as indication of RCS leakage on a high-radiation condition but is not used to measure the amount of RCS leakage. Indication and alarms initiated by the radiation monitor prompt operators to quantify RCS leakage using the other two methods.

In addition to the NuScale DCA Part 2, Tier 1 information, 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. Based on the number of pressure and level channels available to quantify RCS leakage and the required actions associated with losing RCS leakage indication, the staff has determined that NuScale's position of using the CES radiation monitoring for indication of a leak only, and not quantification, is acceptable. Therefore, an ITAAC to test the CES radiation monitor is unnecessary.

Based on the discussion above, the staff finds that the information provided in DCA Part 2, Tier 1, Sections 2.7, 3.9, 3.12, 3.17, and 3.18, is complete and consistent with the plant design basis as described in DCA Part 2, Tier 2, Sections 11.2, "Liquid Waste Management System," 11.3, "Gaseous Waste Management System," 11.4, "Solid Waste Management System," and 11.5, "Process and Effluent Radiation Monitoring Instrumentation and Sampling System." Based on the discussion above, the staff finds that Tier 1 includes the top-level design requirements for the PERMISS and the ITAAC for the PERMISS are acceptable and comply with the requirements of 10 CFR 52.47(b)(1).

DCA Part 2, Tier 1, Section 3.14, Table 3.14-2, Number 3, contains an ITAAC ensuring that the RWB will be designed as RW-IIa in accordance with RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." As discussed in Tier 2, Chapters 3 and 11, the radwaste building is an RW-IIa structure. Therefore, it is appropriate to include an ITAAC for the radwaste building which verifies that the as-built radwaste building maintains its structural integrity of RW-IIa under the design basis loads. In addition, Tier 1, Section 3.14 indicates that the RW-IIa components and piping used for processing gaseous radioactive waste listed in

Table 3.14-1 are constructed to the standards of RW-IIa. Table 3.14-1 lists the degasifiers, guard beds, and decay beds, including associated piping and components up to and including the first isolation valves as being designed to RW-IIa. ITAAC Number 3 in Table 3.14-2, requires a report demonstrating that the as-built RW-IIa components associated with processing gaseous radwaste (i.e. the degasifiers and guard and decay beds, including the piping associated with those components, up to and including the first isolation valves) meet the RW-IIa design criteria. Tier 2, Table 14.3-2, provides more detail regarding the basis and scope of the ITAAC. It specifies that the scope of the ITAAC are 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 radwaste components, these gaseous radwaste system components are the radwaste 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 radwaste 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 the spent resin storage tanks (RW-IIa), phase separator tanks (RW-IIb) and low conductivity waste collection tanks (RW-IIb), and associated components, contained higher quantities of radioactive material, the potential of an uncontrolled release from those components is low because these components contained slurry and/or liquid waste and were located underground in the radwaste building, in their own individual cubicles, which are stainless steel-lined up to a cubicle wall height equivalent to the full tank volume. Furthermore, the low conductivity waste collection tanks are designed with the discharge and drain lines at the lowest point of the tank, and the on/off bottom valve is a minimum distance from the tank bottom to optimize drainage and cleaning capability. 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 Tier 2 specifies that all of the radwaste SSCs are designed in accordance with RG 1.143, the staff determined that the only items requiring ITAAC were those of the gaseous radwaste system, as described above. As a result, the staff found 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 DCA Part 2, Tier 1 design information and ITAAC for radiation protection SSCs, which it credited for demonstrating that a plant incorporating the NuScale design certification satisfies the relevant requirements of 10 CFR Part 20, 10 CFR Part 50, and 10 CFR Part 52. 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 have been constructed and will be

operated in accordance with the design certification, the AEA, and NRC rules and regulations in compliance with 10 CFR 52.47(b)(1). The staff reviewed the information contained in DCA Part 2, Tier 2, Tables 14.3-1 and 14.3-2, and verified it contained information consistent with the Tier 1 ITAAC information reviewed in this section. The staff also concludes that the applicant has included sufficient, top-level design information in Tier 1, consistent with SECY-19-0034.

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

14.3.9.1 Introduction

This section reviews ITAAC and Tier 1 design descriptions applicable to human factors engineering (HFE). The following DCA Part 2, Tier 1, table contains the ITAAC applicable to this review area:

- Table 3.15-1, "Human Factors Engineering ITAAC," Number 1.

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 reviewed the information in DCA Part 2, Tier 1, Section 3.15, related to the human factors engineering design process in accordance with the guidance in SRP Section 14.3.9 and SECY-19-0034. The staff finds that the Tier 1 design description adequately describes the top-level objectives for the applicant's human factors engineering program design process and that Tier 1 includes appropriate information from Tier 2.

The staff also reviewed DCA Part 2, Tier 1, Section 3.15, Table 3.15-1, which includes one HFE ITAAC. The applicant's HFE ITAAC is similar to the format of the second standardized HFE ITAAC in the NRC's 2016 letter; however, it has been modified to address unique aspects of the NuScale application. The Table 3.15-1 does not include the first standardized HFE ITAAC for MCR Integrated System Validation (ISV) because the applicant completed ISV as part of the design certification, making this ITAAC not applicable (discussed below).

The HFE ITAAC verifies that the as-built Main Control Room Human System Interfaces (HSI) 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 integrated system validation as reconciled by the Design Implementation (DI) Implementation Plan (IP). The DI IP was reviewed by the staff in Chapter 18 of this SER. The DI IP describes human factors activities that ensure that changes to the NuScale design that occur after integrated system validation (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 this ITAAC to be an acceptable means of confirming that final as-built control room is consistent with the design validated during the ISV test, and that any deviations from validated design will be assessed, and if needed resolved, according to an acceptable process described in the DI IP.

Review procedures in the SRP, Section 14.3.9 (March 2007 revision), direct the staff to ensure the standard ITAAC entries in SRP Section 14.3, Appendix D are included for each plant system that has alarms, controls or displays. Appendix D of SRP Section 14.3 includes ITAAC entries for alarms, controls, or displays in the MCR and the Remote Shutdown Station (RSS). In addition, entries for such ITAAC are included in the standardized ITAAC contained in the NRC's 2016 letter. Therefore, the staff also reviewed the ITAAC in DCA Part 2, Tier 1, Revision 2, Table 2.5-7, "Module Protection System and Safety Display and Indication System ITAAC," for system-specific displays, controls, and alarms for the MCR and RSS. The staff compared the applicant's ITAAC to the standard 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.

NuScale requested an exemption from GDC 19 to depart from the portion of the rule requiring equipment outside the control room with a potential capability for subsequent cold shutdown of the reactor when the control room is evacuated. The staff evaluated this exemption in SER Section 1.14, "Index of Exemptions." NuScale established PDC 19 to require remote "safe shutdown" capability instead of "cold shutdown." DCA Part 2, Tier 2, Section 3.1.2.10, "Criterion 19 – Control Room," states that the displays, alarms, and controls in the RSS are not credited to meet the criteria of PDC 19 regarding equipment at appropriate locations outside the control room having the capability for safe shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe shutdown. The staff also reviewed DCA Part 2, Tier 2, Section 7.1.1.2.3, "Remote Shutdown Station," which states that operators can achieve safe shutdown of the reactors from outside the main control in the Module Protection System (MPS) equipment rooms. DCA Part 2, Tier 1, Table 2.5-7, "Module Protection System and Safety Display and Indication System ITAAC," ITAAC Numbers 1 and 16 verify this capability.

The staff finds it acceptable that the applicant has excluded ITAAC for displays alarms and controls in the RSS because there is no manual control of safety-related equipment allowed from the RSS, the RSS is not used to satisfy the remote shutdown capabilities of PDC 19, and the application includes existing ITAAC to verify the remote shutdown capability of the MPS. Additonally, as stated in the response to RAI 9401, Question 18-34 (ADAMS Accession No. ML18141A661), the displays, controls, and alarms provided in the RSS are identical to the Module Control System and Plant Control System displays in the MCR and include the parameters necessary to monitor safe shutdown of all units.

14.3.9.5 Combined License Information Items

There are no Combined License Information Items associated with this section.

14.3.9.6 Conclusion

The staff concludes that DCA Part 2, Tier 1 satisfactorily summarizes the top-level human factors engineering program design process objectives that are significant to safety and used to develop the HFE design and is consistent with DCA Part 2, Tier 2, Chapter 18, "Human Factors Engineering." Therefore, the design information associated with DCA Part 2, Tier 1, Section 3.15, "Human Factors Engineering," is acceptable.

Furthermore, the staff concludes that the ITAAC in Tier 1 adequately verify the DCA Part 2, Tier 1 HFE design. Therefore, within the review scope of this section, the staff concludes that the NuScale HFE ITAAC in Tier 1 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 certified NuScale design has been constructed and will be operated in conformity with the applicable portions of the DC, 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 emergency planning specific ITAAC for the design and specified COL item 14.3-1 for a future COL applicant to address ITAAC, as listed in SER Section 13.3.5. The acceptability of a future COL applicant's proposed ITAAC will be evaluated as part of the COL application process.

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

14.3.11.1 Introduction

This section reviews ITAAC and Tier 1 design descriptions applicable to containment and associated systems. The NuScale containment systems ITAAC are listed in the following DCA Part 2, Tier 1 tables:

- Table 2.1-4, "NuScale Power Module ITAAC," Numbers 7–9.
- Table 2.1-4, Number 23.

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, "Containment Systems - Inspections, Tests, Analyses, and Acceptance Criteria," 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 Sections 14.3 and 14.3.11, particularly the applicable review procedures identified in each SRP Section III, 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 combined license. The staff examined the phrasing and format of the ITAAC to determine if 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 DCA Part 2, Tier 1 ITAAC items were derived from the DCA Part 2, Tier 2 information. NuScale DCA Part 2, Tier 2, Table 14.3-1 provides background information associated with the Tier 1 design commitments and a brief description of how to complete the ITAAC listed above. 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 Tier 1 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 containment systems:

- 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 (instrumentation and controls).
- 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 containment system ITAAC items associated with the following DCA Part 2, Tier 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 containment system provides for the isolation of process systems that penetrate the containment vessel (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 as a result of postulated accidents.

NuScale DCA Part 2, Tier 1, Section 2.1, specifies ITAAC for containment isolation. DCA Part 2, Tier 1, Figure 2.1-1, "Containment System (Isolation Valves)," shows the functional arrangement of the containment isolation equipment. DCA Part 2, Tier 1, Section 2.1 includes design commitments requiring that containment isolation valve (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 DCA Part 2, Tier 1, Section 2.1 define the required closure times and piping lengths, and these times and lengths are consistent with DCA Part 2, Tier 2, Section 6.2.4, “Containment Isolation System.” Specifically, DCA Part 2, Tier 1, Table 2.1-4, 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.

DCA Part 2, Tier 1, 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 DCA Part 2, Tier 1, Section 2.1, Table 2.1-4, ITAAC Numbers 8 and 9, and finds the ITAAC to be consistent with staff guidance contained in SRP Section 14.3.11 and the Standardized DCA ITAAC (ADAMS Accession No. ML16096A132) because the valve closure times limit potential releases of radioactivity and the containment isolation valves outside containment are located as close to containment as practical. Because the ITAAC are consistent with staff guidance, the staff finds that the proposed ITAAC are acceptable and meet the requirements in 10 CFR 52.47(b)(1).

14.3.11.4.3   Containment Leakage Testing ITAAC

DCA Part 2, Tier 1, “Design Description,” describes the containment as 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 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” GDC 50, “Containment Design Basis.” This design description contains one of the principal performance characteristics of a leak tight containment. This design description is also consistent with Tier 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 technical specifications. The preoperational and periodic containment leakage testing capability for CNV openings (Type B) and CNV piping penetrations (Type C) are designed to meet the leakage acceptance criteria of 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.”

The applicant has requested an exemption from the integrated leak-rate test requirements in 10 CFR Part 50, Appendix J, for the CNV (Type A) test. The applicant has also requested an exemption from 10 CFR Part 50, Appendix A, GDC 52, “Capability for Containment Leakage Rate Testing.” The staff has determined that this exemption request meets the requirements for an exemption 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 containment isolation valve leakage rates.

The staff reviewed the proposed ITAAC Number 7 in DCA Part 2, Tier 1, Table 2.1-4, 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 meets 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.

Staff has also reviewed the ITAAC Number 23 in DCA Part 2, Tier 1, Table 2.1-4, 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 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

There are no Combined License Information Items associated with this section.

14.3.11.6 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 NuScale standard design nuclear power plant has been constructed and will be operated in accordance with the applicable portions of the design certification, 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 in Tier 1, consistent with SECY-19-0034.

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

14.3.12.1 Introduction

This section reviews ITAAC and Tier 1 design descriptions applicable to physical security systems. The following DCA Part 2, Tier 1 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

See Section 14.3.1.2 of this SER.

DCA Part 2, Tier 1: Section 3.16.1, “Design Description,” describes the NuScale standard design commitments for physical security systems (PSS) (including designation of vital areas) that provide capabilities for detection, assessment, and delay functions that protect against threats up to and including the design-basis threat (DBT) for radiological sabotage and provide defense in depth through the integration of systems, technologies, and equipment.

DCA Part 2, Tier 1, Section 3.0 and Table 3.0-1, “Shared Systems Subject to Inspections, Tests, Analyses, and Acceptance Criteria,” identify the systems that support multiple modules and are verified by ITAAC.

DCA Part 2, Tier 1, Section 3.8, “Plant Lighting System,” describes normal and emergency lighting systems for illuminations inside buildings. DCA Part 2, Tier 1, Table 3.8-1, “Plant Lighting System Inspections, Tests, Analyses, and Acceptance Criteria,” includes the design commitments and ITAAC for plant lighting systems that support physical security.

DCA Part 2, Tier 2: The DCA Part 2, Tier 2, Section 13.6.1, “Physical Security,” states the following:

> *The 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 for defense-in-depth through the integration of systems, technologies, and equipment. The design of physical security systems within the nuclear island and structures is described in Technical Report (TR) 0416-48929 (Reference 13.6-1), which is incorporated by reference to this FSAR.*

Technical Reports: TR-0416-48929, “NuScale Design of Physical Security Systems,” Revision 1, dated January 8, 2019 (ADAMS Accession No. ML19010A036), describes the security considerations in the NuScale standard design and the design bases, analyses, and assumptions for the design of PSS, including plant layout and building configurations, results of evaluations, and identification of vital equipment and areas for the NuScale standard design. The scope of the PSS described in the NuScale standard design is limited to those related to the nuclear islands and structures (i.e., reactor building (RXB) and control building (CRB)) that are within the scope of the NuScale standard design. TR-0416-48929 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 52.47, "Contents of applications; technical information," requires that information submitted for a design certification (DC) must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC.
- The NRC security regulations in 10 CFR Part 73 include 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.
- In accordance with the requirements of 10 CFR 52.79(a)(35) and 10 CFR 73.55(b), the COL applicant must describe a physical protection system and security organization whose objective will be 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. A physical protection system with capabilities to detect, assess, interdict, and neutralize shall be designed to protect against the DBT of radiological sabotage.
- The DBT for radiological sabotage is described in 10 CFR 73.1(a)(1), "Radiological sabotage." The COL applicant must describe how it will meet regulatory requirements and how it will achieve the high-assurance objective for the protection against the DBT of radiological sabotage. The provisions within 10 CFR 73.54, "Protection of digital computer and communication systems and networks"; 10 CFR 73.55, "Requirements for physical protection of licensed activities in nuclear power reactors against radiological sabotage"; 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," establish performance and prescriptive requirements that apply to the design of PSS, operational security requirements, management processes, and programs.
- The requirements in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart B, "Standard Design Certifications," regarding certification of design, limit the application of regulatory requirements that are specific to PSS within the scope of the NuScale standard design. According to 10 CFR 52.79, the operational or administrative controls, programs, and processes (e.g., management systems or controls) for security are addressed by the COL applicant and are not within the scope for certification of the NuScale standard design.
- 10 CFR 52.79(a)(28), which requires COL applicants to provide plans for preoperational testing and initial operations.

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’s technical review determined that the applicant adequately described appropriate ITAs and acceptance criteria verifying the required security functions, and the reliability, availability, or performance of selected PSS in accordance with 10 CFR 52.47(b)(1). The staff has concluded that the Tier 1 design descriptions adequately describe the top-level physical security features and performance characteristics that are prescribed by regulations. The staff also concludes that the Tier 1 design descriptions are based on and consistent with the Tier 2 material.

The PSS described in the NuScale 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 PSS are required to meet applicable performance and prescriptive requirements in 10 CFR Part 73. The design and technical bases for PSS within the scope of the NuScale DCA are described in DCA Part 2, Tier 2, Section 13.6, “Security,” which incorporates by reference TR-0416-48929. 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

DCA Part 2, Tier 1, Section 3.16.1, describes the design of PSS that detect, assess, and delay intrusion; enable onsite and offsite communications; and assist in the response to protect against the design-basis threat for radiological sabotage. The 13 ITAAC described in DCA, Part 2, Tier 1, Table 3.16-1 include those related to vital equipment locations, physical barriers, bullet-resistant structures, physical controls and security measures for vital areas, intrusion detection and assessment systems and subsystems and components, location of the central alarm station (CAS), access controls for vital areas, and communications that meet the requirements of 10 CFR Part 73. DCA Part 2, Tier 1, Table 3.16-1, includes 13 physical security ITAAC. These ITAAC verify the following design commitments for PSS in the scope of the standard NuScale plant, and are consistent with the standard ITAAC in SRP Section 14.3.12:

(1) Vital equipment will be located only within a vital area (Reference: SRP Section 14.3.12, ITAAC Number 1).

(2) Access to vital equipment will require passage through at least two physical barriers (Reference: SRP Section 14.3.12, ITAAC Number 2).

(3) The external walls, doors, ceilings, and floors in the main control room (MCR) and the CAS will be bullet resistant (Reference: SRP Section 14.3.12, ITAAC Number 3).

(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 (Reference: SRP Section 14.3.12, ITAAC Number 4).

(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 (Reference: SRP Section 14.3.12, ITAAC Number 5).

(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 (Reference: SRP Section 14.3.12, ITAAC Number 6).

(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 (Reference: SRP Section 14.3.12, ITAAC Number 7).

(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 (Reference: SRP Section 14.3.12, ITAAC Number 8).

(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 (Reference: SRP Section 14.3.12, ITAAC Number 9).

(10) Emergency exits through the vital area boundaries within the nuclear island and structures will be alarmed with intrusion detection devices and are secured by locking devices that allow prompt egress during an emergency (Reference: SRP Section 14.3.12, ITAAC Number 10).

(11) The CAS will have a landline telephone service with the control room and local law enforcement authorities (Reference: SRP Section 14.3.12, ITAAC Number 11).

(12) The CAS will be capable of continuous communication with on-duty security force personnel (Reference: SRP Section 14.3.12, ITAAC Number 12).

(13) Non-portable communications equipment in the CAS will remain operable from an independent power source in the event of the loss of normal power (Reference: SRP Section 14.3.12, ITAAC Number 13).

DCA Part 2, Tier 1, Table 3.16-1, ITAAC Numbers 1–13, identify 13 physical security ITAAC within the scope for the NuScale standard design that conform to SRP Section 14.3.12, Revision 1, as indicated above. The ITAAC descriptions for the PSS within the scope of the NuScale standard design described above are in the standard format (Design Commitment, Inspection, Test, Analysis, and Acceptance Criteria) for ITAAC.

In addition to engineered systems dedicated to providing security functions, the applicant described nonsafety-related plant systems that provide both safety and security functions. DCA Part 2, Tier 1, Section 3.8, describes the design of the PLS for the RXB and the CRB, which consists of normal and emergency lighting. DCA Part 2, Tier 1 Table 3.8-1 includes ITAAC to verify (1) battery-pack emergency lighting for illumination for post-fire safe shutdown (FSSD) activities outside of the MCR and the remote shutdown station (RSS) where post-FSSD

activities are performed; and (2) the normal and emergency (alternating current and direct current) lighting system illumination for operator workstations and auxiliary panels in the MCR. The PLS provides illumination within the interior of the RXB and the CRB to support security functions and should be verified by PLS ITAAC. The staff's review of this information is documented in SER Section 14.3.6. The design requirement and physical security ITAAC for illumination of the security isolation zones and exterior areas within the protected areas (standard physical security ITAAC Number 5 in SRP Section 14.3.12) are not within the scope of the NuScale standard design and, therefore, are to be addressed by the COL applicant.

DCA Part 2, Tier 2, Section 13.6, COL Item 13.6-4, is acceptable because it indicates that the COL applicant referencing the NuScale DC will address ITAAC related to the site-specific physical protection systems design.

NuScale TR-0416-48929, Table 5-1, identifies 23 commitments that pertain to the site-specific physical security systems and/or programs that a COL applicant referencing the NuScale certified design will address as COL items identified in DCA Part 2, Tier 2, Section 13.6. TR-0416-48929, Table 5-1, Item Numbers 4, 5, 6, 7, 8, 10, 11, 12, 13, 14, 17, 18, 19, 20, 21, 22, and 23, describe the COL applicant's responsibility for the design of security SSCs, parameters for engineered PSS, and configurations for establishing a site-specific physical protection system, for the following:

- location and design details for the secondary alarm station;
- physical security barriers outside the RXB and CRB;
- isolation zone, protected area, and associated intrusion assessment systems;
- vehicle barrier systems;
- exterior personnel, vehicle, and material access control portals;
- secondary alarm station and main security building;
- communication system secondary power supply;
- secondary security power system;
- bounding minimum safe standoff distance, alarm station survivability, and protection against vehicle bombs;
- alarm station functions and redundant capabilities;
- detection and assessment functions;
- illumination of isolation zones and protected areas;
- secondary alarm station communication; and

- uninterruptable power system and in-line generators or other source of backup power.

The staff finds the following:

- DCA Part 2, Tier 1, adequately identifies general design commitments and ITAAC that conform to those described in SRP Section 14.3.12, for vital areas and vital area access controls; bullet-resistant barriers; the central alarm station; the interior intrusion detection and assessment system; alarms, signal displays, and recording; transmission line supervision and monitoring; emergency exit controls; and security communications.

- The applicant has adequately identified other PSS, 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 NuScale standard design and will be addressed by the COL applicant. COL Item 13.6-4 establishes that the descriptions of site-specific PSS design and related ITAAC are to be addressed by the COL applicant that references the NuScale design certification.

- The staff concludes that the security ITAAC described above comply with 10 CFR 52.47(b).

14.3.12.4.2 Verification Program and Processes

DCA Part 2, Tier 1, Table 3.0-1, identifies PSS as a shared system supporting the NuScale Power Modules (NPMs), for 1-12 NPMs. DCA Part 2, Tier 1, Section 3.0, states "[f]or a multi-module plant, satisfactory 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 3.17 shall only be completed once in conjunction with the ITAAC in Chapter 2 for the first NPM." The applicant indicated that the physical security ITAAC identified in DCA Part 2, Tier 1, 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/or CRB. The physical security systems are common (shared) systems that support all 12 NPMs and are verified by ITAAC before fuel load for the first NPM. The staff finds this acceptable.

The staff finds the following:

- The applicant identified COL information items for establishing the test organization and management controls for the verifications of ITAAC, including those related to physical security. The applicant has established that a COL applicant referencing the NuScale standard design will address management controls needed to implement the verifications of physical security ITAAC, including procedure controls that document preparations, reviews, approvals, closeouts, and records.

- The system test process, as described in DCA Part 2, Tier 2, Sections 14.2 and 14.3, which the COL applicant must establish, if adequately implemented, will demonstrate

through testing that credited engineered SSCs will perform their intended security functions.

- The staff concludes that the applicant has established, in the NuScale DCA Part 2, the requirements for a COL applicant referencing the NuScale certified design to establish the management systems, processes, and organization that will verify the installation, construction, and performance of PSS through ITAAC.

14.3.12.4.3 Verification Methods for Physical Security ITAAC

DCA Part 2, Tier 1, Section 1.2.4, indicates that the verification (inspections, tests, or 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 ITAs may be performed as part of other activities, such as construction inspections or preoperational testing, and that the ITAs do not need to be performed as separate or discrete activities.

Performance methodologies for physical security ITAAC are discussed in more detail in DCA Part 2, Tier 2, Sections 14.2 and 14.3. The staff evaluates these methodologies below, in Subsections 14.3.12.4.3.1 through 14.3.12.4.3.3.

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

DCA Part 2, Tier 1, Table 3.16-1, establishes physical security ITAAC for the design commitments for vital equipment locations, vital areas, and access to vital equipment. DCA Part 2, Tier 2, Sections 14.2 and 14.3 describe the performance of these physical security ITAAC.

DCA Part 2, Tier 2, Table 14.3-2, Table 14.2-73, and Table 14.2-74, describe the verification of the physical security ITAAC in DCA Part 2, Tier 1, Table 3.16-1, associated with the vital equipment and vital areas, as follows:

ITAAC Number 1	An ITAAC inspection is performed of the as built vital equipment to verify that the equipment is located within a vital area.
ITAAC Number 2	An ITAAC inspection is performed of the as built vital equipment location to verify that access to vital equipment requires passage through at least two physical barriers.
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 central alarm station. This 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 preoperational 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.
ITAAC Number 5	A preoperational 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 central alarm station.
ITAAC Number 6	An inspection is performed of the as built central alarm station 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 preoperational 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 egress during an emergency.

The objective for ITAAC Number 1 is to demonstrate that vital equipment is located within the vital areas so that it is protected in accordance with regulatory requirements. The methods described in DCA Part 2, Tier 2, Table 14.3-2, include inspections to locate vital equipment and verify that access to each component meet the stated objective.

The list of vital equipment in TR-0416-48929 is information that is needed for verification of physical security ITAAC Number 2. The design commitment of this ITAAC states that "[a]ccess to vital equipment requires passage through at least two physical barriers." The discussion of this ITAAC in Tier 2, Table 14.3-2 includes this statement.

The verification of physical security ITAAC Number 3 includes the type test, analysis, or combination thereof that demonstrate that the structural construction of the MCR and the CAS are bullet resistant.

The DCA Part 2, Tier 2, Section 14.2, Table 14.2-73, describes the test abstract for physical security ITAAC Number 4 for verifying the access control system with a numbered photo-identification badge system, which will limit access to vital areas within the RXB and the CRB to authorized personnel. A preoperational 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.

DCA, Part 2, Tier 2, Section 14.2, Table 14.2-74, describes the test abstract for physical security ITAAC Number 5 for locked and alarmed access into vital areas. The objective is to determine that vital area personnel access barriers are locked and alarmed, unauthorized access is detected, and alarms at the central alarm station annunciate upon an intrusion into a vital area. 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. The test and inspection verifications apply to all vital areas, which are alarmed with intrusion detection systems, and demonstrate that activated intrusion detection systems annunciate in the CAS in the event of an unauthorized and attempted access of an unoccupied vital area.

DCA Part 2, Tier 2, Table 14.3-2, describes the inspection to verify physical security ITAAC Number 6 for the CAS. The objective is to verify that the location of the CAS meets the regulatory requirements for the CAS to be inside the protected area and for the interior of the alarm station to not be visible from the perimeter of the protected area. An inspection is the method used to determine that the CAS is designated a vital area and is not visible from outside the protected area.

In addition, the applicant described the verification of emergency exits for physical security ITAAC Number 10. The test abstract indicates that the objective is to verify that each of the 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 verification methods include inspections and tests of alarm initiation and indication and tests of locking devices. The tests operate the emergency egress locking mechanism in the vital area, verify that an alarm is generated when the door is opened, and that the alarmed information is displayed at the CAS.

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).

The staff finds that the applicant has 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 DCA Part 2, Tier 1, Section 13.6, “Design Description,” and Table 3.16-1.

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

DCA Part 2, Tier 2, Table 14.2-74, describes the physical security ITAAC for intrusion detection, assessment, and alarms as follows:

ITAAC Number 7	A preoperational test demonstrates that: (1) alarm annunciation indicates the type of alarm and location, (2) security alarm devices, including transmission lines to annunciators, are tamper-indicating and self-checking, and (3) an automatic indication is provided when failure of the alarm system or a component occurs or when the system is on standby power.
ITAAC Number 8	A preoperational test demonstrates that the intrusion detection and assessment system, within the nuclear island and structures, provides visual and audible annunciation of alarms in the central alarm station.

ITAAC Number 9 A preoperational test demonstrates that the intrusion detection and assessment system, within the nuclear island and structures, records each onsite security alarm annunciation, including each alarm, false alarm, alarm check, and tamper indication that identifies the type of alarm, location, alarm circuit, date, and time.

DCA Part 2, Tier 2, Table 14.2-74, describes the test abstract for ITAAC Number 7 for security alarms and tamper indications and system supervision of security alarm devices and transmission lines. The stated objectives are consistent with ITAAC Number 7, and the test methods verify the performance of security alarm annunciation, that alarm devices and transmission lines are tamper-indicating and self-checking, and that an automatic indication is provided of failure of the alarm system or a component when the system is on standby power. The acceptance criteria are in accordance with the requirements of 10 CFR 73.55(i)(3)(iv) through 73.55(i)(3)(v). The staff concludes that the test procedure adequately verifies ITAAC Number 7.

DCA Part 2, Tier 2, Table 14.2-74, describes the test abstract for ITAAC Numbers 8 and 9 for intrusion and assessment systems and alarm recording equipment. The objectives are to verify that the intrusion detection and assessment system provides visual and audible alarm annunciation of alarms in the CAS; records each alarm, false alarm, alarm check, and tamper indication; and identifies the type of alarm, location, alarm circuit, date, and time. The test methods include the testing of intrusion detection systems, security alarm annunciation and recording in the CAS.

The tests of the intrusion detection system include verifying system tamper indication, component failure for all devices and transmission lines, backup power, and intrusion alarms. The test abstract establishes the following acceptance criteria for the intrusion detection system: (1) alarm annunciation is received in the CAS, indicating type and location of the alarm; (2) audible and visual alarms are received in the CAS; and (3) the intrusion detection system records each alarm, false alarm, alarm check, and tamper indication, including location of the alarm, type of alarm, alarm circuit, date, and time. Physical Security ITAAC Number 8 and ITAAC Number 9 verify that security alarms have visual and audible features that indicate the types of alarms and their locations in accordance with the requirements of 10 CFR 73.55(i)(3)(i) through 10 CFR 73.55(i)(3)(iii) and that a record of the types of alarms, locations of alarms, alarm circuit, dates, time, and alarm status is maintained in accordance with 10 CFR 73.55(i)(4)(ii)(H). The acceptance criteria identified for the ITAAC related to the CAS are successful inspections and tests that verify alarm indication capabilities in accordance with 10 CFR 73.55(i)(2) for the CAS (one of two required alarm stations). This ITAAC does not cover the secondary alarm station because the COL applicant would be responsible for establishing the secondary alarm station and any associated ITAAC.

The staff finds that the applicant has provided adequate descriptions to support the identified ITAAC related to security alarm, system supervision, assessment, and intrusion detection system records in DCA Part 2, Tier 1, Section 3.16 and Table 3.16-1.

14.3.12.4.3.3 Inspections, Tests, and Analyses for Security Communications

DCA Part 2, Tier 2, Table 14.2-68, describes the following preoperational inspections and tests

that demonstrate the systems 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 central alarm station is equipped with conventional landline telephone service with the MCR and with local law enforcement authorities.
ITAAC Number 12	A preoperational test, inspection, or a combination of test and inspection demonstrates that the central alarm station is capable of continuous communication with on-duty security force personnel.
ITAAC Number 13	A preoperational test, inspection, or a combination of test and inspection demonstrates that nonportable communications equipment in the central alarm station remains operable (without disruption) from an independent power source in the event of loss of normal power.

DCA Part 2, Tier 2, Table 14.2-68, describes the test abstract that addresses physical security ITAAC Numbers 11, 12, and 13 for verifying the capabilities and performance of communication systems to support security requirements. The prerequisites include the complete installation of plant communication systems and components for the public address system, plant telephone system, and wireless communication system and the complete installation of operational communications equipment in the CAS and the MCR.

The test methods include tests of the communications systems to verify the availability of the public address system, plant telephone system, voice communications with offsite local law enforcement authorities, wireless communications system (radios), and the nonportable security communication system. The tests verify communications between the CAS and the MCR, test the portable radio system and backup plant system between the CAS and security personnel, and verify the continuity of communications capabilities on the secondary power supply in the event of loss of normal power. The test methods include verifying the capabilities of the communication systems to provide open and cleared communications that can be heard by plant personnel in areas where they are located. Testing includes the use of conventional (landline) telephone services to communicate between the CAS and the MCR and between the CAS and the local law enforcement authorities. The applicant indicated that dedicated security communication systems and plant communication systems are independent of each other. DCA Part 2, Tier 2, Table 14.2-68, addresses the verification of physical security ITAAC Numbers 11, 12, and 13 and the verification of the designs and installation of plant communication systems addressed in DCA Part 2, Tier 2, Section 9.5, “Other Auxiliary Systems.” The acceptance criteria identified for the physical security ITAAC are in accordance with 10 CFR 73.55(j)(3), 10 CFR 73.55(j)(4)(i) through (4)(ii), and 10 CFR 73.55(j)(5).

The staff finds that the applicant has provided adequate descriptions of the objectives, verification methods, and acceptance criteria that support the identified physical security ITAAC related to security communications in DCA Part 2, Tier 1, Section 3.16.

14.3.12.5 Combined License Information Items

Item No.	Description	DCA Part 2, Tier 2 Section
COL Item 13.6-4	A COL applicant that references the NuScale Power Plant design certification will provide inspections, tests, analyses, and acceptance criteria for site-specific physical security structures, systems, and components (SSC).	13.6

The staff finds this COL item acceptable for the reasons given above.

14.3.12.6 Conclusion

The staff finds the following:

- The applicant has proposed and adequately described attributes for physical security ITAAC verification.
- The applicant has identified an appropriate and reasonable set of test methods (inspections, tests, or analyses), and acceptance criteria for certification of the NuScale standard design in compliance with 10 CFR 52.47(b)(1).
- The applicant has 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) that support the DCA Part 2, Tier 1, descriptions of physical security ITAAC to meet the regulatory requirement of 10 CFR 52.47(b)(1).
- The applicant has identified appropriate descriptions for tests, inspections, and analyses that establish the framework for developing the detailed procedures for the conduct of the ITAAC.
- The applicant has provided adequate descriptions of requirements (i.e., COL Information Item 13.6-4) that indicate that a COL applicant referencing the NuScale standard design will describe the ITAAC for PSS that are outside the scope of the NuScale DC.
- The applicant has included sufficient, top-level design information in Tier 1, consistent with SECY-19-0034.

The staff concludes that the applicant has met 10 CFR 52.47, which requires information submitted for a DC to include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC. The applicant has met 10 CFR 52.47(b)(1), which requires the NuScale DCA to 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 are met, a facility

that incorporates the design certification has been constructed and will be operated in accordance with the design certification, the AEA, and NRC rules and regulations. The staff concludes that the applicant has provided sufficient information in the ITP 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

This section reviews ITAAC and Tier 1 design descriptions applicable to external flooding. The following DCA Part 2, Tier 1, tables contain the ITAAC applicable to this review area:

- Table 3.11-2, “Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria,” Number 3.
- Table 3.13-1, “Control Building Inspections, Tests, Analyses, and Acceptance Criteria,” Number 3.

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, “Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria;” Section 2.0, “Site Characteristics and Site Parameters;” and Section 2.4.2, “Floods,” provide acceptance criteria and additional guidance for this review area.

14.3.13.4 Technical Evaluation

Based on the staff’s review of the external flooding related information in DCA Part 2, Tier 1 and Tier 2, Chapter 2, the staff concludes that Tier 1 design descriptions and ITAAC adequately describe the top-level design features and performance characteristics that are significant to safety because these features and characteristics appropriately require that the RXB and the CRB are protected from external flooding, as discussed below. Also, the staff concludes that the Tier 1 design descriptions and ITAAC are based on and consistent with the Tier 2 material.

For the RXB, DCA Part 2, Tier 1, Section 3.11.1, “Design Description” states:

> *The RXB supports the following systems by housing and providing structural support:*
>
> *− NuScale Power Module*
>
> *− chemical and volume control system (CVCS)*
>
> *− ultimate heat sink*

- *module protection system*

- *nuclear monitoring system*

For the CRB, DCA Part 2, Tier 1, Section 3.13.1, “Design Description” states, “[T]he CRB supports the module protection system by housing and providing structural support.”

The Tier 1 design commitments for the RXB and CRB require that these Seismic Category I structures be protected from external flooding to prevent flooding ingress from affecting the SSCs important to safety. The ITAAC associated with these design commitments require an inspection 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 3 in Table 3.11-2 and ITAAC Number 3 in Table 3.13-1, along with their corresponding Tier 1 design commitments and discussions in DCA Part 2, Tier 2, Table 14.3-2, conform to the Standardized DCA ITAAC, design commitments, and associated Tier 2 discussion in the NRC’s April 8, 2016, letter. The staff finds that the Tier 1 design descriptions require, and the ITAAC are sufficient to demonstrate, that the RXB and CRB safety-related SSCs are adequately protected from external flooding. Based on the above, the staff finds that these ITAAC comply with 10 CFR 52.47(b)(1), and that the associated Tier 1 design descriptions and Tier 2 discussion are acceptable.

14.3.13.5 Combined License Information Items

There are no COL information items listed in DCA Part 2 Tier 2, Table 1.8-2, “Combined License Information Items,” for this area of review.

14.3.13.6 Conclusion

The NRC staff finds that the DCA Part 2, Tier 1, ITAAC for external flooding protection satisfy the requirements in 10 CFR 52.47(b)(1) and that the Tier 1 design descriptions conform to NRC guidance. The staff also finds that the description of how to complete these ITAAC in DCA Part 2, Tier 2, Table 14.3-2 is acceptable.