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NuScale US460 Plant
Standard Design Approval Application

License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC)

PART 8

Revision 2

Part 8 - License Conditions; Inspections, Tests, Analyses & Acceptance Criteria (ITAAC)

1.0 Introduction

This document presents the information developed for the NuScale Power Plant US460 standard design Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).

1.1 Definitions

The definitions below apply to terms that may be used in the NuScale Power Plant Inspections, Tests, Analyses, and Acceptance Criteria and ITAAC Design Descriptions.

Acceptance Criteria refers to the performance, physical condition, or analysis result for structures, systems, and components (SSC), or programs that demonstrate that the design commitment is met.

Analysis means a calculation, mathematical computation, or engineering or technical evaluation. Engineering or technical evaluations could include, but are not limited to, comparisons with operating experience or design of similar SSC.

Approved design means the design as described in the updated final safety analysis report (UFSAR), including any changes to the final safety analysis report (FSAR) since submission to the NRC of the last update of the FSAR.

As-built means the physical properties of SSC following the completion of their installation or construction activities at its final location at the plant site. In cases where it is technically justifiable, determination of physical properties of the as-built SSC may be based on measurements, inspections, or tests that occur before installation, provided that subsequent fabrication, handling, installation, and testing do not alter the properties.

ASME Code means Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as incorporated by reference in 10 CFR 50.55a with specific conditions or in accordance with relief granted or alternatives authorized by the NRC pursuant to 10 CFR 50.55a, unless a different section of the ASME Code is specifically referenced.

ASME Code Data Report means a document that certifies that a component or system is constructed in accordance with the requirements of the ASME Code. These data are recorded on a form approved by the ASME.

Component, as used for reference to ASME Code components, means a vessel, concrete containment, pump, pressure relief valve, line valve, storage tank, piping system, or core support structure that is designed, constructed, and stamped in accordance with the rules of the ASME Code. ASME Code Section III classifies a metal containment as a vessel.

Design Commitment means that portion of the ITAAC design description that is verified by ITAAC.

Inspect or **Inspection** means visual observations, physical examinations, or reviews of records based on visual observation or physical examination that compare (a) the SSC condition to one or more design commitments or (b) the program implementation elements to one or more program commitments, as applicable. Examples include walkdowns, configuration checks, measurements of dimensions, or nondestructive examinations. The terms, inspect and inspection, also apply to the review of Emergency Planning ITAAC requirements to determine whether ITAAC are met.

The ITAAC are those Inspections, Tests, Analyses, and Acceptance Criteria identified in the combined license (COL) that if met by the licensee are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act, as amended, and the Commission's rules and regulations.

Module-Specific or **Unit-Specific** ITAAC means ITAAC that are associated with SSC that are specific to and support operation of a single, individual NuScale Power Module (NPM). Module-specific ITAAC shall be satisfactorily completed for each NuScale Power Module.

NuScale Power Module is a collection of systems, sub-systems, and components that together constitute a modularized, movable, nuclear steam supply system. The NPM comprises a reactor core, a pressurizer (PZR), and two steam generators (SGs) integrated within a reactor pressure vessel (RPV) and housed in a compact steel containment vessel (CNV).

Reconciliation or **Reconciled** means the identification, assessment, and disposition of differences between a design feature as described in the UFSAR and an as-built plant design feature. For ASME Code piping systems, it is the reconciliation of differences between the design as described in the UFSAR and the as-built piping system. For structural features, it is the reconciliation of differences between the design as described in the UFSAR and the as-built structural feature.

Report, as used in the ITAAC table Acceptance Criteria column, means a document that verifies that the acceptance criteria of the subject ITAAC have been met and references the supporting documentation. The report may be a simple form that consolidates necessary information related to the closure package for supporting successful completion of the ITAAC.

Safe Shutdown Earthquake (SSE) Ground Motion is the site-specific vibratory ground motion for which safety-related SSC are designed to remain functional. The SSE for a site is a smoothed spectra developed to envelop the ground motion response spectra. The SSE is characterized at the free ground surface.

Shared or Common ITAAC means ITAAC that are associated with shared SSC and activities that support multiple NPMs. This includes (1) SSC that are shared by multiple NPMs, and for which the interface and functional performance requirements between the shared SSC and each NPM are identical, or (2) analyses or other generic design and qualification activities that are identical for each NPM (e.g., environmental qualification of equipment). For a multi-module plant, satisfactory completion of a shared ITAAC for the lead NPM shall constitute satisfactory completion of the shared ITAAC for associated NPMs.

Test means actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of as-built SSC, unless explicitly stated otherwise, to determine whether ITAAC are met.

Type Test means a test on one or more sample components of the same type and manufacturer to qualify other components of the same type and manufacturer. A type test is not necessarily a test of an as-built SSC.

1.2 General Provisions

1.2.1 Systems Within the Scope of Inspections, Tests, Analyses, and Acceptance Criteria

The results of the ITAAC screenings of SSC that are either fully or partially within the scope of the NuScale Power Plant standard design are provided in each system's ITAAC system description tables. These tables identify those SSC that are addressed by ITAAC.

ITAAC does not include systems that have been determined to not contain top-level design features or performance characteristics that require verification through ITAAC.

1.2.2 Inspections, Tests, Analyses, and Acceptance Criteria Design Descriptions

The ITAAC design descriptions pertain only to the SSC of the standard design and not to their operation and maintenance after fuel load. In the event of an inconsistency between the ITAAC design descriptions and the FSAR information, the FSAR shall govern.

The ITAAC design descriptions consist of ITAAC system descriptions, ITAAC system description tables, ITAAC system description figures, and design commitments. The ITAAC system description tables and ITAAC system description figures are only used when appropriate. The ITAAC system description provides a concise description of the scope and top-level design features and performance characteristics of the SSC system functions. The ITAAC system description only describes those portions of the system or structure that are important to the top-level design features and performance characteristics of the system or structure. Design commitments are provided in numbered paragraphs that are used to develop the Design Commitment column in the ITAAC table. These commitments address top-level design features and performance characteristics such as:

- seismic classification
- ASME Code classification
- Class 1E SSC
- equipment to be qualified for harsh environments
- instrumentation and controls (I&C) equipment to be qualified for other than harsh environments

When the term "operate," "operates," or "operation" is used with respect to equipment discussed in the acceptance criteria, it refers to the actuation or control of the equipment.

1.2.3 Interpretation of Inspections, Tests, Analyses, and Acceptance Criteria System Description Tables

Cells with no values in ITAAC system description tables contain an "N/A" to denote that the cell is "not applicable."

1.2.4 Interpretation of Inspections, Tests, Analyses, and Acceptance Criteria System Description Figures

Figures are provided for some systems or structures with the amount of information depicted based on their safety significance. These figures may represent a functional diagram, general structural representation, or other general illustration. Unless specified, these figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built SSC. In particular, the as-built attributes of SSC may vary from the attributes depicted on these figures, provided that the top-level design features discussed in the ITAAC design description pertaining to the figure are not adversely affected. Valve position indications shown on system description figures do not represent a specific operational state.

The figure legends in FSAR Section 1.7 are used to interpret ITAAC system description figures.

1.2.5 Implementation of Inspections, Tests, Analyses, and Acceptance Criteria

Design commitments, inspections, tests, analyses, and acceptance criteria are provided in ITAAC tables with the following format:

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each commitment in the "Design Commitment" column of the ITAAC tables has one or more associated requirements for inspections, tests, or analyses specified in the "Inspections, Tests, Analyses" column. Each inspection, test, or analysis has one or more associated acceptance criteria in the fourth column of the ITAAC tables that demonstrate the Design Commitment in the second column is met.

Inspections, tests, or analyses may be performed by the licensee or by its authorized vendors, contractors, or consultants.

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 issuance of the COL for those ITAAC that do not require as-built equipment.
- performed at a location other than the construction site.

Additionally, inspections, tests, or analyses may be performed as part of other activities such as construction inspections and preoperational testing. Therefore, inspections, tests, or analyses need not be performed as a separate or discrete activity.

If an acceptance criterion does not specify the temperature, pressure, or other conditions under which an inspection or test must be performed, then the inspection or test conditions are not constrained.

Many of the Acceptance Criteria state that a report or analysis “exists and concludes that....” When these words are used, it indicates that the ITAAC for that Design Commitment will be met when it is confirmed that appropriate documentation exists and the documentation shows that the Design Commitment is met.

For the acceptance criteria, appropriate documentation may be a single document or a collection of documents that show that the stated acceptance criteria are met. The following are examples of appropriate documentation:

- design reports
- test reports
- inspection reports
- analysis reports
- evaluation reports
- design and manufacturing procedures
- certified data sheets
- commercial grade dedication procedures and records
- quality assurance records
- calculation notes
- equipment qualification data packages

Conversion or extrapolation of test results from the test conditions to the design conditions may be necessary to satisfy an ITAAC. Suitable justification should be provided for any conversions or extrapolations of test results necessary to satisfy an ITAAC.

1.2.6 Acronyms and Abbreviations

The acronyms and abbreviations contained in FSAR Table 1.1-1 are applicable to ITAAC.

2.0 Module-Specific Structures, Systems, and Components Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design Descriptions and ITAAC

This chapter provides ITAAC Design Descriptions and ITAAC for those SSC that are specific to and support operation of a single NPM. Module-specific ITAAC shall be satisfactorily completed for each NPM.

2.1 NuScale Power Module

2.1.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the NPM and its associated systems. The systems contained within the boundary of the NPM are the

- reactor coolant system (RCS), including the RPV, PZR, SG, reactor vessel internals (RVI), and associated piping and valves. The RCS piping is located inside the CNV and connects to containment piping located outside the CNV via CNV nozzles. Steam generator system (SGS) piping is located inside the CNV. The SGS steam piping connects to containment system (CNTS) steam piping located outside the CNV via CNV nozzles. The SGS feedwater piping connects to the decay heat removal system (DHRS) condenser condensate line inside the CNV.
- control rod drive system (CRDS), including the control rod drive mechanisms (CRDMs).
- CNTS, including the CNV and CIVs and associated piping. Containment piping is located outside the CNV with the exception of CNTS piping used for containment flooding and drain.
- emergency core cooling system (ECCS), including associated valves, supplemental boron dissolvers, and CNV lower mixing tubes.
- DHRS, including associated piping and valves. The DHRS steam piping is located outside the CNV and connects to containment piping outside the CNV. The DHRS condensate lines connect the DHR condensers to the SGS feedwater piping inside the CNV.

The NPM includes the pressure retaining structures of these systems because they are part of either the reactor coolant pressure boundary (RCPB) or the CNV pressure boundary. Therefore, the mechanical design and arrangement of the piping, CRDS, and NPM valves (emergency core cooling, reactor safety, and containment isolation) are included in this section.

The NPM performs the following safety-related functions that are verified by ITAAC:

- The RCS supports the CNTS by supplying the RCPB and a fission product boundary via the RPV and other appurtenances.
- The CRDS supports the RCS by maintaining the pressure boundary of the RPV.
- The SGS supports the RCS by supplying part of the RCPB.
- The ECCS supports the RCS by providing a portion of the RCPB for maintaining the RCPB integrity.
- The CNTS supports the Reactor Building (RXB) by providing a barrier to contain mass, energy, and fission product release from a degradation of the RCPB.

- The ECCS supports the CNTS by providing a portion of the containment boundary for maintaining containment integrity.
- The CNTS supports the DHRS by providing the required pressure boundary for DHRS operation.
- The RCS supports the SGS by providing physical support for the SG tube supports and for the integral steam and feed plenums.
- The RCS supports the reactor core by the RVI providing mechanical support to orient, position, and seat the fuel assemblies.
- The RCS supports the CRDS by the RPV and the RVI supporting and aligning the control rods.
- The CNTS supports the DHRS by providing structural support for the DHRS piping.
- The CNTS supports the neutron monitoring system (NMS) by providing structural support for the ex-core detectors.
- The RCS supports the ECCS by providing mechanical support for the ECCS valves.
- The RCS supports the in-core instrumentation system by providing structural support of the in-core instrumentation guide tubes.
- The CNTS supports the CRDS by providing structural support for the CRDMs.
- The CNTS supports the RCS by providing structural support for the RPV.
- The CNTS supports the ECCS by providing structural support of the trip and reset valves for the ECCS reactor vent valves (RVVs) and reactor recirculation valves (RRVs).
- The CNTS supports the RCS by closing the CIVs for PZR spray, chemical and volume control system (CVCS) makeup, CVCS letdown, and RPV high point degasification when actuated by the module protection system (MPS) for RCS isolation.
- The CNTS supports the RXB by providing a barrier to contain mass, energy, and fission product release by closure of the CIVs upon containment isolation signal.
- The CNTS supports the DHRS by closing CIVs for main steam and feedwater and opening DHRS actuation valves when actuated by MPS for DHRS operation.
- The ECCS supports the RCS by opening the ECCS reactor vent valves and RRVs when their respective trip valve is actuated by MPS.
- The DHRS supports the RCS by opening the DHRS actuation valves on a DHRS actuation signal.
- The CNTS supports the MPS by providing MPS actuation instrument information signals through the CNV.
- The ECCS supports the RCS by providing boric acid to recirculated coolant from containment to ensure the core remains subcritical for design basis events.
- The CNTS maintains an inert containment atmosphere following design-basis events.

The NPM performs the following nonsafety-related, risk-significant function that is verified by ITAAC. The CNTS supports the Reactor Building crane (RBC) by providing lifting attachment points that the RBC can connect to so that the NPM can be lifted.

The NPM performs the following nonsafety-related functions that are verified by ITAAC:

- The CNTS supports the SGS by providing structural support for the SGS piping.
- The CNTS supports the CRDS by providing structural support for the CRDS piping.
- The CNTS supports the RCS by providing structural support for the RCS piping.
- The CNTS supports the feedwater system by providing structural support for the feedwater system piping.

Design Commitments

- The NuScale Power Module ASME Code Class 1, 2, and 3 piping systems listed in Table 2.1-3 and NuScale Power Module ASME Code Class 1, 2, 3, and CS components listed in Table 2.1-4 comply with ASME Code Section III requirements.
- The NuScale Power Module ASME Code Class 1, 2, and 3 components listed in Table 2.1-4 conform to the rules of construction of ASME Code Section III.
- The NuScale Power Module ASME Code Class CS components listed in Table 2.1-4 conform to the rules of construction of ASME Code Section III.
- Safety-related SSC are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.
- The ECCS supplemental boron components are installed such that ECCS can perform the safety-related emergency supplemental boron function.
- Each CNTS containment electrical penetration assembly (EPA) listed in Table 2.1-5 is rated either (i) to withstand fault and overload currents for the time required to clear the fault from its power source, or (ii) to withstand the maximum fault and overload current for its circuits without a circuit interrupting device.
- The CNV serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.
- Closure times for CIVs listed in Table 2.1-5 limit potential releases of radioactivity.
- The length of piping listed in Table 2.1-3 shall be minimized between the containment penetration and the associated outboard CIVs.
- The CNTS containment electrical penetration assemblies listed in Table 2.1-5 are sized to power their design loads.
- The ECCS valves, CIVs, and DHRS actuation valves listed in Table 2.1-4, and their associated hydraulic lines, are installed such that each valve can perform its safety function.

- The remotely operated CNTS containment isolation valves listed in Table 2.1-5 change position under design-basis temperature, differential pressure, and flow conditions.
- The ECCS reactor recirculation valves and RVVs listed in Table 2.1-4 change position under design-basis temperature, differential pressure, and flow conditions.
- The DHRS valves listed in Table 2.1-4 change position under design-basis temperature, differential pressure, and flow conditions.
- The CNV top support structure (TSS) supports its rated load.
- The CNV top support structure is constructed to provide assurance that a single failure does not result in the uncontrolled movement of the lifted load.
- The CNTS hydraulic-operated valves listed in Table 2.1-4 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.
- The ECCS reactor recirculation valves and RVVs listed in Table 2.1-4 fail to (or maintain) their safety-related position on loss of electrical power to their corresponding trip valves under design-basis temperature, differential pressure, and flow conditions.
- The DHRS hydraulic-operated valves listed in Table 2.1-4 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.
- The CNTS check valves listed in Table 2.1-4 change position under design-basis temperature, differential pressure, and flow conditions.
- The CNTS passive autocatalytic recombiner (PAR) is installed such that it can perform its safety-related function to maintain an inert containment atmosphere.
- The CNV has sufficient net free volume to maintain peak containment pressure below containment design pressure during design-basis events.

2.1.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.1-1 contains the ITAAC for the NPM.

Table 2.1-1: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.01.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The NuScale Power Module ASME Code Class 1, 2, and 3 piping systems listed in Table 2.1-3 and NuScale Power Module ASME Code Class 1, 2, 3, and CS components listed in Table 2.1-4 comply with ASME Code Section III requirements.	i. An inspection will be performed of the NuScale Power Module ASME Code Class 1, 2, and 3 as-built piping system Design Reports for systems listed in Table 2.1-3 required by ASME Code Section III.	i. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the NuScale Power Module ASME Code Class 1, 2, and 3 as-built piping systems listed in Table 2.1-3 meet the requirements of ASME Code Section III.
		ii. An inspection will be performed of the NuScale Power Module ASME Code Class 1, 2, and 3 as-built component Design Reports for components listed in Table 2.1-4 required by ASME Code Section III.	ii. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the NuScale Power Module ASME Code Class 1, 2, and 3 as-built components listed in Table 2.1-4 meet the requirements of ASME Code Section III.
		iii. An inspection will be performed of the NuScale Power Module ASME Code Class CS as-built component Design Reports for components listed in Table 2.1-4 required by ASME Code Section III.	iii. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the NuScale Power Module ASME Code Class CS as-built components listed in Table 2.1-4 meet the requirements of ASME Code Section III.
02.	The NuScale Power Module ASME Code Class 1, 2, and 3 components listed in Table 2.1-4 conform to the rules of construction of ASME Code Section III.	An inspection will be performed of the NuScale Power Module ASME Code Class 1, 2, and 3 as-built component Data Reports for components listed in Table 2.1-4 required by ASME Code Section III.	ASME Code Section III Data Reports for the NuScale Power Module ASME Code Class 1, 2, and 3 components listed in Table 2.1-4 and interconnecting piping exist and conclude that the requirements of ASME Code Section III are met.
03.	The NuScale Power Module ASME Code Class CS components listed in Table 2.1-4 conform to the rules of construction of ASME Code Section III.	An inspection will be performed of the NuScale Power Module ASME Code Class CS as-built component Data Reports for components listed in Table 2.1-4 required by ASME Code Section III.	ASME Code Section III Data Reports for the NuScale Power Module ASME Code Class CS components listed in Table 2.1-4 exist and conclude that the requirements of ASME Code Section III are met.
04.	Safety-related SSC are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.	An inspection and analysis will be performed of the as-built high- and moderate-energy piping systems and protective features for the safety-related SSC.	Protective features are installed in accordance with the as-built Pipe Break Hazard Analysis Report, and safety-related SSC are protected against or qualified to withstand the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.

Table 2.1-1: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.01.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
05.	The ECCS supplemental boron components are installed such that the ECCS can perform the safety-related emergency supplemental boron function.	An inspection will be performed of the ECCS supplemental boron components.	A report exists and concludes that the ECCS • supplemental boron dissolvers, • CNV lower mixing tubes, • dissolver condensate channels, and • lower mixing tube condensate channels are installed in accordance with the associated installation specification.
06.	Each CNTS containment EPA listed in Table 2.1-5 is rated either (i) to withstand fault and overload currents for the time required to clear the fault from its power source, or (ii) to withstand the maximum fault and overload current for its circuits without a circuit interrupting device.	An analysis will be performed of each CNTS as-built containment EPA listed in Table 2.1-5.	For each CNTS containment EPA listed in Table 2.1-5, either (i) a circuit interrupting device coordination analysis exists and concludes that the current-carrying capability for the CNTS containment EPA is greater than the analyzed fault and overload currents for the time required to clear the fault from its power source, or (ii) an analysis of the CNTS containment electrical penetration maximum fault and overload current exists and concludes the fault and overload current is less than the current-carrying capability of the CNTS containment electrical penetration.
07.	The CNV serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.	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.
08.	Closure times for CIVs listed in Table 2.1-5 limit potential releases of radioactivity.	A test will be performed of the automatic CIVs listed in Table 2.1-5.	Each CIV listed in Table 2.1-5 travels from the full open to full closed position in less than or equal to the time listed in FSAR Section 6.2.4 after receipt of a containment isolation signal.
09.	The length of piping listed in Table 2.1-3 shall be minimized between the containment penetration and the associated outboard CIVs.	An inspection will be performed of the as-built piping listed in Table 2.1-3 between containment penetrations and associated outboard CIVs.	The length of piping between each containment penetration and its associated outboard CIV is less than or equal to the length identified in Table 2.1-3.

Table 2.1-1: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.01.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10.	The CNTS containment EPA listed in Table 2.1-5 are sized to power their design loads.	i. An analysis will be performed of the CNTS as-designed containment EPA listed in Table 2.1-5. ii. An inspection will be performed of CNTS as-built containment EPA listed in Table 2.1-5.	i. An electrical rating report exists that defines and identifies the required design electrical rating to power the design loads of each CNTS containment EPA listed in Table 2.1-5. ii. The electrical rating of each CNTS containment EPA listed in Table 2.1-5 is greater than or equal to the required design electrical rating as specified in the electrical rating report.
11.	The ECCS valves, CIVs, and DHRS actuation valves listed in Table 2.1-4, and their associated hydraulic lines, are installed such that each valve can perform its safety function.	An inspection will be performed of each ECCS valve, CIV, and DHRS actuation valve listed in Table 2.1-4, and associated hydraulic line.	A report exists and concludes each ECCS valve, CIV, and DHRS actuation valve listed in Table 2.1-4, and the associated hydraulic line, is installed in accordance with its associated installation specification.
12.	The CNV serves as an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment.	A preservice design pressure leakage test of the CNV will be performed.	No water leakage is observed at CNV bolted flange connections.
13.	The remotely-operated CNTS containment isolation valves listed in Table 2.1-5 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the remotely-operated CNTS containment isolation valves listed in Table 2.1-5 under preoperational temperature, differential pressure, and flow conditions.	Each remotely-operated CNTS containment isolation valve listed in Table 2.1-5 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
14.	The ECCS RRVs and RVVs listed in Table 2.1-4 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the ECCS RRVs and RVVs listed in Table 2.1-4 under preoperational temperature, differential pressure, and flow conditions.	Each ECCS RRV and RVV listed in Table 2.1-4 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
15.	The DHRS valves listed in Table 2.1-4 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the DHRS valves listed in Table 2.1-4 under preoperational temperature, differential pressure, and flow conditions.	Each DHRS valve listed in Table 2.1-4 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
16.	The CNV top support structure supports its rated load.	A rated load test will be performed of the CNV top support structure.	The CNV top support structure supports a load of at least 150 percent of the manufacturer's rated capacity.
17.	The CNV top support structure is constructed to provide assurance that a single failure does not result in the uncontrolled movement of the lifted load.	An inspection will be performed of the as-built CNV top support structure.	The CNV top support structure is single-failure-proof.

Table 2.1-1: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.01.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
18.	The CNTS hydraulic-operated valves listed in Table 2.1-4 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the CNTS hydraulic-operated valves listed in Table 2.1-4 under preoperational temperature, differential pressure, and flow conditions.	Each CNTS hydraulic-operated valve listed in Table 2.1-4 fails to (or maintains) its safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
19.	The ECCS RRVs and RVVs listed in Table 2.1-4 fail to (or maintain) their safety-related position on loss of electrical power to their corresponding trip valves under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the ECCS RRVs and RVVs listed in Table 2.1-4 under preoperational temperature, differential pressure, and flow conditions.	Each ECCS RRV and RVV listed in Table 2.1-4 fails to (or maintains) its safety-related position on loss of electrical power to its corresponding trip valve under preoperational temperature, differential pressure, and flow conditions.
20.	The DHRS hydraulic-operated valves listed in Table 2.1-4 fail to (or maintain) their safety-related position on loss of electrical power under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the DHRS hydraulic-operated valves listed in Table 2.1-4 under preoperational temperature, differential pressure, and flow conditions.	Each DHRS hydraulic-operated valve listed in Table 2.1-4 fails to (or maintains) its safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
21.	The CNTS check valves listed in Table 2.1-4 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the CNTS check valves listed in Table 2.1-4 under preoperational temperature, differential pressure, and flow conditions.	Each CNTS check valve listed in Table 2.1-4 strokes fully open and closed (under forward and reverse flow conditions, respectively) under preoperational temperature, differential pressure, and flow conditions.
22.	The CNTS passive autocatalytic recombiner is installed such that it can perform its safety-related function to maintain an inert containment atmosphere.	An inspection will be performed of the PAR.	A report exists and concludes that the PAR is installed in accordance with the associated installation specification.
23.	The as-built CNV has sufficient net free volume to maintain peak containment pressure below containment design pressure during design-basis events.	A reconciliation analysis will be performed of the as-built containment net free volume.	A report exists and concludes the as-built containment net free volume is greater than or equal to the free volume listed in FSAR Table 6.2-2.

Table 2.1-2: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
02.01.01	As required by ASME Code Section III NCA-1210, each ASME Code Class 1, 2, and 3 component (including piping systems) of a nuclear power plant requires a Design Report in accordance with NCA-3550. NCA-3551.1 requires that the drawings used for construction be in agreement with the Design Report before it is certified and be identified and described in the Design Report. It is the responsibility of the N Certificate Holder to furnish a Design Report for each component and support, except as provided in NCA-3551.2 and NCA-3551.3. NCA-3551.1 also requires that the Design Report be certified by a registered professional engineer when it is for Class 1 components and supports, Class CS core support structures, Class MC vessels and supports, Class 2 vessels designed to NC-3200 (NC-3131.1), or Class 2 or Class 3 components designed to Service Loadings greater than Design Loadings. A Class 2 Design Report shall be prepared for Class 1 piping NPS 1 or smaller that is designed in accordance with the rules of Subsection NC. NCA-3554 requires that any modification of any document used for construction, from the corresponding document used for design analysis, shall be reconciled with the Design Report. An ITAAC inspection is performed of the NuScale Power Module ASME Code Class 1, 2, and 3 as-built piping system Design Reports and the ASME Code Class 1, 2, 3, and CS as-built component Design Reports to verify that the requirements of ASME Code Section III are met. The inventory of the NuScale Power Module ASME Code Class, 1, 2, and 3 piping systems verified by ITAAC 02.01.01 is contained in Table 2.1-3. The inventory of the NuScale Power Module ASME Code Class, 1, 2, 3, and CS components verified by ITAAC 02.01.01 is contained in Table 2.1-4. FSAR Figure 6.6-1, ASME Class Boundaries for NuScale Power Module Piping Systems, provides a graphical representation of the NPM piping systems inventoried in Table 2.1-3.
02.01.02	The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code Class 1, 2 and 3 components conform to the requirements of the Code. As defined in NCA-9000, a component can be a vessel, pump, pressure relief valve, line valve, storage tank, piping system, or core support structure that is designed, constructed, and stamped in accordance with the rules of Section III. The NuScale Power Module ASME Code Class 1, 2, and 3 components require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the inspector as specified by NCA-8410. The type of individual Data Report forms necessary to record the required code data are specified in Table NCA-8100-1. An ITAAC inspection is performed of the Data Reports for NuScale Power Module ASME Code Class 1, 2, and 3 as-built components listed in Table 2.1-4 and interconnecting piping to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1, (2) ensure that the certificate holder or owner and the authorized nuclear inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code are met.
02.01.03	The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code CS components conform to the requirements of the Code. The ASME Code Class CS components require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the Inspector as specified by NCA-8410. The type of individual Data Report Forms necessary to record the required Code Data is identified in Table NCA-8100-1. An ITAAC inspection is performed of the Data Reports for the ASME Code Class CS as-built components listed in Table 2.1-4 to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1, (2) ensure that the certificate holder or owner and the inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code Section III are met.

Table 2.1-2: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.01.04	FSAR Section 3.6, Protection against Dynamic Effects Associated with Postulated Rupture of Piping, provides the design bases and criteria for the analysis required to demonstrate that safety-related SSC are not impacted by the adverse effects of a high-and moderate-energy pipe failure within the plant. An ITAAC inspection is performed to verify that the as-built protective features credited in the reconciled Pipe Break Hazards Analysis Report such as pipe whip restraints, pipe whip or jet impingement barriers, jet impingement shields, or guard pipe have been installed in accordance with design drawings of sufficient detail to show the existence and location of the protective hardware. The as-built inspection is intended to verify that changes to postulated pipe failure locations and protective features or protected equipment made during construction do not adversely affect the safety-related functions of the protected equipment.
02.01.05	Quality Control inspection hold points are used to ensure the as-built ECCS supplemental boron components are installed consistent with their associated installation specifications, and therefore capable of performing their safety functions. To demonstrate the acceptance criterion for ITAAC 02.01.05 is satisfied and the associated design commitment fully met, a report will exist and conclude Quality Control inspection hold points exist and have been completed in accordance with the Quality Assurance Program for the ECCS • supplemental boron dissolvers, • CNV lower mixing tubes, • dissolver condensate channels, and • lower mixing tube condensate channels.
02.01.06	The CNTS electrical penetrations listed in Table 2.1-5 may be one of two types, one with or without a circuit interrupting device. An ITAAC confirms that each type of penetration is evaluated to confirm it can withstand its maximum fault current. A circuit interrupting device coordination analysis confirms and concludes in a report that the as-built containment electrical penetration assembly listed in Table 2.1-5 that has a circuit interrupting device can withstand fault currents for the time required to clear the fault from its power source. FSAR Section 8.3.1 discusses electrical penetration assemblies that are not equipped with protection devices whose maximum fault current in these circuits would not damage the electrical penetration assembly if that fault current is available indefinitely. An analysis of a CNTS as-built containment penetration without a circuit interrupting device confirms and concludes in a report that the maximum fault current is less than the current carrying capability of the CNTS containment electrical penetration.
02.01.07	FSAR Section 6.2.6, Containment Leakage Testing, provides a discussion of the leakage testing requirements of the CNV, which serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. As discussed in FSAR Section 6.2.6, the NuScale CNV is exempted from the integrated leak rate testing specified in the General Design Criterion (GDC) 52. In accordance with FSAR Table 14.2-38, a preoperational test demonstrates that the leakage rate for local leak rate tests (Type B and Type C) for pressure containing or leakage-limiting boundaries and CIVs meet the leakage acceptance criterion of 10 CFR Part 50 Appendix J.
02.01.08	FSAR Section 6.2.4, Containment Isolation System, provides a discussion of how the CNTS containment isolation valves close within the required closure time after receipt of a containment isolation signal to meet containment isolation requirements following a radiological release in the CNV. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that each automatic CIV listed in Table 2.1-5 travels from the full open to full closed position in less than or equal to the time listed in FSAR Section 6.2.4 after receipt of a containment isolation signal.

Table 2.1-2: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.01.09	FSAR Section 6.2.4, provides a discussion of the isolation valves outside containment that are located as close to the containment as practical. FSAR Section 6.2.4 specifies that certain CIVs are directly welded to containment isolation test fixtures, which are directly welded to CNV nozzle safe-ends. For main steam lines, the nozzle includes an extended nozzle safe-end that is welded directly to the main steam isolation valve. An ITAAC inspection is performed to verify the length of piping between each containment penetration and its associated outboard CIVs is less than or equal to the length identified in Table 2.1-3.
02.01.10	FSAR Section 8.3.1, Alternating Current Power Systems, discusses the electrical design requirements for EPAs with respect to RG 1.63. An analysis determines the required design electrical rating needed to power the design loads of each NuScale Power Module CNTS containment EPA listed in Table 2.1-5. An ITAAC inspection is performed to verify that the electrical rating of each NuScale Power Module CNTS containment EPA listed in Table 2.1-5 is greater than or equal to the required design electrical rating. This ITAAC inspection may be performed any time after manufacture of the CNTS containment EPAs.
02.01.11	Quality Control inspection hold points are used to ensure the as-built ECCS valves, CIVs, DHRS actuation valves, and their associated hydraulic lines are installed consistent with their associated installation specifications, and therefore capable of performing their safety functions. To demonstrate the acceptance criterion for ITAAC 02.01.11 is satisfied and the associated design commitment fully met, a report will exist and conclude the following: 1) Quality Control inspection hold points exist and have been completed in accordance with the Quality Assurance Program for each of the following attributes of the ECCS valves, CIVs, and DHRS actuation valves: a. geometric configuration b. orientation c. accessibility 2) Quality Control inspection hold points exist and have been completed in accordance with the Quality Assurance Program for each of the following attributes of routing of the hydraulic lines of the ECCS valves, CIVs, and DHRS actuation valves: a. twisting b. bend radii c. crimping d. support e. line separation f. safe shipment feature removal
02.01.12	FSAR Section 6.2.6, Containment Leakage Testing, provides the test requirements for a preservice design pressure leakage test of the CNV. The test verifies no observed leakage at the CNV bolted flange connections under design pressure. The test may be performed any time after manufacture of the CNV, before the NPM being placed into service.

Table 2.1-2: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.01.13	The CNTS remotely-operated CNTS containment isolation valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that the CNTS remotely operated CNTS containment isolation valves listed in Table 2.1-5 stroke fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
02.01.14	The ECCS safety-related RRVs and RVVs are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. For the first NPM only, a test is conducted under preoperational test conditions that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations. The test is initiated with an initial RPV to CNV differential pressure greater than the inadvertent actuation block threshold pressure of 900 psid in accordance with FSAR Table 14.2-40 and demonstrates that the ECCS safety-related RRVs and RVVs listed in Table 2.1-4 stroke fully open by remote operation. In accordance with FSAR Table 14.2-56, a test is performed that demonstrates that the ECCS safety-related RRVs and RVVs listed in Table 2.1-4 stroke fully closed by remote operation. For subsequent NPMs, a test is conducted at reduced pressure and temperature in accordance with FSAR Table 14.2-56 to demonstrate that the ECCS safety-related RRVs and RVVs listed in Table 2.1-4 stroke fully open and fully closed by remote operation.
02.01.15	The DHRS safety-related valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that the DHRS safety-related valves listed in Table 2.1-4 stroke fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
02.01.16	FSAR Section 9.1.5, Overhead Heavy Load Handling Systems, discusses that the CNV top support structure represent a single-failure-proof lifting device in accordance with the requirements of ANSI N14.6. As described in FSAR Section 9.1.5, the CNV top support structure is load tested to 150 percent (+5%, -0%) of the manufacturer's rating in accordance with ANSI N14.6. As part of the rated load test, critical areas of the CNV top support structure, including load-bearing welds, will undergo nondestructive testing as required by ANSI N14.6. This ITAAC test may be performed any time after manufacture of the CNV top support structure (at the factory or later).

Table 2.1-2: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.01.17	FSAR Section 9.1.5 discusses that the CNV top support structure represents a single-failure-proof lifting device in accordance with the requirements of ANSI N14.6. An ITAAC inspection is performed of the CNV top support structure to verify the existence of dual load paths. This ITAAC inspection may be performed any time after manufacture of the CNV top support structure (at the factory or later).
02.01.18	The CNTS safety-related hydraulic-operated valves are tested to demonstrate the capability to perform their function to fail to or maintain their safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that each CNTS safety-related hydraulic-operated valves listed in Table 2.1-4 repositions to or maintains its safety-related position on loss of motive power (electric power to the valve actuating solenoid(s) is lost, or hydraulic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
02.01.19	The ECCS safety-related RRVs and RVVs are tested to demonstrate the capability to perform their function to fail to or maintain their safety-related position on loss of electrical power under preoperational temperature, differential pressure, and flow conditions. For the first NPM only, a test is conducted under preoperational test conditions that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations. The test is initiated with an initial RPV to CNV differential pressure greater than the inadvertent actuation block threshold pressure of 900 psid in accordance with FSAR Table 14.2-40 and demonstrates that each ECCS safety-related valve listed in Table 2.1-4 fails open on loss of electrical power to its corresponding trip valve. For subsequent NPMs a test is conducted at reduced pressure and temperature in accordance with FSAR Table 14.2-56 to demonstrate that each ECCS safety-related valve listed in Table 2.1-4 fails open on loss of electrical power to its corresponding trip valve.
02.01.20	The DHRS safety-related hydraulic-operated valves are tested to demonstrate the capability to perform their function to fail to or maintain their safety-related position on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that each DHRS safety-related hydraulic-operated valves listed in Table 2.1-4 fails open loss of motive power (electric power to the valve actuating solenoid(s) is lost, or hydraulic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
02.01.21	The CNTS safety-related check valves are tested to demonstrate the capability to perform their function to transfer open and transfer closed (under forward and reverse flow conditions, respectively) under preoperational temperature, differential pressure, and flow conditions. Check valves are tested in accordance with the requirements of the ASME OM Code, ISTC-5220, Check Valves. In accordance with FSAR Table 14.2-38, a preoperational test demonstrates that the CNTS check valves listed in Table 2.1-4 strokes fully open and closed under forward and reverse flow conditions, respectively. Preoperational test conditions are established that approximate design basis temperature, differential pressure and flow conditions to the extent practicable, consistent with preoperational test limitations.

Table 2.1-2: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1] (Continued)

ITAAC No.	Discussion
02.01.22	Quality Control inspection hold points are used to ensure the as-built CNTS passive autocatalytic recombiner is installed consistent with the associated installation specification, and therefore capable of performing its safety-related function. To demonstrate the acceptance criterion for ITAAC 02.01.22 is satisfied, and the associated design commitment fully met, a report will exist and conclude Quality Control inspection hold points exist and have been completed for the location and orientation of the PAR.
02.01.23	FSAR Section 6.2.1, Containment Functional Design, and FSAR Table 6.2-2, Containment Pressure, Temperature Response Analysis Initial Conditions, discuss the CNV net free volume that is analyzed in the CNV pressure and temperature response. For the first ever NPM only, a reconciliation analysis of the as-built CNV net free volume is performed to ensure it bounds the assumed CNV net free volume used in the CNV pressure and temperature response. For subsequent NPMs, this ITAAC is not performed.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 2.1-3: NuScale Power Module Piping Systems

Piping System Description	ASME Code Section III Class	High/ Moderate Energy	Length of Containment Piping (ft)
Containment System			
RCS injection line valves CVC-HOV-0331 & CVC-HOV-0330 at CNV nozzle CNV6 to and including CVC-CKV-0329	3	High	0 (Note 1)
RCS pressurizer spray line valves CVC-HOV-0325 & CVC-HOV-0324 at CNV nozzle CNV7 to and including CVC-CKV-0323	3	High	0 (Note 1)
RPV discharge line from valves CVC-HOV-0334 & CVC-HOV-0335 at CNV nozzle CNV13 to and including CVC-AOV-0336	3	High	0 (Note 1)
RPV high point degasification line from valves CVC-HOV-0401 & CVC-HOV-0402 at CNV nozzle CNV14 to and including CVC-SV-0404	3	High	0 (Note 1)
Containment evacuation system (CES) line valves CE-HOV-0001 & CE-HOV-0002 at CNV nozzle CNV10	N/A (Note 2)	No	0 (Note 1)
Containment flood and drain system (CFDS) line valves CFD-HOV-0022 & CFD-HOV-0021 at CNV nozzle CNV11	N/A (Note 2)	No	0 (Note 1)
Reactor component cooling water system (RCCWS) supply line valves RCCW-HOV-0185 & RCCW-HOV-0184 at CNV nozzle CNV12	N/A (Note 2)	No	0 (Note 1)
RCCWS return line valves RCCW-HOV-0190 & RCCW-HOV-0191 at CNV nozzle CNV5	N/A (Note 2)	No	0 (Note 1)
Feedwater line #1 valves FW-HOV-0137 & FW-CKV-0136 at CNV nozzle CNV1	N/A (Note 2)	High	0 (Note 1)
Feedwater line #2 valves FW-HOV-0237 & FW-CKV-0236 at CNV nozzle CNV2	N/A (Note 2)	High	0 (Note 1)
Main steam line #1 from CNV nozzle CNV3 to and including valves MS-HOV-0101 & MS-HOV-0103	2	High	4 (Note 3)
Main steam line #2 from CNV nozzle CNV4 to and including valves MS-HOV-0201 & MS-HOV-0203	2	High	4 (Note 3)
Decay Heat Removal System			
DHRS #1 lines from SG #1 steam line to DHRS condenser train 1 including valves DHR-HOV-0111 and DHR-HOV-0121	2	High	N/A
DHRS #1 condensate line from DHRS condenser train 1 to CNV nozzle CNV22	2	High	N/A
DHRS #2 lines from SG #2 steam line to DHRS condenser train 2 including valves DHR-HOV-0211 and DHR-HOV-0221	2	High	N/A
DHRS #2 condensate line from DHRS condenser train 2 to CNV nozzle CNV23	2	High	N/A
DHRS #1 condensate line from CNV nozzle CNV22 to SG #1 feedwater line	2	High	N/A
DHRS #2 condensate line from CNV nozzle CNV23 to SG #2 feedwater line	2	High	N/A
Reactor Coolant System			
RCS injection line from RPV nozzle RPV11 to CNV nozzle CNV6	1	High	N/A
PZR spray line from RPV nozzles RPV14 and RPV15 to CNV nozzle CNV7	1	High	N/A
RCS discharge line from RPV nozzle RPV12 to CNV nozzle CNV13	1	High	N/A
RPV high point degasification line from RPV nozzle RPV20 to CNV nozzle CNV14	1	High	N/A
Steam Generator System			
SG #1 feedwater line from RPV nozzles RPV3 and RPV5 to CNV nozzle CNV1	2	High	N/A
SG #2 feedwater line from RPV nozzles RPV4 and RPV6 to CNV nozzle CNV2	2	High	N/A

Table 2.1-3: NuScale Power Module Piping Systems (Continued)

Piping System Description	ASME Code Section III Class	High/ Moderate Energy	Length of Containment Piping (ft)
SG #1 steam line from RPV nozzles RPV8 and RPV10 to CNV nozzle CNV3	2	High	N/A
SG #2 steam line from RPV nozzles RPV7 and RPV9 to CNV nozzle CNV4	2	High	N/A

Note:

1) The listed component is welded directly to the associated containment isolation test fixture, which is welded directly to the safe-end, which is part of the CNV.
2) There is no ASME Class 1, 2, or 3 piping between the listed CNTS valves and the associated CNTS removable spool piece flange. The piping between the valves and the CNTS flange is classified as ASME B31.1.
3) The main steam line CNV nozzles include an extended nozzle safe-end. The main steam isolation valve is welded directly to the safe-end.

Table 2.1-4: NuScale Power Module Mechanical Equipment

Equipment Name	Equipment Identifier	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
Reactor Coolant System				
RCS integral RPV/SG/Pressurizer	RPV-VSL-0001	1	N/A	N/A
RVI upper core plate	N/A	CS	N/A	N/A
RVI core barrel	N/A	CS	N/A	N/A
RVI reflector blocks	N/A	CS	N/A	N/A
RVI lower core plate	N/A	CS	N/A	N/A
RVI upper support blocks	N/A	CS	N/A	N/A
RVI core support mounting brackets	RCS-RVI-1003	CS	N/A	N/A
Upper SG supports	N/A	1	N/A	N/A
RCS reactor safety valves	RCS-PSV-0003A RCS-PSV-0003B	1	N/A	No
SG #1 relief valve	SG-RV-0102	2	N/A	Yes
SG #2 relief valve	SG-RV-0202	2	N/A	Yes
RPV instrument seal assemblies	RCS-ISA-039 RCS-ISA-040 RCS-ISA-041 RCS-ISA-042 RCS-ISA-085 RCS-ISA-086	1	N/A	N/A
PZR liquid temperature elements	RCS-TE-1016A RCS-TE-1016B	1	N/A	N/A
PZR vapor temperature elements	RCS-TE-1017A RCS-TE-1017B	1	N/A	N/A
RCS cold temperature elements	RCS-TE-1009A1 RCS-TE-1009B1 RCS-TE-1009C1 RCS-TE-1009D1 RCS-TE-1009A2 RCS-TE-1009B2 RCS-TE-1009C2 RCS-TE-1009D2	1	N/A	N/A
RCS hot temperature elements	RCS-TE-1005A1 RCS-TE-1005B1 RCS-TE-1005C1 RCS-TE-1005D1 RCS-TE-1005A2 RCS-TE-1005B2 RCS-TE-1005C2 RCS-TE-1005D2	1	N/A	N/A
Emergency Core Cooling System				
RVVs	ECC-POV-0001A ECC-POV-0001B	1	Hydraulic	No
RVV trip valves	ECC-SV-0101A ECC-SV-0101B ECC-SV-0102A ECC-SV-0102B	1	Solenoid	No
RVV reset valves	ECC-SV-0103A ECC-SV-0103B	1	Solenoid	No
RVV valve venturis	ECC-FV-0001A ECC-FV-0001B	1	N/A	No

Table 2.1-4: NuScale Power Module Mechanical Equipment (Continued)

Equipment Name	Equipment Identifier	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
RRVs	ECC-POV-0002A ECC-POV-0002B	1	Hydraulic	No
RRV trip valves	ECC-SV-0104A ECC-SV-0104B ECC-SV-0105A ECC-SV-0105B	1	Solenoid	No
ECCS RRV reset valves	ECC-SV-0106A ECC-SV-0106B	1	Solenoid	No
RRV valve venturis	ECC-FV-0002A ECC-FV-0002B	1	N/A	No
Containment System				
CNV	CNT-VSL-0001	1	N/A	N/A
RPV high point degas solenoid valve	CVC-SV-0404	3	Solenoid	No
PZR spray flow check valve	CVC-CKV-0323	3	N/A	No
CVCS injection flow check valve	CVC-CKV-0329	3	N/A	No
CVCS discharge air operated valve	CVC-AOV-0336	3	Air	No
CVCS injection inboard CIV	CVC-HOV-0331	1	Electro-hydraulic	Yes
CVCS injection outboard CIV	CVC-HOV-0330	1	Electro-hydraulic	Yes
PZR spray inboard CIV	CVC-HOV-0325	1	Electro-hydraulic	Yes
PZR spray outboard CIV	CVC-HOV-0324	1	Electro-hydraulic	Yes
CVCS discharge inboard CIV	CVC-HOV-0334	1	Electro-hydraulic	Yes
CVCS discharge outboard CIV	CVC-HOV-0335	1	Electro-hydraulic	Yes
RPV high point degas inboard CIV	CVC-HOV-0401	1	Electro-hydraulic	Yes
RPV high point degas outboard CIV	CVC-HOV-0402	1	Electro-hydraulic	Yes
CES inboard CIV	CE-HOV-0001	2	Electro-hydraulic	Yes
CES outboard CIV	CE-HOV-0002	2	Electro-hydraulic	Yes
CFDS inboard CIV	CFD-HOV-0022	2	Electro-hydraulic	Yes
CFDS outboard CIV	CFD-HOV-0021	2	Electro-hydraulic	Yes
RCCWS supply inboard CIV	RCCW-HOV-0185	2	Electro-hydraulic	Yes
RCCWS supply outboard CIV	RCCW-HOV-0184	2	Electro-hydraulic	Yes
RCCWS return inboard CIV	RCCW-HOV-0190	2	Electro-hydraulic	Yes
RCCWS return outboard CIV	RCCW-HOV-0191	2	Electro-hydraulic	Yes
Feedwater isolation valve (FWIV) #1	FW-HOV-0137	2	Electro-hydraulic	Yes
Feedwater isolation check valve #1	FW-CKV-0136	2	N/A	No

Table 2.1-4: NuScale Power Module Mechanical Equipment (Continued)

Equipment Name	Equipment Identifier	ASME Code Section III Class	Valve Actuator Type	Containment Isolation Valve
FWIV #2	FW-HOV-0237	2	Electro-hydraulic	Yes
Feedwater isolation check valve #2	FW-CKV-0236	2	N/A	No
Main steam isolation valve (MSIV) #1	MS-HOV-0101	2	Electro-hydraulic	Yes
Main steam isolation bypass valve (MSIBV) #1	MS-HOV-0103	2	Electro-hydraulic	Yes
MSIV #2	MS-HOV-0201	2	Electro-hydraulic	Yes
MSIBV #2	MS-HOV-0203	2	Electro-hydraulic	Yes
I&C Division I EPA	CNV8	1	N/A	N/A
I&C Division II EPA	CNV9	1	N/A	N/A
PZR heater power division I nozzle EPA	CNV15	1	N/A	N/A
PZR heater power division II nozzle EPA	CNV16	1	N/A	N/A
I&C Channel A instrument seal assembly	CNV17	1	N/A	N/A
I&C Channel C instrument seal assembly	CNV18	1	N/A	N/A
I&C Channel B instrument seal assembly	CNV19	1	N/A	N/A
I&C Channel D instrument seal assembly	CNV20	1	N/A	N/A
CRDM power 1 nozzle EPA	CNV37	1	N/A	N/A
RPI group #1 EPA	CNV38	1	N/A	N/A
RPI group #2 EPA	CNV39	1	N/A	N/A
I&C separation group A EPA	CNV40	1	N/A	N/A
I&C separation group B EPA	CNV41	1	N/A	N/A
I&C separation group C EPA	CNV42	1	N/A	N/A
I&C separation group D EPA	CNV43	1	N/A	N/A
CRDM power 2 nozzle EPA	CNV44	1	N/A	N/A
Feedwater #1 containment isolation text fixture (CITF) valve	FW-HV-0901	2	N/A	No
Feedwater #2 CITF valve	FW-HV-0902	2	N/A	No
RCCWS supply CITF valve	RCCW-HV-0912	2	N/A	No
RCCWS return CITF valve	RCCW-HV-0905	2	N/A	No
CVCS injection CITF valve	CVC-HV-0906	1	N/A	No
CVCS pressurizer spray CITF valve	CVC-HV-0907	1	N/A	No
CVCS discharge CITF valve	CVC-HV-0913	1	N/A	No
RCS high point degas CITF valve	CVC-HV-0914	1	N/A	No
CES CITF valve	CE-HV-0910	2	N/A	No
CFDS CITF valve	CFD-HV-0911	2	N/A	No
Decay Heat Removal System				
DHRS actuation valves	DHR-HOV-0111 DHR-HOV-0121 DHR-HOV-0211 DHR-HOV-0221	2	Electro-hydraulic	No
DHRS condensers	DHR-CND-0103 DHR-CND-0203	2	N/A	N/A
Control Rod Drive System				
CRDM pressure housing and removable top plug	N/A	1	N/A	N/A
CRDM-to-RPV joint sealing components	CRDS-PBS-10xx (xx = 01:16)	1	N/A	N/A

Table 2.1-5: NuScale Power Module Electrical Equipment

Equipment Name	Equipment Identifier	Remotely Operated	Loss of Motive Power Position	Containment Isolation Valve
Emergency Core Cooling System				
RVV trip valves	ECC-SV-0101A-1 ECC-SV-0101A-2 ECC-SV-0101B-1 ECC-SV-0101B-2	Yes	Open	No
RVV reset valves	ECC-SV-0103A ECC-SV-0103B	Yes	Close	No
RRV trip valves	ECC-SV-0102A-1 ECC-SV-0102A-2 ECC-SV-0102B-1 ECC-SV-0102B-2	Yes	Open	No
RRV reset valves	ECC-SV-0104A ECC-SV-0104B	Yes	Close	No
Containment System				
CVCS injection inboard CIV	CVC-HOV-0331	Yes	Closed	Yes
CVCS injection outboard CIV	CVC-HOV-0330	Yes	Closed	Yes
PZR spray inboard CIV	CVC-HOV-0325	Yes	Closed	Yes
PZR spray outboard CIV	CVC-HOV-0324	Yes	Closed	Yes
CVCS discharge inboard CIV	CVC-HOV-0334	Yes	Closed	Yes
CVCS discharge outboard CIV	CVC-HOV-0335	Yes	Closed	Yes
RPV high point degas inboard CIV	CVC-HOV-0401	Yes	Closed	Yes
RPV high point degas outboard CIV	CVC-HOV-0402	Yes	Closed	Yes
CES inboard CIV	CE-HOV-0001	Yes	Closed	Yes
CES outboard CIV	CE-HOV-0002	Yes	Closed	Yes
CFDS inboard CIV	CFD-HOV-0022	Yes	Closed	Yes
CFDS outboard CIV	CFD-HOV-0021	Yes	Closed	Yes
RCCWS supply inboard CIV	RCCW-HOV-0185	Yes	Closed	Yes
RCCWS supply outboard CIV	RCCW-HOV-0184	Yes	Closed	Yes
RCCWS return inboard CIV	RCCW-HOV-0190	Yes	Closed	Yes
RCCWS return outboard CIV	RCCW-HOV-0191	Yes	Closed	Yes
FWIV #1	FW-HOV-0137	Yes	Closed	Yes
FWIV #2	FW-HOV-0237	Yes	Closed	Yes
MSIV #1	MS-HOV-0101	Yes	Closed	Yes
MSIBV #1	MS-HOV-0103	Yes	Closed	Yes
MSIV #2	MS-HOV-0201	Yes	Closed	Yes
MSIBV #2	MS-HOV-0203	Yes	Closed	Yes
I&C Division I EPA	CNV8	N/A	N/A	N/A
I&C Division II EPA	CNV9	N/A	N/A	N/A
PZR heater power division I nozzle EPA	CNV15	N/A	N/A	N/A
PZR heater power division II nozzle EPA	CNV16	N/A	N/A	N/A
I&C Channel A instrument seal assembly	CNV17	N/A	N/A	N/A
CNTS I&C Channel C instrument seal assembly	CNV18	N/A	N/A	N/A
CNTS I&C Channel B instrument seal assembly	CNV19	N/A	N/A	N/A
CNTS I&C Channel D instrument seal assembly	CNV20	N/A	N/A	N/A
CNTS control rod drive mechanism power 1 nozzle EPA	CNV37	N/A	N/A	N/A
CNTS rod position indicator group #1 EPA	CNV38	N/A	N/A	N/A
CNTS rod position indicator group #2 EPA	CNV39	N/A	N/A	N/A
I&C separation group A EPA	CNV40	N/A	N/A	N/A

Table 2.1-5: NuScale Power Module Electrical Equipment (Continued)

Equipment Name	Equipment Identifier	Remotely Operated	Loss of Motive Power Position	Containment Isolation Valve
I&C separation group B EPA	CNV41	N/A	N/A	N/A
I&C separation group C EPA	CNV42	N/A	N/A	N/A
I&C separation group D EPA	CNV43	N/A	N/A	N/A
CRDM power 2 nozzle EPA	CNV44	N/A	N/A	N/A
Decay Heat Removal System				
DHRS actuation valves	DHR-HOV-0111 DHR-HOV-0121 DHR-HOV-0211 DHR-HOV-0221	Yes	Open	No

Table 2.1-6: NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
02.01.01	X					
02.01.02	X					
02.01.03	X					
02.01.04	X	X				
02.01.05	X					
02.01.06	X					
02.01.07	X					
02.01.08	X					
02.01.09	X					
02.01.10	X					
02.01.11	X					
02.01.12	X					
02.01.13	X					
02.01.14	X					
02.01.15	X					
02.01.16				X		
02.01.17				X		
02.01.18	X					
02.01.19	X					
02.01.20	X					
02.01.21	X					
02.01.22	X					
02.01.23	X					

2.2 Chemical and Volume Control System Module

2.2.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the CVCS, which is described in FSAR Section 9.3.4. Each NPM has its own module-specific CVCS.

The CVCS performs the following safety-related system function that is verified by ITAAC.

- The CVCS supports the RCS by isolating dilution sources.

Design Commitments

- The CVCS demineralized water supply isolation valves listed in Table 2.2-3 change position under design-basis temperature, differential pressure, and flow conditions.
- The CVCS demineralized water supply isolation valves listed in Table 2.2-3 perform their function to fail to (or maintain) the closed position on loss of motive power under design-basis temperature, differential pressure, and flow conditions.
- The CVCS demineralized water supply isolation valves listed in Table 2.2-3 comply with ASME Code Section III requirements for Code Class 3 components.

2.2.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.2-1 contains the ITAAC for the CVCS.

Table 2.2-1: Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.02.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The CVCS demineralized water supply isolation valves listed in Table 2.2-3 change position under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the CVCS demineralized water supply isolation valves listed in Table 2.2-3 under preoperational temperature, differential pressure, and flow conditions.	Each CVCS demineralized water supply isolation valve listed in Table 2.2-3 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
02.	The CVCS demineralized water supply isolation valves listed in Table 2.2-3 perform their function to fail to (or maintain) the closed position on loss of motive power under design-basis temperature, differential pressure, and flow conditions.	A test will be performed of the CVCS demineralized water supply isolation valves listed in Table 2.2-3 under preoperational temperature, differential pressure and flow conditions.	Each CVCS demineralized water supply isolation valve listed in Table 2.2-3 performs its function to fail to (or maintain) the closed position on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
03.	The CVCS demineralized water supply isolation valves listed in Table 2.2-3 comply with ASME Code Section III requirements for Code Class 3 components.	i. An inspection will be performed of the ASME Code Class 3 as-built component Design Reports required by ASME Code Section III for the CVCS demineralized water supply isolation valves listed in Table 2.2-3. ii. An inspection will be performed of the ASME Code Class 3 as-built component Data Reports required by ASME Code Section III for the CVCS demineralized water supply isolation valves listed in Table 2.2-3.	i. The ASME Code Section III Design Reports (NCA-3550) exist and conclude that the as-built CVCS demineralized water supply isolation valves listed in Table 2.2-3 meet the ASME Code Section III requirements for Code Class 3 components. ii. The ASME Code Section III Data Reports exist and conclude that the as-built CVCS demineralized water supply isolation valves listed in Table 2.2-3 meet the ASME Code Section III requirements for Code Class 3 components.

Table 2.2-2: Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
02.02.01	The safety-related chemical and volume control demineralized water supply isolation valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with the information provided in FSAR Table 14.2-33, a preoperational test demonstrates that the safety-related chemical and volume control demineralized water supply isolation valves stroke fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
02.02.02	The safety-related chemical and volume control demineralized water supply isolation valves are tested to demonstrate the capability to perform their function to fail to the closed position on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with FSAR Table 14.2-33, a preoperational test demonstrates that each safety-related chemical and volume control demineralized water supply isolation valve fails closed on loss of motive power (electric power to the valve actuating solenoid(s) is lost, or pneumatic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
02.02.03	As required by ASME Code Section III NCA-1210, each ASME Code Class 3 component of a nuclear power plant requires a Design Report in accordance with NCA-3550. NCA-3551.1 requires that the drawings used for construction be in agreement with the Design Report before it is certified and be identified and described in the Design Report. It is the responsibility of the N certificate holder to furnish a Design Report for each component and support, except as provided in NCA-3551.2 and NCA-3551.3. NCA-3551.1 also requires that the Design Report be certified by a registered professional engineer when it is for Class 3 components designed to service loadings greater than design loadings. NCA-3554 requires that any modification of any document used for construction, from the corresponding document used for design analysis, shall be reconciled with the Design Report. An ITAAC inspection is performed of the ASME Code Class 3 as-built Design Reports for the CVCS demineralized water supply isolation valves listed in Table 2.2-3 to verify that the requirements of ASME Code Section III are met. The ASME Code Section III requires that documentary evidence be available at the construction or installation site before use or installation to ensure that ASME Code Class 3 components conform to the requirements of the Code. As defined in NCA-9000, a component can be a vessel, pump, pressure relief valve, line valve, storage tank, piping system, or core support structure that is designed, constructed, and stamped in accordance with the rules of Section III. The ASME Code Class 3 CVCS demineralized water supply isolation valves listed in Table 2.2-3 require a Data Report as specified by NCA-1210. The Data Report is prepared by the certificate holder or owner and signed by the certificate holder or owner and the inspector as specified by NCA-8410. The type of individual Data Report forms necessary to record the required code data is specified in Table NCA-8100-1. An ITAAC inspection is performed of the ASME Code Class 3 as-built Data Reports for the CVCS demineralized water supply isolation valves listed in Table 2.2-3 to (1) ensure that the appropriate Data Reports have been provided as specified in Table NCA-8100-1, (2) ensure that the certificate holder or owner and the authorized nuclear inspector have signed the Data Reports, and (3) verify that the requirements of ASME Code Section III are met.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 2.2-3: Chemical and Volume Control System Mechanical Equipment

Equipment Name	Equipment Identifier	ASME Section III Code Class	Loss of Motive Power Position
Demineralized water system supply (DWS) isolation valves	CVC-AOV-0089 CVC-AOV-0090	3	Closed

Table 2.2-4: Chemical and Volume Control System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
02.02.01	X					
02.02.02	X					
02.02.03	X					

2.3 Containment Evacuation System

2.3.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the CES, which is described in FSAR Section 9.3.6. Each NPM has its own module-specific CES.

The CES performs the following nonsafety-related system functions that are verified by ITAAC. The CES supports the RCS by providing RCS leak detection monitoring capability.

Design Commitments

- The CES sample vessel level instrumentation supports RCS leakage detection.
- The CES inlet pressure instrumentation supports RCS leakage detection.

2.3.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.3-1 contains the ITAAC for the CES.

Table 2.3-1: Containment Evacuation System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.03.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The CES sample vessel level instrumentation supports RCS leakage detection.	A test will be performed of the CES sample vessel level instrumentation.	The CES sample vessel level instrumentation detects a level increase in the CES sample vessel, which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour.
02.	The CES inlet pressure instrumentation supports RCS leakage detection.	A test will be performed of the CES inlet pressure instrumentation.	The CES inlet pressure instrumentation detects a pressure increase in CES inlet pressure, which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour.

Table 2.3-2: Containment Evacuation System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
02.03.01	FSAR Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection, discusses that RCS leakage detection systems are designed to detect and, to the extent practicable, identify the source of reactor coolant leakage. The RCS leakage detection systems conform to the guidance of RG 1.45, regarding detection, monitoring, quantifying, and identification of reactor coolant leakage. In accordance with the information provided in FSAR Table 14.2-36, a preoperational test demonstrates that the CES detects a level increase in the CES sample vessel, which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour. Water vapor and non-condensable gases are removed from the CNV by the CES. The water vapor is collected and condensed in the CES sample vessel where it is monitored using level and temperature instrumentation. The CES sample vessel level instrumentation is used to quantify and trend leak rates in the containment.
02.03.02	FSAR Section 5.2.5, Reactor Coolant Pressure Boundary Leakage Detection, discusses that RCS leakage detection systems are designed to detect and, to the extent practicable, identify the source of reactor coolant leakage. The RCS leakage detection systems conform to the guidance of RG 1.45, regarding detection, monitoring, quantifying, and identification of reactor coolant leakage. In accordance with FSAR Table 14.2-36, a preoperational test demonstrates that the CES is capable of detecting a pressure increase in the CES inlet pressure instrumentation (PIT-1001/PIT-1019), which correlates to a detection of an unidentified RCS leakage rate of one gpm within one hour.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 2.3-3: Containment Evacuation System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
02.03.01	X					
02.03.02	X					

2.4 Equipment Qualification - Module-Specific

2.4.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is equipment qualification of equipment specific to each NPM. Equipment qualification applies to safety-related electrical and mechanical equipment and safety-related digital I&C equipment.

Additionally, this section applies to a limited population of module-specific, nonsafety-related equipment that has augmented Seismic Category I or environmental qualification requirements. The nonsafety-related equipment in this section has one of the following design features. Nonsafety-related mechanical and electrical equipment located within the boundaries of the NPM that has an augmented Seismic Category I or environmental qualification design requirement. Nonsafety-related mechanical and electrical equipment that performs a credited function in Chapter 15 analyses (secondary MSIVs, feedwater regulating valves (FWRVs) and secondary feedwater check valves).

Design Commitments

- The module-specific Seismic Category I equipment listed in Table 2.4-3, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after an SSE.
- The module-specific electrical equipment located in a harsh environment listed in Table 2.4-3, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, anticipated operational occurrences (AOOs), design basis accidents (DBAs), and post-accident conditions, and performs its function for the period of time required to complete the function.
- The non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.4-3 perform their function up to the end of their qualified life in the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) experienced during normal operations, AOOs, DBAs, and post-accident conditions.
- The Class 1E computer-based instrumentation and control systems listed in Table 2.4-3 located in a mild environment withstand design basis mild environmental conditions without loss of safety-related functions.
- The Class 1E digital equipment listed in Table 2.4-3 performs its safety-related function when subjected to the design basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA.
- The valves listed in Table 2.4-3 are functionally designed and qualified to perform their safety-related function under the full range of fluid flow, differential pressure, electrical, temperature, and fluid conditions up to and including DBA conditions.

- The safety-related relief valves listed in Table 2.4-3 provide overpressure protection.
- The DHRS condensers listed in Table 2.4-3 have the capacity to transfer their design heat load.
- The CNTS containment electrical penetration assemblies listed in Table 2.4-3, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and post-accident conditions, and performs its function for the period of time required to complete the function.
- The CNTS passive autocatalytic recombiner provides the safety-related function to control combustible gas within the CNV for design-basis events.
- The CNTS passive autocatalytic recombiner performs its function up to the end of its qualified life in the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) experienced during normal operations, AOOs, DBAs, and post-accident conditions.

2.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.4-1 contains the ITAAC for the equipment qualification - module-specific equipment.

Table 2.4-1: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.04.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The module-specific Seismic Category I equipment listed in Table 2.4-3, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after an SSE.	i. A type test, analysis, or a combination of type test and analysis will be performed of the module-specific Seismic Category I equipment listed in Table 2.4-3, including its associated supports and anchorages.	i. A Seismic Qualification Report exists and concludes that the module-specific Seismic Category I equipment listed in Table 2.4-3, including its associated supports and anchorages, will withstand the design basis seismic loads and perform its function(s) during and after an SSE.
		ii. An inspection will be performed of the module-specific Seismic Category I as-built equipment listed in Table 2.4-3, including its associated supports and anchorages.	ii. The module-specific Seismic Category I equipment listed in Table 2.4-3, including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the equipment's Seismic Qualification Report.
02.	The module-specific electrical equipment located in a harsh environment listed in Table 2.4-3, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and post-accident conditions and performs its function for the period of time required to complete the function.	i. A type test or a combination of type test and analysis will be performed of the module-specific electrical equipment listed in Table 2.4-3, including associated connection assemblies.	i. An EQ record form exists and concludes that the module-specific electrical equipment listed in Table 2.4-3, including associated connection assemblies, perform their function under the environmental conditions specified in the EQ record form for the period of time required to complete the function.
		ii. An inspection will be performed of the module-specific as-built electrical equipment listed in Table 2.4-3, including associated connection assemblies.	ii. The module-specific electrical equipment listed in Table 2.4-3, including associated connection assemblies, are installed in their design location in a configuration bounded by the EQ record form.
03.	The non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.4-3 perform their function up to the end of their qualified life in the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) experienced during normal operations, AOOs, DBAs, and post-accident conditions.	A type test or a combination of type test and analysis will be performed of the non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.4-3.	A qualification record form exists and concludes that the non-metallic parts, materials, and lubricants used in module-specific mechanical equipment listed in Table 2.4-3 perform their function up to the end of their qualified life under the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) specified in the qualification record form.

Table 2.4-1: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.04.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
04.	The Class 1E computer-based instrumentation and control systems listed in Table 2.4-3 located in a mild environment withstand design basis mild environmental conditions without loss of safety-related functions.	i. A type test or a combination of type test and analysis will be performed of the Class 1E computer-based instrumentation and control systems listed in Table 2.4-3 located in a mild environment. ii. An inspection will be performed of the Class 1E as-built computer- based instrumentation and control systems listed in Table 2.4-3 located in a mild environment.	i. An EQ record form exists and concludes that the Class 1E computer-based instrumentation and control systems listed in Table 2.4-3 located in a mild environment perform their function under the environmental conditions specified in the EQ record form. ii. The Class 1E computer-based instrumentation and control systems listed in Table 2.4-3 located in a mild environment are installed in their design location in a configuration bounded by the EQ record form.
05.	The Class 1E digital equipment listed in Table 2.4-3 performs its safety-related function when subjected to the design basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA.	A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E digital equipment listed in Table 2.4-3.	An EQ record form exists and concludes that the Class 1E digital equipment listed in Table 2.4-3 withstands the design basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA without loss of safety-related function.
06.	The valves listed in Table 2.4-3 are functionally designed and qualified to perform their safety-related function under the full range of fluid flow, differential pressure, electrical, temperature, and fluid conditions up to and including DBA conditions.	A type test or a combination of type test and analysis will be performed of the valves listed in Table 2.4-3.	A Qualification Report exists and concludes that the valves listed in Table 2.4-3 are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical, temperature, and fluid conditions up to and including DBA conditions.
07.	The safety-related relief valves listed in Table 2.4-3 provide overpressure protection.	i. A vendor test will be performed of each safety-related relief valve listed in Table 2.4-3. ii. An inspection will be performed of each safety-related as-built relief valve listed in Table 2.4-3.	i. An ASME Code Section III Data Report exists and concludes that the relief valves listed in Table 2.4-3 meet the valve's required set pressure, capacity, and overpressure design requirements. ii. Each relief valve listed in Table 2.4-3 is provided with an ASME Code Certification Mark that identifies the set pressure, capacity, and overpressure.
08.	The DHRS condensers listed in Table 2.4-3 have the capacity to transfer their design heat load.	A type test or a combination of type test and analysis will be performed of the DHRS condensers listed in Table 2.4-3.	A report exists and concludes that the DHRS condensers listed in Table 2.4-3 have a heat removal capacity sufficient to transfer their design heat load.

Table 2.4-1: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.04.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
09.	The CNTS containment electrical penetration assemblies listed in Table 2.4-3, including associated connection assemblies, withstand the design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and post-accident conditions and performs its function for the period of time required to complete the function.	i. A type test or a combination of type test and analysis will be performed of the CNTS containment electrical penetration assemblies listed in Table 2.4-3 including associated connection assemblies.	i. An EQ record form exists and concludes that the CNTS electrical penetration assemblies listed in Table 2.4-3, including associated connection assemblies, performs their function under the environmental conditions specified in the EQ record form for the period of time required to complete the function.
		ii. An inspection will be performed of the containment CNTS electrical penetration assembles listed in Table 2.4-3, including associated connection assemblies.	ii. The CNTS electrical penetration assemblies listed in Table 2.4-3, including associated connection assemblies, are installed in their design location in a configuration bounded by the EQ record form.
10.	The CNTS passive autocatalytic recombiner provides the safety-related function to control combustible gas within the CNV for design-basis events.	A type test, analysis, or a combination of type test and analysis will be performed of the CNTS passive autocatalytic recombiner.	A report exists and concludes that the PAR has sufficient capacity to meet or exceed the minimum required oxygen recombination rate.
11.	The CNTS passive autocatalytic recombiner performs its function up to the end of its qualified life in the design basis harsh environmental conditions (both internal service conditions and external environmental conditions) experienced during normal operations, AOOs, DBAs, and post-accident conditions.	An analysis will be performed of the CNTS passive autocatalytic recombiner.	A qualification record form exists and concludes that the CNTS passive autocatalytic recombiner performs its function up to the end of its qualified life under the design basis harsh environmental conditions specified in the qualification record form.

Table 2.4-2: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
02.04.01	FSAR Section 3.10, Seismic and Dynamic Qualifications of Mechanical and Electrical Equipment, presents information to demonstrate that the Seismic Category I equipment, including its associated supports and anchorages, is qualified by type test, analysis, or a combination of type test and analysis to perform its function under the design basis seismic loads during and after an SSE. The qualification method employed for the Seismic Category I equipment is the same as the qualification method described for that type of equipment in Section 3.10. This method conforms to Institute of Electrical and Electronics Engineers (IEEE)-344 and ASME QME-1 (as referenced in FSAR Section 3.10). The scope of equipment for this design commitment is module-specific, safety-related equipment, and module-specific, nonsafety-related equipment that has one of the following design features: • Nonsafety-related mechanical and electrical equipment located within the boundaries of the NPM that has an augmented Seismic Category I design requirement. • Nonsafety-related mechanical and electrical equipment that performs a credited function in Chapter 15 analyses (secondary MSIVs, FWRVs, and secondary feedwater check valves.) The ITAAC verifies that: (1) a Seismic Qualification Report exists for each Seismic Category I component type, and (2) the Seismic Qualification Report concludes that the Seismic Category I equipment listed in Table 2.4-3, including its associated supports and anchorages, performs its function under the seismic design basis load conditions specified in the Seismic Qualification Report. After installation in the plant, an ITAAC inspection is performed to verify that the Seismic Category I equipment listed in Table 2.4-3, including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the Seismic Qualification Report.
02.04.02	FSAR Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the electrical equipment, including its connection assemblies, located in a harsh environment is qualified by type test or a combination of type test and analysis to perform its function under design basis harsh environmental conditions, experienced during normal operations, AOOs, DBAs, and post-accident conditions in accordance with 10 CFR 50.49. As defined in IEEE-Std-572-2006, IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations, a connection assembly is any connector or termination combined with related cables or wires as an assembly. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in FSAR Section 3.11. The scope of equipment for this design commitment is module-specific, Class 1E equipment located within a harsh environment, and module-specific, nonsafety-related equipment with an augmented equipment qualification design requirement located within the boundaries of the NPM. The ITAAC verifies that: (1) an equipment qualification record form exists for the electrical equipment listed in Table 2.4-3 and addresses connection assemblies, (2) the equipment qualification record form concludes that the electrical equipment, including its connection assemblies, performs its function under the environmental conditions specified in Section 3.11 and the equipment qualification record form, and (3) the required post-accident operability time for the electrical equipment in the equipment qualification record form is in agreement with Section 3.11. After installation in the plant, an ITAAC inspection is performed to verify that the electrical equipment listed in Table 2.4-3, including its connection assemblies, is installed in its design location in a configuration bounded by the equipment qualification record form.

Table 2.4-2: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1] (Continued)

ITAAC No.	Discussion
02.04.03	FSAR Section 3.11 presents information to demonstrate that the non-metallic parts, materials, and lubricants used in mechanical equipment located in a harsh environment are qualified using a type test or a combination of type test and analysis to perform their function up to the end of their qualified life in design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and post-accident conditions. Environmental conditions include both internal service conditions and external environmental conditions for the nonmetallic parts, materials, and lubricant. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in FSAR Section 3.11. The scope of equipment for this design commitment is module-specific, safety-related mechanical equipment, and module-specific, nonsafety-related mechanical equipment that performs a credited function in Chapter 15 analyses (secondary MSIVs, FWRVs, and secondary feedwater check valves). The ITAAC verifies that: (1) an equipment qualification record form or ASME QME-1 report exists for the non-metallic parts, materials, and lubricants used in mechanical equipment designated for a harsh environment, and (2) the qualification record form concludes that the non-metallic parts, materials, and lubricants used in mechanical equipment listed in Table 2.4-3 perform their intended function up to the end of its qualified life under the design basis environmental conditions (both internal service conditions and external environmental conditions) specified in the qualification record form.
02.04.04	FSAR Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, and Appendix 3C, Methodology for Environmental Qualification of Electrical and Mechanical Equipment, presents information to demonstrate that the Class 1E computer-based I&C systems located in a mild environment is qualified by type test or a combination of type test and analysis to perform its safety-related function under the design basis mild environmental conditions. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in Section 3.11 and Appendix 3C. This method conforms to IEEE-323-2003 (or later editions), as accepted by the NRC staff in RG 1.209, Revision 0 (or later revision), with specific revision years and numbers as presented in FSAR Section 3.10. The ITAAC verifies that: (1) an equipment qualification record form exists for the Class 1E computer-based I&C systems listed in Table 2.4-3, and (2) the equipment qualification record form concludes that the Class 1E computer-based I&C systems performs its safety-related function under the design basis mild environmental conditions specified in Section 3.11 and Appendix 3C and the equipment qualification record form. After installation in the plant, an ITAAC inspection is performed to verify that the Class 1E computer-based I&C systems listed in Table 2.4-3 is installed in its design location in a configuration bounded by its equipment qualification record form.
02.04.05	FSAR Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the Class 1E digital equipment is qualified using a type test, analysis, or a combination of type test and analysis to perform its safety-related function when subjected to electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA. The qualification method employed for Class 1E digital equipment is the same as the qualification method described for that type of equipment in FSAR Section 3.11. The ITAAC verifies that: (1) an equipment qualification record form exists for the Class 1E digital equipment listed in Table 2.4-3, and (2) the equipment qualification record form concludes that the Class 1E digital equipment withstands the design basis electromagnetic interference, radio frequency interference, and electrical surges that would exist before, during, and following a DBA without loss of safety-related function.

Table 2.4-2: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.04.06	FSAR Section 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, and Section 3.10.2, Methods and Procedures for Qualifying Mechanical and Electrical Equipment and Instrumentation, discuss that the functional qualification of safety-related valves is performed in accordance with ASME QME-1 (as referenced in FSAR Section 3.9). The qualification method employed for the valves agrees with the qualification method described in FSAR Section 3.10.2. The ITAAC verifies that: (1) a Qualification Report exists for the safety-related valves listed in Table 2.4-3, and (2) the Qualification Report concludes that safety-related valves are capable of performing their safety-related function under the full range of fluid flow, differential pressure, electrical conditions, temperature conditions, and fluid conditions up to and including DBA conditions.
02.04.07	FSAR Section 3.9.3, ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures, discusses that relief valves provide overpressure protection in accordance with the ASME Code Section III. The ITAAC verifies that: (1) the test for each relief valve listed in Table 2.4-3 meets the set pressure, capacity, and overpressure design requirements; and (2) each relief valve listed in Table 2.4-3 is provided with an ASME Code Certification Mark that identifies the valve's set pressure, capacity, and overpressure.
02.04.08	FSAR Section 5.4.3, Decay Heat Removal System, discusses that the DHRS passive condensers provide the safety-related function of transferring their design heat load from the DHRS during shutdown. After manufacture of the DHRS passive condensers, a type test or a combination of type test and analysis is performed to validate that the DHRS passive condensers are capable of meeting the specified heat transfer performance requirements. FSAR Section 5.4.3 discusses the design heat transfer capability of the DHRS passive condensers. The ITAAC verifies that the safety-related passive condensers listed in Table 2.4-3 have a heat removal capacity sufficient to transfer their design heat load.
02.04.09	FSAR Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the CNTS electrical penetration assemblies, including its connection assemblies, located in a harsh environment are qualified by type test or a combination of type test and analysis to perform its safety-related function under design basis harsh environmental conditions, experienced during normal operations, AOOs, DBAs, and post-accident conditions in accordance with 10 CFR 50.49. As defined in IEEE-Std-572-2006, IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations, a connection assembly is any connector or termination combined with related cables or wires as an assembly. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in FSAR Section 3.11. The ITAAC verifies that: (1) an equipment qualification record form exists for the CNTS electrical penetration assemblies listed in Table 2.4-3 and addresses connection assemblies; (2) the equipment qualification record form concludes that the CNTS electrical penetration assemblies, including its connection assemblies, performs its safety-related function under the environmental conditions specified in FSAR Section 3.11 and the equipment qualification record form; and (3) the required post-accident operability time for the CNTS electrical penetration assemblies in the equipment qualification record form is in agreement with FSAR Section 3.11. After installation in the plant, an ITAAC inspection is performed to verify that the CNTS electrical penetration assemblies listed in Table 2.4-3, including its connection assemblies, is installed in its design location in a configuration bounded by the equipment qualification record form.

Table 2.4-2: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1] (Continued)

ITAAC No.	Discussion
02.04.10	FSAR Section 6.2.5, Combustible Gas Control in the Containment Vessel, discusses that the PAR provides the safety-related function of maintaining an inert atmosphere (i.e., less than 4 percent oxygen by volume) in the CNV, which is achieved by the continuous recombination of oxygen. FSAR Section 6.2.5 lists the minimum design oxygen recombination rate (in moles per hour) for the PAR to ensure the CNV atmosphere remains inert following design-basis events. This ITAAC verifies that the PAR oxygen recombination rate meets or exceeds the minimum required oxygen recombination rate specified in FSAR Section 6.2.5 to maintain the CNV atmosphere inert during design-basis events.
02.04.11	FSAR Section 3.11 presents information to demonstrate that the CNTS passive autocatalytic recombiner located in a harsh environment is qualified using an analysis to perform its function up to the end of its qualified life in design basis harsh environmental conditions experienced during normal operations, AOOs, DBAs, and post-accident conditions. Environmental conditions include both internal service conditions and external environmental conditions for the PAR. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in FSAR Section 3.11. The ITAAC verifies that: (1) an equipment qualification record form exists for the PAR, and (2) the qualification record form concludes that the PAR listed in Table 2.4-3 perform its intended function up to the end of its qualified life under the design basis environmental conditions (both internal service conditions and external environmental conditions) specified in the qualification record form.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[(1)]
Containment System						
CNV8	I&C Division I EPA	Harsh	Electrical Mechanical	Yes	Yes	A
CNV9	I&C Division II EPA	Harsh	Electrical Mechanical	Yes	Yes	A
CNV15	PZR heater power division I nozzle EPA	Harsh	Electrical Mechanical	Yes	No	B
CNV16	PZR heater power division II nozzle EPA	Harsh	Electrical Mechanical	Yes	No	B
CNV17	I&C Channel A instrument seal assembly	Harsh	Mechanical	Yes	No	B
CNV18	I&C Channel C instrument seal assembly	Harsh	Mechanical	Yes	No	B
CNV19	I&C Channel B instrument seal assembly	Harsh	Mechanical	Yes	No	B
CNV20	I&C Channel D instrument seal assembly	Harsh	Mechanical	Yes	No	B
CNV37	CRDM power 1 nozzle EPA	Harsh	Electrical Mechanical	Yes	No	B
CNV38	RPI group #1 EPA	Harsh	Electrical Mechanical	Yes	No	B
CNV39	RPI group #2 EPA	Harsh	Electrical Mechanical	Yes	No	B
CNV40	I&C separation group A EPA	Harsh	Electrical Mechanical	Yes	Yes	A
CNV41	I&C separation group B EPA	Harsh	Electrical Mechanical	Yes	Yes	A
CNV42	I&C separation group C EPA	Harsh	Electrical Mechanical	Yes	Yes	A
CNV43	I&C separation group D EPA	Harsh	Electrical Mechanical	Yes	Yes	A
CNV44	CRDM power 2 nozzle EPA	Harsh	Electrical Mechanical	Yes	No	B
CNV-CRD-1000	CNV CRDM Support Frame (Upper)	N/A	N/A	Yes	N/A	N/A
CNV-CRD-1001	CRDM Support Structure (Lower)	N/A	N/A	Yes	N/A	N/A
None	CNV-RPV Support Ledge	N/A	N/A	Yes	N/A	N/A
MS-HOV-0101	MSIV #1	Harsh	Mechanical	Yes	Yes	A B

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
MS-HOV-0201	MSIV #2	Harsh	Mechanical	Yes	Yes	A B
MS-HOV-0103	MSIBV #1	Harsh	Mechanical	Yes	Yes	A B
MS-HOV-0203	MSIBV #2	Harsh	Mechanical	Yes	Yes	A B
FW-HOV-0137	FWIV #1	Harsh	Mechanical	Yes	Yes	A B
FW-HOV-0237	FWIV #2	Harsh	Mechanical	Yes	Yes	A B
FW-CKV-0136	Feedwater isolation check valve #1	Harsh	Mechanical	Yes	N/A	A B
FW-CKV-0236	Feedwater isolation check valve #2	Harsh	Mechanical	Yes	N/A	A B
FW-HV-0901	Feedwater #1 CITF valve	Harsh	Mechanical	Yes	N/A	B
FW-HV-0902	Feedwater #2 CITF valve	Harsh	Mechanical	Yes	N/A	B
RCCW-HV-0905	RCCW return CITF valve	Harsh	Mechanical	Yes	N/A	B
CVC-HV-0906	CVC injection CITF valve	Harsh	Mechanical	Yes	N/A	B
CVC-HV-0907	PZR spray CITF valve	Harsh	Mechanical	Yes	N/A	B
CE-HV-0910	Containment evacuation CITF valve	Harsh	Mechanical	Yes	N/A	B
CFD-HV-0911	Containment flood and drain CITF valve	Harsh	Mechanical	Yes	N/A	B
RCCW-HV-0912	RCCW supply CITF valve	Harsh	Mechanical	Yes	N/A	B
CVC-HV-0913	CVC discharge CITF valve	Harsh	Mechanical	Yes	N/A	B
CVC-HV-0914	RCS high point degas CITF valve	Harsh	Mechanical	Yes	N/A	B
CVC-HOV-0334	CVCS discharge inboard CIV	Harsh	Mechanical	Yes	Yes	A B
CVC-HOV-0335	CVCS discharge outboard CIV	Harsh	Mechanical	Yes	Yes	A B
CVC-HOV-0331	CVCS injection inboard CIV	Harsh	Mechanical	Yes	Yes	A B
CVC-HOV-0330	CVCS injection outboard CIV	Harsh	Mechanical	Yes	Yes	A B
CVC-HOV-0325	PZR spray inboard CIV	Harsh	Mechanical	Yes	Yes	A B

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
CVC-HOV-0324	PZR spray outboard CIV	Harsh	Mechanical	Yes	Yes	A B
CVC-HOV-0401	RPV high point degas inboard CIV	Harsh	Mechanical	Yes	Yes	A B
CVC-HOV-0402	RPV high point degas outboard CIV	Harsh	Mechanical	Yes	Yes	A B
RCCW-HOV-0185	RCCWS supply inboard CIV	Harsh	Mechanical	Yes	Yes	A B
RCCW-HOV-0184	RCCWS supply outboard CIV	Harsh	Mechanical	Yes	Yes	A B
RCCW-HOV-0190	RCCWS return inboard CIV	Harsh	Mechanical	Yes	Yes	A B
RCCW-HOV-0191	RCCWS return outboard CIV	Harsh	Mechanical	Yes	Yes	A B
CE-HOV-0001	CES inboard CIV	Harsh	Mechanical	Yes	Yes	A B
CE-HOV-0002	CES outboard CIV	Harsh	Mechanical	Yes	Yes	A B
CFD-HOV-0022	CFDS inboard CIV	Harsh	Mechanical	Yes	Yes	A B
CFD-HOV-0021	CFDS outboard CIV	Harsh	Mechanical	Yes	Yes	A B
CNT-SKD-0500 CNT-SKD-0600	Central hydraulic power unit skids	Harsh	Electrical Mechanical	Yes	Yes	A B
CNT-PE-1001A CNT-PE-1001B CNT-PE-1001C CNT-PE-1001D	Containment narrow range pressure elements	Harsh	Electrical	Yes	Yes	A
CNT-PT-1001A CNT-PT-1001B CNT-PT-1001C CNT-PT-1001D	Containment narrow range pressure transmitters	Mild	Electrical	Yes	Yes	E
CNT-PE-1002A CNT-PE-1002B	Containment wide range pressure elements	Harsh	Electrical	Yes	No	A
CNT-PT-1002A CNT-PT-1002B	Containment wide range pressure transmitters	Mild	Electrical	Yes	Yes	E

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
CNT-LE-1003A CNT-LE-1003B CNT-LE-1003C CNT-LE-1003D	Containment level indication	Harsh	Electrical	Yes	Yes	A
CNT-LT-1003A CNT-LT-1003B CNT-LT-1003C CNT-LT-1003D	Containment level transmitters	Mild	Electrical	Yes	Yes	E
CNT-PAR-0001	Passive autocatalytic recombiner	Harsh	Mechanical	Yes	N/A	A
MS-TE-1001A MS-TE-1001B MS-TE-1001C MS-TE-1001D	SG #1 main steam temperature indication	Harsh	Electrical	Yes	Yes	A
MS-TE-2001A MS-TE-2001B MS-TE-2001C MS-TE-2001D	SG #2 main steam temperature indication	Harsh	Electrical	Yes	Yes	A
CE-ZSC-0001	CES inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CE-ZSO-0001	CES inboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CE-PT-0001	CES inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CE-ZSC-0002	CES outboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CE-ZSO-0002	CES outboard CIV open Position indicator	Harsh	Electrical	Yes	No	A
CE-PT-0002	CES outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CFD-ZSC-0022	CFDS inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CFD-ZSO-0022	CFDS inboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CFD-PT-0022	CFDS inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CFD-ZSC-0021	CFDS outboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CFD-ZSO-0021	CFDS outboard CIV open Position indicator	Harsh	Electrical	Yes	No	A
CFD-PT-0021	CFDS outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0334	CVCS discharge inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0334	CVCS discharge inboard CIV open position indicator	Harsh	Electrical	Yes	No	A

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[(1)]
CVC-PT-0334	CVCS discharge inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0335	CVCS discharge outboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0335	CVCS discharge outboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CVC-PT-0335	CVCS discharge outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0331	CVCS injection inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0331	CVCS injection inboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CVC-PT-0331	CVCS injection inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0330	CVCS injection outboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0330	CVCS injection outboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CVC-PT-0330	CVCS injection outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0325	PZR spray inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0325	PZR spray inboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CVC-PT-0325	PZR spray inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0324	PZR spray outboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0324	PZR spray outboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CVC-PT-0324	PZR spray outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0401	RPV high point degas inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0401	RPV high point degas inboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CVC-PT-0401	CVC high point degas inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
CVC-ZSC-0402	RPV high point degas outboard CIV close position indicator	Harsh	Electrical	Yes	No	A
CVC-ZSO-0402	RPV high point degas outboard CIV open position indicator	Harsh	Electrical	Yes	No	A
CVC-PT-0402	CVCS high point degas outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
RCCW-ZSC-0185	RCCWS supply inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
RCCW-ZSO-0185	RCCWS supply inboard CIV open position indicator	Harsh	Electrical	Yes	No	A
RCCW-PT-0185	RCCWS supply inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
RCCW-ZSC-0184	RCCWS supply outboard CIV close position indicator	Harsh	Electrical	Yes	No	A

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
RCCW-ZSO-0184	RCCWS supply outboard CIV open position indicator	Harsh	Electrical	Yes	No	A
RCCW-PT-0184	RCCWS supply outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
RCCW-ZSC-0190	RCCWS return inboard CIV close position indicator	Harsh	Electrical	Yes	No	A
RCCW-ZSO-0190	RCCWS return inboard CIV open position indicator	Harsh	Electrical	Yes	No	A
RCCW-PT-0190	RCCWS return inboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
RCCW-ZSC-0191	RCCWS return outboard CIV close position indicator	Harsh	Electrical	Yes	No	A
RCCW-ZSO-0191	RCCWS return outboard CIV open position indicator	Harsh	Electrical	Yes	No	A
RCCW-PT-0191	RCCWS return outboard CIV nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
FW-ZSO-0137A FW-ZSO-0137B	FWIV #1 opened position indication	Harsh	Electrical	Yes	No	A
FW-ZSC-0137A FW-ZSC-0137B	FWIV #1 closed position indication	Harsh	Electrical	Yes	No	A
FW-PT-0137	FWIV #1 nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
FW-ZSC-0237A FW-ZSC-0237B	FWIV #2 closed position indication	Harsh	Electrical	Yes	No	A
FW-ZSO-0237A FW-ZSO-0237B	FWIV #2 opened position indication	Harsh	Electrical	Yes	No	A
FW-PT-0237	FWIV #2 nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
MS-ZSC-0101A MS-ZSC-0101B	MSIV #1 closed position indication	Harsh	Electrical	Yes	No	A
MS-ZSO-0101A MS-ZSO-0101B	MSIV #1 open position indication	Harsh	Electrical	Yes	No	A
MS-PT-0101	MSIV #1 nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
MS-ZSC-0103A MS-ZSC-0103B	MSIBV #1 closed position indication	Harsh	Electrical	Yes	No	A
MS-ZSO-0103A MS-ZSO-0103B	MSIBV #1 open position indication	Harsh	Electrical	Yes	No	A
MS-PT-0103	MSIBV #1 nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
MS-ZSC-0201A MS-ZSC-0201B	MSIV #2 closed position indications	Harsh	Electrical	Yes	No	A
MS-ZSO-0201A MS-ZSO-0201B	MSIV #2 open position indications	Harsh	Electrical	Yes	No	A

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
MS-PT-0201	MSIV #2 nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
MS-ZSC-0203A MS-ZSC-0203B	MSIBV #2 closed position indication	Harsh	Electrical	Yes	No	A
MS-ZSO-0203A MS-ZSO-0203B	MSIBV #2 open position indication	Harsh	Electrical	Yes	No	A
MS-PT-0203	MSIBV #2 nitrogen accumulator pressure transmitter	Harsh	Mechanical	Yes	No	B
Steam Generator System						
None	SG tubes and tube supports	N/A	N/A	Yes	N/A	N/A
RPV7 RPV8 RPV9 RPV10	Steam plenums	N/A	N/A	Yes	N/A	N/A
RPV3 RPV4 RPV5 RPV6	Feedwater plenums	N/A	N/A	Yes	N/A	N/A
None	Flow restrictors	N/A	N/A	Yes	N/A	N/A
SG-RV-0102 SG-RV-0202	Thermal relief valves	Harsh	Mechanical	Yes	N/A	B
Control Rod Drive System						
CRDS-DS-00xx (xx = 01:16)	Control rod drive shaft assemblies (16 total)	Harsh	Mechanical	Yes	N/A	B
CRDS-MJ-00xx (xx = 01:16)	CRDM magnetic jack assemblies (16 total)	Harsh	Mechanical	Yes	N/A	B
CRDS-PH-00xx (xx = 01:16)	CRDM pressure housing (16 total)	Harsh	Mechanical	Yes	N/A	B
CRDS-PBS-20xx (xx = 01:16)	CRDM top plug assemblies (16 total)	Harsh	Mechanical	Yes	N/A	B
CRDS-SS-00xx (xx = 01:16)	CRDM seismic supports (16 total)	N/A	N/A	Yes	N/A	N/A
CRDS-PBS-10xx (xx = 01:16)	CRDM-to-RPV joint sealing components (16 total)	Harsh	Mechanical	Yes	N/A	B
Control Rod Assembly						
None	All components	N/A	N/A	Yes	N/A	N/A

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
Neutron Source Assembly						
None	Primary and secondary neutron source rodlets	N/A	N/A	Yes	N/A	N/A
None	Spider body, hub or coupling housing	N/A	N/A	Yes	N/A	N/A
Reactor Coolant System						
None	RVI	N/A	N/A	Yes	N/A	N/A
RCS-ZSO-0003A RCS-ZSO-0003B	Reactor safety valve open position indicators	Harsh	Electrical	Yes	No	A
RCS-ZSC-0003A RCS-ZSC-0003B	Reactor safety valve close position indicators	Harsh	Electrical	Yes	No	A
RCS-PSV-0003A RCS-PSV-0003B	Reactor safety valves	Harsh	Mechanical	Yes	N/A	A
RCS-PE-1013A RCS-PE-1013B RCS-PE-1013C RCS-PE-1013D	Narrow range PZR pressure elements	Harsh	Electrical Mechanical	Yes	Yes	A B
RCS-PT-1013A RCS-PT-1013B RCS-PT-1013C RCS-PT-1013D	Narrow range PZR pressure transmitters	Mild	Electrical	Yes	Yes	E
RCS-PE-1014A RCS-PE-1014B RCS-PE-1014C RCS-PE-1014D	Wide range RCS pressure elements	Harsh	Electrical Mechanical	Yes	Yes	A
RCS-PT-1014A RCS-PT-1014B RCS-PT-1014C RCS-PT-1014D	Wide range RCS pressure transmitters	Mild	Electrical	Yes	Yes	E
RCS-LE-1008A RCS-LE-1008B RCS-LE-1008C RCS-LE-1008D	PZR level elements	Harsh	Electrical Mechanical	Yes	Yes	A B
RCS-LT-1008A RCS-LT-1008B RCS-LT-1008C RCS-LT-1008D	PZR level transmitters	Mild	Electrical	Yes	Yes	E

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
RCS-LE-1015A RCS-LE-1015B RCS-LE-1015C RCS-LE-1015D	RPV riser level elements	Harsh	Electrical Mechanical	Yes	Yes	A
RCS-LT-1015A RCS-LT-1015B RCS-LT-1015C RCS-LT-1015D	RPV riser level transmitters	Mild	Electrical	Yes	Yes	E
RCS-TE-1005A1 RCS-TE-1005B1 RCS-TE-1005C1 RCS-TE-1005D1 RCS-TE-1005A2 RCS-TE-1005B2 RCS-TE-1005C2 RCS-TE-1005D2	RCS hot temperature elements	Harsh	Electrical	Yes	Yes	A
RCS-TE-1009A1 RCS-TE-1009B1 RCS-TE-1009C1 RCS-TE-1009D1 RCS-TE-1009A2 RCS-TE-1009B2 RCS-TE-1009C2 RCS-TE-1009D2	RCS cold temperature elements	Harsh	Electrical	Yes	Yes	B
RCS-FE-1012A RCS-FE-1012B RCS-FE-1012C RCS-FE-1012D	RCS flow elements	Harsh	Electrical	Yes	Yes	A
RCS-FT-1012A RCS-FT-1012B RCS-FT-1012C RCS-FT-1012D	RCS flow transmitters	Mild	Electrical	Yes	Yes	E
RCS-HT-0002A RCS-HT-0002B	PZR heaters	Harsh	Mechanical	Yes	No	B
RCS-TE-1017A RCS-TE-1017B	PZR vapor temperature elements	N/A	N/A	Yes	No	N/A

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
RCS-TE-1016A RCS-TE-1016B	PZR liquid temperature elements	N/A	N/A	Yes	No	N/A
RCS-ISA-039 RCS-ISA-040 RCS-ISA-041 RCS-ISA-042 RCS-ISA-085 RCS-ISA-086	RPV instrument seal assemblies	Harsh	Mechanical	Yes	Yes	B
Chemical and Volume Control System						
CVC-AOV-0089 CVC-AOV-0090	DWS supply isolation valves	Harsh	Mechanical	Yes	Yes	A B
Emergency Core Cooling						
ECC-POV-0001A ECC-POV-0001B	RVVs	Harsh	Mechanical	Yes	No	A B
ECC-FV-0001A ECC-FV-0001B	RVV valve venturis	N/A	N/A	Yes	N/A	N/A
ECC-ZSO-0001A ECC-ZSO-0001B	RVV open position indications	Harsh	Electrical	Yes	No	A
ECC-ZSC-0001A ECC-ZSC-0001B	RVV close position indications	Harsh	Electrical	Yes	No	A
ECC-POV-0002A ECC-POV-0002B	RRVs	Harsh	Mechanical	Yes	No	A B
ECC-FV-0002A ECC-FV-0002B	RRV valve venturis	N/A	N/A	Yes	N/A	N/A
ECC-ZSO-0002A ECC-ZSO-0002B	RRV open position indications	Harsh	Electrical	Yes	No	A
ECC-ZSC-0002A ECC-ZSC-0002B	RRV close position indications	Harsh	Electrical	Yes	No	A
ECC-SV-0101A ECC-SV-0101B ECC-SV-0102A ECC-SV-0102B	RVV trip valves	Harsh	Electrical Mechanical	Yes	Yes	A B
ECC-ZSO-0101A ECC-ZSO-0101B ECC-ZSO-0102A ECC-ZSO-0102B	RVV trip valve open position indications	Harsh	Electrical	Yes	No	A

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
ECC-ZSC-0101A ECC-ZSC-0101B ECC-ZSC-0102A ECC-ZSC-0102B	RVV trip valve close position indications	Harsh	Electrical	Yes	No	A
ECC-SV-0104A ECC-SV-0104B ECC-SV-0105A ECC-SV-0105B	RRV trip valves	Harsh	Electrical Mechanical	Yes	Yes	A B
ECC-ZSO-0104A ECC-ZSO-0104B ECC-ZSO-0105A ECC-ZSO-0105B	RRV trip valve open position indications	Harsh	Electrical	Yes	No	A
ECC-ZSC-0104A ECC-ZSC-0104B ECC-ZSC-0105A ECC-ZSC-0105B	RRV trip valve close position indications	Harsh	Electrical	Yes	No	A
ECC-SV-0103A ECC-SV-0103B	RVV reset valves	Harsh	Mechanical	Yes	No	B
ECC-SV-0106A ECC-SV-0106B	RRV reset valves	Harsh	Mechanical	Yes	No	B
ECC-DIS-0001 ECC-DIS-0002	Supplemental boron dissolvers	N/A	N/A	Yes	N/A	N/A
ECC-MTB-0001 ECC-MTB-0002	CNV lower mixing tube	N/A	N/A	Yes	N/A	N/A
Decay Heat Removal System						
DHR-HOV-0111 DHR-HOV-0121 DHR-HOV-0211 DHR-HOV-0221	DHRS actuation valves	Harsh	Mechanical	Yes	Yes	A B
DHR-PT-0111 DHR-PT-0121 DHR-PT-0211 DHR-PT-0221	DHRS actuation valve nitrogen accumulator pressure transmitters	Harsh	Mechanical	Yes	No	B

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
DHR-TE-1003A DHR-TE-1003B DHR-TE-2003A DHR-TE-2003A	DHRS condenser temperature elements	N/A	N/A	Yes	No	N/A
DHR-PT-1004A DHR-PT-1004B DHR-PT-1004C DHR-PT-2004A DHR-PT-2004B DHR-PT-2004C	DHRS condenser outlet pressure transmitters	N/A	N/A	Yes	No	N/A
DHR-PE-1004A DHR-PE-1004B DHR-PE-1004C DHR-PE-2004A DHR-PE-2004B DHR-PE-2004C	DHRS condenser outlet pressure elements	Harsh	Mechanical	Yes	No	B
DHR-ZSO-0111 DHR-ZSO-0121 DHR-ZSO-0211 DHR-ZSO-0221	DHRS valve position open indicators	Harsh	Electrical	Yes	No	A
DHR-ZSC-0111 DHR-ZSC-0121 DHR-ZSC-0211 DHR-ZSC-0221	DHRS valve position close indicators	Harsh	Electrical	Yes	No	A
DHR-CND-0103 DHR-CND-0203	Condensers	N/A	N/A	Yes	N/A	N/A
DHR-FO-0102 DHR-FO-0202	DHRS flow orifice	N/A	N/A	Yes	N/A	N/A
DHR-PT-1001A DHR-PT-1001B DHR-PT-1001C DHR-PT-1001D DHR-PT-2001A DHR-PT-2001B DHR-PT-2001C DHR-PT-2001D	SG main steam pressure transmitters	Harsh	Electrical	Yes	Yes	A

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[(1)]
DHR-PE-1001A DHR-PE-1001B DHR-PE-1001C DHR-PE-1001D DHR-PE-2001A DHR-PE-2001B DHR-PE-2001C DHR-PE-2001D	SG main steam pressure elements	Harsh	Electrical Mechanical	Yes	Yes	A B
DHR-LS-1002A DHR-LS-1002B DHR-LS-1002C DHR-LS-1002D DHR-LS-2002A DHR-LS-2002B DHR-LS-2002C DHR-LS-2002D	DHRS level switches	Harsh	Mechanical	Yes	No	B
Main Steam System						
MS-V-0102	SG1 secondary MSIV	Harsh	Electrical Mechanical	Yes	No	A B
MS-V-0202	SG2 secondary MSIV	Harsh	Electrical Mechanical	Yes	No	A B
MS-V-0104	SG1 secondary MSIV bypass valve	Harsh	Electrical Mechanical	Yes	No	A B
MS-V-0204	SG2 secondary MSIV bypass valve	Harsh	Electrical Mechanical	Yes	No	A B
Condensate and Feedwater System						
FW-AOV-0134	SG1 feedwater regulating valve	Harsh	Mechanical	Yes	No	A B
FW-AOV-0234	SG2 feedwater regulating valve	Harsh	Mechanical	Yes	No	A B
FW-CKV-0135	SG1 containment feedwater check valve	Harsh	Mechanical	Yes	N/A	B
FW-CKV-0235	SG2 containment feedwater check valve	Harsh	Mechanical	Yes	N/A	B

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[1]
Module Protection System						
None	Safety-related MPS modules Safety function modules Hard-wired modules Scheduling and bypass modules Equipment interface modules Scheduling and voting modules	Mild	Electrical	Yes	Yes	E
None	Power isolation, conversion and monitoring Devices	Mild	Electrical	Yes	Yes	E
MPS-EE-AN-0001 MPS-EE-AN-0002 MPS-EE-BN-0001 MPS-EE-BN-0002 MPS-EE-CN-0001 MPS-EE-CN-0002 MPS-EE-DN-0001 MPS-EE-DN-0002	Low voltage AC electrical distribution system voltage sensors	Mild	Electrical	Yes	Yes	E
MPS-TE-AE-0001 MPS-TE-BE-0001 MPS-TE-CE-0001 MPS-TE-DE-0001	Under-the-bioshield temperature sensors	Harsh	Electrical	Yes	Yes	A
MPS-BKR-1E-0002A MPS-BKR-1E-0002B MPS-BKR-2E-0002A MPS-BKR-2E-0002B	PZR heater trip breakers	Mild	Electrical	Yes	Yes	E
MPS-BKR-2E-0001A MPS-BKR-2E-0001B	Reactor trip breakers (RTBs)	Mild	Electrical	Yes	Yes	E
MPS-HS-AE-TB01 through TB15 MPS-HS-BE-TB01 through TB15 MPS-HS-CE-TB01 through TB15 MPS-HS-DE-TB01 through TB15	Safety function module trip/bypass switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-0001 MPS-HS-2E-0001	Enable nonsafety control switches	Mild	Electrical	Yes	Yes	E

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category[1]
MPS-HS-1E-0004 MPS-HS-2E-0004	Main control room (MCR) isolation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA08 MPS-HS-2E-MA08	Manual PZR heater breaker trip switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA07 MPS-HS-2E-MA07	Manual low temperature overpressure protection actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA06 MPS-HS-2E-MA06	Manual ECCS actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA05 MPS-HS-2E-MA05	Manual DWS isolation actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA04 MPS-HS-2E-MA04	Manual DHRS actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA03 MPS-HS-2E-MA03	Manual CVCS isolation actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA02 MPS-HS-2E-MA02	Manual containment system isolation actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA01 MPS-HS-2E-MA01	Manual reactor trip switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA09 MPS-HS-2E-MA09	Manual secondary system isolation actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-MA10 MPS-HS-2E-MA10	Manual ECCS timer block actuation switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-0002 MPS-HS-2E-0002	Override switches	Mild	Electrical	Yes	Yes	E
MPS-HS-1E-0003 MPS-HS-2E-0003	Operating bypass switches	Mild	Electrical	Yes	Yes	E
Neutron Monitoring System						
NMS-NE-AE-0001 NMS-NE-BE-0001 NMS-NE-CE-0001 NMS-NE-DE-0001	Neutron monitoring system (NMS)-excore neutron detector and moderator assemblies	Harsh	Electrical	Yes	Yes	A
NMS-ASY-0001A NMS-ASY-0001B NMS-ASY-0001C NMS-ASY-0001D	NMS-excore operating bay positioning support mechanisms	Harsh	Mechanical	Yes	N/A	B

Table 2.4-3: Module-Specific Mechanical and Electrical/Instrumentation and Controls Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category I	Class 1E	EQ Category(1)
None	Excore signal conditioning and processing equipment	Mild	Electrical	Yes	Yes	E
None	Excore power isolation, conversion and monitoring devices	Mild	Electrical	Yes	Yes	E
In-Core Instrumentation System						
ICI-ASY-0001-B ICI-ASY-0002-B ICI-ASY-0003-C ICI-ASY-0004-C ICI-ASY-0005-B ICI-ASY-0006-B ICI-ASY-0007-C ICI-ASY-0008-C ICI-ASY-0009-C ICI-ASY-0010-B ICI-ASY-0011-C ICI-ASY-0012-B	In-core instrument stringer assemblies	Harsh	Electrical Mechanical	Yes	No	A
ICI-TE-0001F-B ICI-TE-0002F-B ICI-TE-0003F-C ICI-TE-0004F-C ICI-TE-0005F-B ICI-TE-0006F-B ICI-TE-0007F-C ICI-TE-0008F-C ICI-TE-0009F-C ICI-TE-0010F-B ICI-TE-0011F-C ICI-TE-0012F-B	Core exit thermocouples	Harsh	Electrical	Yes	No	A

Note:

1. EQ Categories:
   - A - Equipment that will experience the environmental conditions of DBAs for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.
   - B - Equipment that will experience the environmental conditions of DBAs through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand the accident environment for the time during which it must not fail with safety margin to failure.
   - E - Equipment that will not experience environmental conditions of DBAs and that will be qualified to demonstrate operability under the expected extremes of its nonaccident service environment.

Table 2.4-4: Equipment Qualification - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
02.04.01	X					
02.04.02	X					
02.04.03	X					
02.04.04	X					
02.04.05	X					
02.04.06	X					
02.04.07	X					
02.04.08	X					
02.04.09	X					
02.04.10	X					
02.04.11	X					

2.5 Module Protection System and Safety Display and Indication System

2.5.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the MPS and its associated components in the safety display and indication system (SDIS), which are described in FSAR Section 7.0.4. Each NPM has its own independent MPS and SDIS.

The MPS performs the following safety-related system functions that are verified by ITAAC:

- The MPS supports the CNTS by removing electrical power to the trip solenoids of the following CIVs on a CNTS isolation actuation signal:
  - RCS injection CIVs
  - RCS discharge CIVs
  - PZR spray CIVs
  - RPV high point degasification CIVs
  - feedwater CIVs
  - main steam CIVs
  - main steam bypass valves
  - containment evacuation system CIVs
  - reactor component cooling water CIVs
  - containment flooding and drain system CIVs
- The MPS supports the CNTS by removing electrical power to the trip solenoids of the following valves on a DHRS actuation signal:
  - DHRS actuation valves
  - main steam CIVs
  - main steam bypass valves
  - feedwater CIVs
- The MPS supports the CNTS by removing electrical power to the trip solenoids of the following valves on an SSI actuation signal:
  - main steam CIVs
  - main steam bypass valves
  - feedwater CIVs
- The MPS supports the ECCS by removing electrical power to the trip solenoids of the RVVs and RRVs on an ECCS actuation signal.

- The MPS supports the CNTS by removing electrical power to the trip solenoids of the following CIVs on a chemical and volume control isolation actuation signal:
  - RCS injection CIVs
  - RCS discharge CIVs
  - PZR spray CIVs
  - RPV high point degasification CIVs
- The MPS supports the CVCS by removing electrical power to the trip solenoids of the DWS supply isolation valves on a DWS isolation actuation signal.
- The MPS supports the ECCS by removing electrical power to the trip solenoids of the RVVs on a low temperature overpressure protection actuation signal.
- The MPS supports the low voltage AC electrical distribution system by removing electrical power to the PZR heaters on a PZR heater trip actuation signal.
- The MPS supports the normal DC power system by removing electrical power to the CRDS for a reactor trip.
- The MPS supports the following systems by providing power to sensors for reactor trip and engineered safety features actuation system (ESFAS) actuation:
  - CNTS
  - RCS
  - DHRS (main steam system (MSS) pressure sensors)
- The MPS performs the following nonsafety-related system function that is verified by ITAAC. The MPS supports the CNTS and the RCS by providing power to sensors for post-accident monitoring (PAM) Type B and Type C variables.
- The SDIS performs the following nonsafety-related system function that is verified by ITAAC:
  - The SDIS supports the MCR by providing displays of PAM Type B and Type C variables.

Design Commitments

- The MPS design and software are implemented using a quality process comprised of the following system design life-cycle phases, with each phase having outputs that satisfy the requirements of that phase:
  - system concept phase
  - system requirements phase
  - system design phase
  - system implementation phase
  - system test phase
  - system installation and checkout phase
- Protective measures are provided to restrict modifications to the MPS tunable parameters.

- Communications independence exists between Separation Groups A, B, C, and D of the Class 1E MPS.
- Communications independence exists between Divisions I and II of the Class 1E MPS.
- The MPS automatically initiates a reactor trip signal for reactor trip functions listed in FSAR Table 7.1-3.
- The MPS automatically initiates an engineered safety features (ESF) actuation signal for ESF functions listed in FSAR Table 7.1-4.
- The MPS automatically actuates a reactor trip.
- The MPS manually actuates a reactor trip.
- The reactor trip logic fails to a safe state such that loss of electrical power to a MPS separation group results in a trip state for that separation group.
- The ESFs logic fails to a safe state such that loss of electrical power to a MPS separation group results in a safe state listed in Table 2.1-5.
- The MPS interlocks listed in FSAR Table 7.1-5 automatically establish an operating bypass for the specified reactor trip or ESF actuations when the interlock condition is met, and the operating bypass is automatically removed when the interlock condition is no longer satisfied.
- The MPS permissives listed in FSAR Table 7.1-5 allow the manual bypass of the specified reactor trip or ESF actuations when the permissive condition is met, and the operating bypass is automatically removed when the permissive condition is no longer satisfied.
- The O-1 Override listed in FSAR Table 7.1-5 is established when the manual Override switch is active and the RT-1 interlock is established. The Override switch must be manually taken out of Override when the O-1 Override is no longer needed.
- The MPS is capable of performing its safety-related functions when any one of its separation groups is out of service.
- The RTBs are installed and arranged as shown in FSAR Figure 7.0-6 in order to successfully accomplish the reactor trip function.
- Two of the four separation groups and one of the two divisions of reactor trip system (RTS) and ESFAS will utilize a different programmable technology.
- Physical separation exists (i) between each separation group of the MPS Class 1E instrumentation and control current-carrying circuits, (ii) between each division of the MPS Class 1E instrumentation and control current-carrying circuits, and (iii) between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.
- Electrical isolation exists (i) between each separation group of the MPS Class 1E instrumentation and control circuits, (ii) between each division of the MPS Class 1E instrumentation and control circuits, and (iii) between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.

- Electrical isolation exists between the augmented DC power system (EDAS) module-specific subsystem non-Class 1E circuits and connected MPS 1E circuits to prevent the propagation of credible electrical faults.
- Communications independence exists between the Class 1E MPS and non-Class 1E digital systems.
- The MPS automatically actuates the ESF equipment to perform its safety-related function listed in FSAR Table 7.1-4.
- The MPS manually actuates the ESF equipment to perform its safety-related function listed in FSAR Table 7.1-4.
- An MPS signal, once initiated (automatically or manually), results in an intended sequence of protective actions that continue until completion, and requires deliberate operator action in order to return the safety systems to normal.
- The MPS response times from sensor output through equipment actuation for the reactor trip functions and ESF functions are less than or equal to the value required to satisfy the design basis safety analysis response time assumptions.
- The MPS operational bypasses are indicated in the MCR.
- The MPS maintenance bypasses are indicated in the MCR.
- The MPS self-test features detect faults in the system and provide an alarm in the MCR.
- The PAM Type B and Type C displays are indicated on the SDIS displays in the MCR.

2.5.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5-1 contains the ITAAC for the MPS and SDIS.

Table 2.5-1: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.05.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01	The-design and software are implemented using a quality process comprised of the following system design life-cycle phases, with each phase having outputs that satisfy the requirements of that phase.		
	1.a System Concept Phase	1.a An analysis will be performed of the output documentation of the System Concept Phase.	1.a The output documentation of the Concept Phase satisfies the requirements of the System Concept Phase.
	1.b System Requirements Phase	1.b An analysis will be performed of the output documentation of the System Requirements Phase.	1.b The output documentation of the Requirements Phase satisfies the requirements of the System Requirements Phase.
	1.c System Design Phase	1.c An analysis will be performed of the output documentation of the System Design Phase.	1.c The output documentation of the Design Phase satisfies the requirements of the System Design Phase.
	1.d System Implementation Phase	1.d An analysis will be performed of the output documentation of the System Implementation Phase.	1.d The output documentation of the Implementation Phase satisfies the requirements of the System Implementation Phase.
	1.e System Test Phase	1.e An analysis will be performed of the output documentation of the System Test Phase.	1.e The output documentation of the MPS Test Phase satisfies the requirements of the System Test Phase.
	1.f System Installation and Checkout Phase	1.f An analysis will be performed of the output documentation of the System Installation and Checkout Phase.	1.f The output documentation of the MPS Installation and Checkout Phase satisfies the requirements of the System Installation and Checkout Phase.
	ii. Protective measures are provided to restrict modifications to the MPS tunable parameters.	ii. Test will be performed on the access control features associated with MPS tunable parameters.	ii. Protective measures restrict modification to the MPS tunable parameters without proper configuration and authorization.
	iii.a Communications independence exists between Separation Groups A, B, C, and D of the Class 1E MPS.	iii.a A test will be performed of the Class 1E MPS.	iii.a Communications independence between Separation Groups A, B, C, and D of the Class 1E MPS is provided.
	iii.b Communications independence exists between Divisions I and II of the Class 1E MPS.	iii.b A test will be performed of the MPS.	iii.b Communications independence between Division I and II of the Class 1E MPS is provided.
	iv. The MPS automatically initiates a reactor trip signal for reactor trip functions listed in FSAR Table 7.1-3.	iv. A test will be performed of the MPS.	iv. Reactor trip signal is automatically initiated for each reactor trip function listed in FSAR Table 7.1-3.
	v. The MPS automatically initiates an ESF actuation signal for ESF functions listed in FSAR Table 7.1-4.	v. A test will be performed of the MPS.	v. An ESF actuation signal is automatically initiated for each ESF function listed in FSAR Table 7.1-4.

Table 2.5-1: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.05.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	vi. The MPS automatically actuates a reactor trip.	vi. A test will be performed of the MPS.	vi. The RTBs open upon an injection of a single simulated MPS reactor trip signal.
	vii. The MPS manually actuates a reactor trip.	vii. A test will be performed of the MPS.	vii. The RTBs open when a reactor trip is manually initiated from the MCR.
	viii. The reactor trip logic fails to a safe state such that loss of electrical power to a MPS separation group results in a trip state for that separation group.	viii. A test will be performed of the MPS.	viii. Loss of electrical power in a separation group results in a trip state for that separation group.
	ix. The ESFs logic fails to a safe state such that loss of electrical power to a MPS separation group results in a safe state listed in Table 2.1-5.	ix. A test will be performed of the MPS.	ix. Loss of electrical power in a separation group results in the safe state listed in Table 2.1-5.
	x. The MPS interlocks listed in FSAR Table 7.1-5 automatically establish an operating bypass for the specified reactor trip or ESF actuations when the interlock condition is met, and the operating bypass is automatically removed when the interlock condition is no longer satisfied.	x. A test will be performed of the MPS.	x. The MPS interlocks listed in FSAR Table 7.1-5 automatically establish an operating bypass for the specified reactor trip or ESF actuations when the interlock condition is met. The operating bypass is automatically removed when the interlock condition is no longer satisfied.
	xi. The MPS permissives listed in FSAR Table 7.1-5 allow the manual bypass of the specified reactor trip or ESF actuations when the permissive condition is met, and the operating bypass is automatically removed when the permissive condition is no longer satisfied.	xi. A test will be performed of the MPS.	xi. The MPS permissives listed in FSAR Table 7.1-5 allow the manual bypass of the specified reactor trip or ESF actuations when the permissive condition is met. The operating bypass is automatically removed when the permissive condition is no longer satisfied.
	xii. The O-1 Override listed in FSAR Table 7.1-5 is established when the manual override switch is active and the RT-1 interlock is established. The Override switch must be manually taken out of Override when the O-1 Override is no longer needed.	xii. A test will be performed of the MPS.	xii. The O-1 Override listed in FSAR Table 7.1-5 is established when the manual override switch is active and the RT-1 interlock is established. The Override switch must be manually taken out of Override when the O-1 Override is no longer needed.
	xiii. The MPS is capable of performing its safety-related functions when any one of its separation groups is out of service.	xiii. A test will be performed of the MPS.	xiii. The MPS performs its safety-related functions if any one of its separation groups is out of service.

Table 2.5-1: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.05.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
	xiv. The RTBs are installed and arranged as shown in FSAR Figure 7.0-6 in order to successfully accomplish the reactor trip function.	xiv. An inspection will be performed of the as-built RTBs, including the connections for the shunt and undervoltage trip mechanism and auxiliary contacts.	xiv. The RTBs have the proper connections for the shunt and undervoltage trip mechanisms and auxiliary contacts, and are arranged as shown in FSAR Figure 7.0-6 to successfully accomplish the reactor trip function.
	xv. Two of the four separation groups and one of the two divisions of RTS and ESFAS will utilize a different programmable technology.	xv. An inspection will be performed of the as-built MPS.	xv. Separation groups A & C and Division I of RTS and ESFAS utilize a different programmable technology from separation groups B & D and Division II of RTS and ESFAS.
02.	Physical separation exists (i) between each separation group of the MPS Class 1E instrumentation and control current-carrying circuits, (ii) between each division of the Class 1E instrumentation and control current-carrying circuits, and (iii) between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.	An inspection will be performed of the MPS Class 1E as-built instrumentation and control current-carrying circuits.	i. Physical separation between each separation group of the MPS Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers. ii. Physical separation between each division of the MPS Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers. iii. Physical separation between MPS Class 1E instrumentation and control current-carrying circuits and nonClass 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers.

Table 2.5-1: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.05.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
03.	Electrical isolation exists (i) between each separation group of the MPS Class 1E instrumentation and control circuits, (ii) between each division of the Class 1E instrumentation and control circuits, and (iii) between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.	An inspection will be performed of the MPS Class 1E as-built instrumentation and control circuits.	1. Class 1E electrical isolation devices are installed between each separation group of the MPS Class 1E instrumentation and control circuits. ii. Class 1E electrical isolation devices are installed between each division of the MPS Class 1E instrumentation and control circuits. iii. Class 1E electrical isolation devices are installed between MPS Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits.
04.	Electrical isolation exists between the EDAS module-specific subsystem non-Class 1E circuits and connected MPS Class 1E circuits to prevent the propagation of credible electrical faults.	i. A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E isolation devices. ii. An inspection will be performed of the MPS Class 1E as-built circuits.	i. The Class 1E circuit does not degrade below defined acceptable operating levels when the non-Class 1E side of the isolation device is subjected to the maximum credible voltage, current transients, shorts, grounds, or open circuits. ii. Class 1E electrical isolation devices are installed between the EDAS module specific Subsystem non-Class 1E circuits and connected MPS Class 1E circuits.
05.	Communications independence exists between the Class 1E MPS and non-Class 1E digital systems.	A test will be performed of the Class 1E MPS.	Communications independence between the Class 1E MPS and non-Class 1E digital systems is provided.
06.	The MPS automatically actuates the ESF equipment to perform its safety-related function listed in FSAR Table 7.1-4.	A test will be performed of the MPS.	The ESF equipment automatically actuates to perform its safety-related function listed in FSAR Table 7.1-4 upon an injection of a single simulate MPS signal.
07.	The MPS manually actuates the ESF equipment to perform its safety-related function listed in FSAR Table 7.1-4.	A test will be performed of the MPS.	The MPS actuates the ESF equipment to perform its safety-related function listed in FSAR Table 7.1-4 when manually initiated.

Table 2.5-1: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.05.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
08.	An MPS signal once initiated (automatically or manually), results in an intended sequence of protective actions that continue until completion, and requires deliberate operator action in order to return the safety systems to normal.	A test will be performed of the MPS reactor trip and ESF signals.	i. Upon initiation of a real or simulated MPS reactor trip signal listed in FSAR Table 7.1-3, the RTBs open, and the RTBs do not automatically close when the MPS reactor trip signal clears. ii. Upon initiation of a real or simulated MPS engineered safety feature actuation signal listed in FSAR Table 7.1-4, the ESF equipment actuates to perform its safety-related function and continues to maintain its safety-related position and perform its safety-related function when the MPS engineered safety feature actuation signal clears.
09.	The MPS response times from sensor output through equipment actuation for the reactor trip functions and ESF functions are less than or equal to the value required to satisfy the design basis safety analysis response time assumptions.	A test will be performed of the MPS.	The MPS reactor trip functions listed in FSAR Table 7.1-3 and ESF functions listed in FSAR Table 7.1-4 have response times that are less than or equal to the design basis safety analysis response time assumptions.
10.	MPS operational bypasses are indicated in the MCR.	A test will be performed of the MPS.	Each operational MPS manual or automatic bypass is indicated in the MCR.
11.	MPS maintenance bypasses are indicated in the MCR.	A test will be performed of the MPS.	Each maintenance bypass is indicated in the MCR.
12.	The MPS self-test features detect faults in the system and provide an alarm in the MCR.	A test will be performed of the MPS.	A report exists and concludes that • self-testing features verify that faults requiring detection are detected. • self-testing features verify that upon detection, the system responds according to the type of fault. • self-testing features verify that faults are detected and responded within a sufficient time frame to ensure safety function is not lost. • the presence and type of fault is indicated by the MPS alarms and displays.
13.	The PAM Type B and Type C displays are indicated on the SDIS displays in the MCR.	An inspection will be performed for the ability to retrieve the as-built PAM Type B and Type C displays on the SDIS displays in the MCR.	The PAM Type B and Type C displays listed in FSAR Table 7.1-7 are retrieved and displayed on the SDIS displays in the MCR.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
02.05.01	MPS Design and Implementation FSAR Section 7.2.1, Quality, discusses the software life-cycle phases for the MPS. The purpose is to verify software implementation based on licensing commitments to 10 CFR Part 50, Appendix A, GDC 1 (Quality), Appendix B (Quality Assurance Criteria), RGs 1.28, 1.152, 1.168, 1.169, 1.170, 1.171, 1.172, and 1.173, and the associated IEEE standards. The licensee shall perform analyses for each phase and generate technical reports to conclude that the life-cycle phases are implemented per the licensing commitments. The requirements of IEEE Std 829-2008 “IEEE Standard for Software and System Test Documentation,” are tailored to the NuScale I&C development life cycle, which is different than that of the conceptual waterfall life cycle listed in RG 1.152. The ITAAC verifies that output documentation of each software life-cycle phase satisfies the requirements of that phase for the MPS, and that software are implemented per licensing commitments to 10 CFR Part 50, Appendix A, GDC 1 (Quality), Appendix B (Quality Assurance Criteria), RGs 1.28, 1.152, 1.168, 1.169, 1.170, 1.171, 1.172, and 1.173, and the associated IEEE standards. Tunable Parameters FSAR Section 7.2.9, Control of Access, Identification, and Repair, discusses the protective measures that prevent modification of the MPS tunable parameters without proper configuration and authorization. Guidance on this issue is provided in DI&C-ISG-04, “Highly-Integrated Control Rooms - Communications Issues,” under interdivisional communications, staff position 10. A test demonstrates that protective measures restrict modification to the MPS tunable parameters without proper configuration and authorization. This test will be performed by attempting to modify the tunable parameters with the MPS not in the correct configuration or without authorization. Communication Independence FSAR Section 7.1.2, Independence, discusses the communication independence between redundant Class 1E digital communication system divisions. The purpose is to verify proper data isolation between redundant divisions. Requirements for independence are given in IEEE Std. 603-1991. Guidance for providing independence between redundant divisions of the Class 1E digital communication system is provided in Digital I&C Interim Staff Guidance (ISG) 04. A test demonstrates that independence between redundant divisions of the Class 1E MPS is provided. Automatic Reactor Trip Signals FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes automatic and manual reactor trips, variables that are monitored to provide input into automatic reactor trip signals, and the features of the RTS. The reactor trip functions are listed in FSAR Table 7.1-3: Reactor Trip Functions. The reactor trip logic for the monitored variables is provided in figures in FSAR Section 7.1.1. The MPS initiates an automatic reactor trip signal when the associated plant condition(s) exist. A test demonstrates that a reactor trip signal is automatically initiated for each reactor trip function listed in FSAR Table 7.1-3. The actuation of RTBs is not required for this test. The verification of the existence of a reactor trip signal is accomplished using MCR displays.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
	<u>Automatic ESF Actuation Signals</u> FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes automatic and manual ESFs actuations, variables that are monitored to provide input into automatic ESFs signals, and the features of the ESF systems. The ESFs functions are listed in FSAR Table 7.1-4: Engineered Safety Feature Actuation System Functions. The ESFs logic for the monitored variables is provided in figures in FSAR Section 7.1.1. The MPS initiates an automatic ESF actuation signal when the associated plant condition(s) exist. A test demonstrates that an automatic ESF actuation signal is automatically initiated for each of the ESF functions listed in Table 7.1-4. The actuation of ESFs equipment is not required for this test. The verification of the existence of an ESF actuation signal is accomplished using MCR displays. <u>RTBs Open, Automatic Trip</u> FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes automatic and manual reactor trips, variables that are monitored to provide input into automatic reactor trip signals, and the features of the RTS. The reactor trip functions are listed in FSAR Table 7.1-3: Reactor Trip Functions. The reactor trip logic for the monitored variables is provided in figures in FSAR Section 7.1.1. The MPS initiates an automatic reactor trip signal for the reactor trip functions when the associated plant condition(s) exist. A test demonstrates that the RTBs open when any one of the automatic reactor trip functions is initiated from the MCR. The RTBs are only opened once to satisfy this test objective. <u>RTBs Open, Manual Trip</u> FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes automatic and manual reactor trips, variables that are monitored to provide input into automatic reactor trip signals, and the features of the RTS. A manual reactor trip is one of the MPS manually actuated functions. A test demonstrates that the RTBs open when a reactor trip is manually initiated from the MCR. <u>Reactor Trip Logic</u> FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes that consistent with GDC 23, the MPS has sufficient functional diversity to prevent the loss of a protection function, to fail into a safe state or into a state demonstrated to be acceptable if conditions such as disconnection of the system, loss of power, or postulated adverse environments are experienced. A test demonstrates that when the loss of electrical power is detected in a separation group of the MPS that separation group fails to a safe state resulting in a reactor trip state for that separation group.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
	ESF Trip Logic Guidance provided in Design Specific Review Standard Section 7.2.3, Reliability, Integrity, and Completion of Protective Action, states that the design incorporate protective measures that provide for I&C safety systems to fail in a safe state, or into a state that is demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of power, or adverse environments, are experienced. FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes that consistent with GDC 23, the MPS has sufficient functional diversity to prevent the loss of a protection function, to fail into a safe state or into a state demonstrated to be acceptable if conditions such as disconnection of the system, loss of power, or postulated adverse environments are experienced. For an ESF function this predefined safe state may be that the actuated component remains as-is. A test demonstrates that when the loss of electrical power is detected in a separation group of the MPS that separation group fails to a safe state for that separation group. MPS Interlocks FSAR Section 7.2.4, Operating and Maintenance Bypasses, describes MPS operating bypasses for reactor trip functions and ESF actuations. The operating bypasses are applied automatically when plant conditions dictate that the safety function is not needed, or that the safety function prevents proper plant operation at a specific mode of operation. A test demonstrates that the MPS interlocks listed in FSAR Table 7.1-5 automatically establish an operating bypass for the specified reactor trip or ESF actuations when a real or simulated signal simulates that the associated interlock condition is met; and are automatically removed when the real or simulated signal simulates that the associated permissive condition is no longer satisfied. MPS Permissives FSAR Section 7.2.4, Operating and Maintenance Bypasses, describes MPS operating bypasses for reactor trip functions and ESF actuations. The only manual operating bypasses used for the design use a permissive in conjunction with the manual bypass in order to achieve the function of the bypass. A test demonstrates that the MPS permissives listed in FSAR Table 7.1-5 allow the manual bypass of the specified reactor trip or ESF actuations when a real or simulated signal simulates that the associated permissive condition is met; and are automatically removed when the real or simulated signal simulates that the associated permissive condition is no longer satisfied. MPS Overrides FSAR Section 7.2.4, Operating and Maintenance Bypasses, describes MPS operating bypasses for reactor trip functions and ESF actuations. For beyond design-basis events and for a limited number of actuated equipment, a safety-related Override switch can be used to prioritize a nonsafety signal over certain automatic signals. A test demonstrates that the MPS overrides listed in FSAR Table 7.1-5 are established when the manual Override switch is active and a real or simulated RT-1 interlock is established.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
	Maintenance Bypass FSAR Section 7.2.4, Operating and Maintenance Bypasses, describes the MPS maintenance bypass operation mode. An individual protection channel can be placed in a maintenance bypass operation mode to allow manual testing and maintenance during power operation, while ensuring that the minimum redundancy required by the Technical Specifications is maintained. The reactor trip functions are listed in FSAR Table 7.1-3: Reactor Trip Functions. The ESFs functions are listed in FSAR Table 7.1-4: Engineered Safety Feature Actuation System Functions. A test demonstrates that with a safety function module out of service switch activated, the safety function is placed in trip or bypass based on the position of the safety function module trip/bypass switch. Each separation group of the reactor trip functions listed in FSAR Table 7.1-3 and each separation group of the ESFs signals listed in FSAR Table 7.1-4 is tested by placing the separation group in maintenance bypass. RTB Arrangement FSAR Section 7.0.4, System Descriptions, discusses the arrangement of the protection system RTBs. FSAR Figure 7.0-6: Reactor Trip Breaker Arrangement provides the arrangement of the RTBs. This ITAAC verifies that the RTBs conform to the arrangement indicated in FSAR Figure 7.0-6. In addition, the ITAAC inspection verifies proper connection of the shunt and undervoltage trip mechanisms and other auxiliary contacts. Different Programmable Technology FSAR Section 7.1.5, Diversity and Defense-in-Depth, discusses that two of the four separation groups and one of the two divisions of RTS and ESFAS will utilize a different programmable technology. An ITAAC inspection is performed to verify that MPS separation groups A & C and Division I of RTS and ESFAS utilize a different programmable technology from separation groups B & D and Division II of RTS and ESFAS.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.05.02	FSAR Sections 7.1.2, Independence, discusses the independence of the MPS Class 1E I&C current-carrying circuits per the guidance of RG 1.75, which endorses IEEE Std. 384-1992. Physical separation is provided to maintain the independence of Class 1E I&C current-carrying circuits so that the safety functions required during and following any design basis event can be accomplished. Minimum separation distance (as defined in IEEE Std. 384-1992), or barriers or any combination thereof may achieve physical separation as specified in IEEE Std. 384-1992. Separate ITAAC inspections are performed to verify the independence provided by physical separation and the independence provided by electrical isolation. This ITAAC verifies the independence of Class 1E current-carrying circuits by physical separation. An ITAAC inspection is performed of physical separation of the MPS Class 1E current-carrying circuits. The physical separation ITAAC inspection results verify that the following physical separation criteria are met: Physical separation between redundant divisions of the MPS Class 1E I&C current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers; and such physical separation satisfies the criteria of RG 1.75. The configuration of each as-built barrier agrees with its associated as-built drawing. Physical separation between the MPS Class 1E I&C current-carrying circuits and non-Class 1E I&C current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers; and such physical separation satisfies the criteria of RG 1.75. The configuration of each as-built barrier agrees with its associated as-built drawing.
02.05.03	FSAR Sections 7.1.2, Independence, discusses the independence of the MPS Class 1E I&C circuits per the criteria of RG 1.75, which endorses IEEE Std. 384-1992. Electrical isolation is provided between the redundant divisions of the MPS Class 1E I&C circuits, and between Class 1E I&C circuits and non-Class 1E I&C circuits by Class 1E isolation devices so a failure in an I&C circuit does not prevent safety-related function completion in a different Class 1E I&C circuit. An ITAAC inspection is performed to verify the following electrical isolation criteria are met: (1) Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between redundant divisions of the MPS Class 1E I&C circuits. (2) Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between the MPS Class 1E I&C circuits and non-Class 1E I&C circuits.
02.05.04	FSAR Sections 7.1.2, Independence, discusses the independence of MPS Class 1E circuits per the criteria of RG 1.75, which endorses IEEE Std. 384-1992. Electrical isolation is provided between Class 1E circuits and non-Class 1E circuits by Class 1E isolation devices so a failure in a non-Class 1E circuit does not prevent the safety-related function completion in the Class 1E circuit. The ITAAC verifies that: (1) an equipment qualification data report exists for the Class 1E isolation devices, and (2) the equipment qualification data report concludes that the Class 1E isolation devices performs its safety-related function under the design basis environmental conditions specified in the equipment qualification data report. An ITAAC inspection is performed to verify that Class 1E electrical isolation devices are installed between MPS Class 1E circuits and non-Class 1E circuits, which satisfy the guidance of RG 1.75.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.05.05	FSAR Section 7.1.2, Independence, discusses the communication independence between Class 1E digital communication systems and non-Class 1E digital communication systems. The purpose is to verify that logical or software malfunction of the nonsafety-related system cannot affect the functions of the safety system. Requirements for independence are given in IEEE Std. 603-1991. Guidance for providing independence between the Class 1E digital communication system and non-Class 1E digital communication systems is provided in Digital Instrumentation and Controls ISG 04. A vendor test demonstrates that independence between the Class 1E MPS and non-Class 1E digital systems is provided.
02.05.06	FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes automatic and manual ESFs actuations, variables that are monitored to provide input into automatic ESFs signals, and the features of the ESF systems. The ESFs functions are listed in FSAR Table 7.1-4: Engineered Safety Feature Actuation System Functions. The ESFs logic for the monitored variables is provided in figures in FSAR Section 7.1. The MPS initiates an automatic ESF actuation signal for the functions listed in FSAR Table 7.1-4 when the associated plant condition(s) exist. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that ESF equipment automatically actuates to perform its safety-related function listed in FSAR Table 7.1-4 upon an injection of a single simulated MPS signal.
02.05.07	FSAR Section 7.1.1, Design Bases and Additional Design Considerations, describes manual ESFs actuation variables that are monitored to provide input into automatic ESFs signals, and the features of the ESF system. The ESFs functions that can be manually actuated are shown in figures in FSAR Section 7.1. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that the MPS actuates the ESF equipment to perform its safety-related function listed in FSAR Table 7.1-4 when manually initiated.
02.05.08	FSAR Section 7.2.3, Reliability, Integrity, and Completion of Protective Action, describes compliance with requirements for completion of protective actions, which requires that, once initiated, the reactor trip and ESF proceed to completion and remain in their required position/condition until the actuation system is reset and operator action is taken. IEEE 603-1991 Clause 5.2 states that “The safety systems shall be designed so that, once initiated automatically or manually, the intended sequence of protective actions of the execute features shall continue until completion. Deliberate operation action shall be required to return the safety systems to normal. This requirement shall not preclude the use of equipment protective devices identified in [Clause] 4.11 of the design basis or the provisions for deliberate operator interventions. Seal-in of individual channels is not required.” In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that i. upon an MPS reactor trip signal listed in FSAR Table 7.1-3, the RTBs open and the RTBs do not automatically close when the MPS reactor trip signal clears. ii. upon an MPS ESF actuation signal listed in FSAR Table 7.1-4, the ESF equipment actuates to perform its safety-related function and continues to maintain its safety-related position and perform its safety-related function when the MPS ESF actuation signal clears.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
02.05.09	FSAR Section 7.2.12, Automatic and Manual Control, describes the signals and initiating logic for each reactor trip and ESF actuation. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that the measured time for the reactor trip functions listed in FSAR Table 7.1-3 is less than or equal to the maximum values assumed in the accident analyses. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that the measured time for the ESF functions listed in FSAR Table 7.1-4 is less than or equal to the maximum values assumed in the accident analyses. Technical Specification SR 3.0.1 bases states that surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire surveillance is performed within the specified frequency. The technical specification bases also describe an allowance for response time to be verified by any series of sequential, overlapping, or total channel measurements.
02.05.10	FSAR Section 7.2.4, Operating and Maintenance Bypasses, describes the MPS operating bypasses. These bypasses either automatically or manually block certain protective actions that otherwise prevent mode changes during plant operation. Indication is provided in the control room if some part of the system has been bypassed or taken out of service. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that each operational MPS manual or automatic bypass is indicated in the MCR.
02.05.11	FSAR Section 7.2.4, Operating and Maintenance Bypasses, describes the MPS maintenance bypass operation mode. An individual protection channel can be placed in a maintenance bypass operation mode to allow manual testing and maintenance during power operation, while ensuring that the minimum redundancy required by the technical specifications is maintained. FSAR Section 7.2.4 discusses the status indication of MPS maintenance bypasses placed in maintenance bypass operation mode. In accordance with FSAR Table 14.2-56, a preoperational test demonstrates that each MPS maintenance bypass is indicated in the MCR.
02.05.12	This ITAAC is intended to address self-testing features credited towards surveillance or other operational testing. Given the nature of this ITAAC, it is acceptable to verify ITAAC completion during the factory acceptance testing. Self-testing features include, but are not limited to, watchdog timers, automated channel checks, and signal input comparisons. FSAR Section 7.2.15, Capability for Test and Calibration, discusses the self-testing features of the MPS, including the types of faults that should be detected, the system responses to such faults, the required response times, and the ability for alarms and displays in the MCR to provide indication of such faults' existence. These tests of the MPS self-testing features ensure that a) faults requiring detection are detected, b) the system responds appropriately to each fault based on the type of fault, c) the response occurs within a sufficient time frame to ensure safety function is not lost, and d) that alarms and indications in the MCR indicate the type of fault present. A vendor test demonstrates and a report exists and concludes that • self-testing features verify that faults requiring detection are detected. • self-testing features verify that upon detection, the system responds according to the type of fault. • self-testing features verify that faults are detected and responded within a sufficient time frame to ensure safety function is not lost. • self-testing features verify that detected faults are indicated by alarms and displays.

Table 2.5-2: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1] (Continued)

ITAAC No.	Discussion
02.05.13	FSAR Section 7.1.1, Design Bases and Additional Design Considerations, and FSAR Section 7.2.13, Displays and Monitoring, describe the PAM Type B and C displays and alarms indicated on the SDIS displays in the MCR. The PAM Type B and C variables are developed in accordance with the guidance in RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” which endorses (with certain clarifying regulatory positions specified in Section C of this guide) IEEE Std. 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations.” In accordance with FSAR Table 14.2-59, a preoperational test demonstrates the ability to retrieve and display the various PAM Type B and C parameters at the as-built safety display indication displays in the MCR. The intent is to verify that the displays function during testing of the integrated as-built system; however, separate testing of the actual operation of the PAM displays using simulated signals may be acceptable where this is not practicable.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 2.5-3: Module Protection System and Safety Display and Indication System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
02.05.01	X					
02.05.02	X					
02.05.03	X					
02.05.04	X					
02.05.05	X					
02.05.06	X					
02.05.07	X					
02.05.08	X					
02.05.09	X					
02.05.10	X					
02.05.11	X					
02.05.12	X					
02.05.13	X					

2.6 Neutron Monitoring System

2.6.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the NMS, which is described in FSAR Section 7.0.4. Each NPM has its own module-specific NMS.

The NMS performs the following safety-related system function that is verified by ITAAC:

- The NMS supports the MPS by providing neutron flux data for various reactor trips.

Design Commitments

- Electrical isolation exists between the NMS Class 1E circuits and connected non-Class 1E circuits to prevent the propagation of credible electrical faults.
- Physical separation exists between the redundant divisions of the NMS Class 1E instrumentation and control current-carrying circuits, and between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.
- Electrical isolation exists between the redundant divisions of the NMS Class 1E instrumentation and control circuits, and between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.

2.6.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.6-1 contains the ITAAC for the NMS.

Table 2.6-1: Neutron Monitoring System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.06.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	Electrical isolation exists between the NMS Class 1E circuits and connected non-Class 1E circuits to prevent the propagation of credible electrical faults.	i. A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E isolation devices.	i. The Class 1E circuit does not degrade below defined acceptable operating levels when the non-Class 1E side of the isolation device is subjected to the maximum credible voltage, current transients, shorts, grounds, or open circuits.
		ii. An inspection will be performed of the NMS Class 1E as-built circuits.	ii. Class 1E electrical isolation devices are installed between NMS Class 1E circuits and connected non-Class 1E circuits.
02.	Physical separation exists between the redundant divisions of the NMS Class 1E instrumentation and control current-carrying circuits, and between Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits.	An inspection will be performed of the NMS Class 1E as-built instrumentation and control current-carrying circuits.	i. Physical separation between redundant divisions of NMS Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers. ii. Physical separation between NMS Class 1E instrumentation and control current-carrying circuits and non-Class 1E instrumentation and control current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers.
03.	Electrical isolation exists between the redundant divisions of the NMS Class 1E instrumentation and control circuits, and between Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits to prevent the propagation of credible electrical faults.	An inspection will be performed of the NMS Class 1E as-built instrumentation and control circuits.	i. Class 1E electrical isolation devices are installed between redundant divisions of NMS Class 1E instrumentation and control circuits. ii. Class 1E electrical isolation devices are installed between NMS Class 1E instrumentation and control circuits and non-Class 1E instrumentation and control circuits.

Table 2.6-2: Neutron Monitoring System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
02.06.01	FSAR Section 7.1.2, Independence, discusses the independence of the NMS Class 1E circuits. Electrical isolation is provided between Class 1E circuits and non-Class 1E circuits by Class 1E isolation devices so a failure in a non-Class 1E circuit does not prevent the safety-related function completion in the Class 1E circuit. A type test, analysis, or a combination of type test and analysis will be performed of the Class 1E isolation devices to verify that the Class 1E circuit does not degrade below defined acceptable operating levels when the non-Class 1E side of the isolation device is subjected to the maximum credible voltage, current transients, shorts, grounds, or open circuits. An ITAAC inspection is performed to verify that Class 1E electrical isolation devices are installed between NMS Class 1E circuits and non-Class 1E circuits.
02.06.02	FSAR Section 7.1.2, Independence discusses the independence of the NMS Class 1E I&C current-carrying circuits per the guidance of RG 1.75, which endorses IEEE Std. 384-1992. Physical separation is provided to maintain the independence of Class 1E I&C current-carrying circuits so that the safety functions required during and following any design basis event can be accomplished. Minimum separation distance (as defined in IEEE Std. 384-1992), or barriers or any combination thereof may achieve physical separation as specified in IEEE Std. 384-1992. Separate ITAAC inspections are performed to verify the independence provided by physical separation and the independence provided by electrical isolation. This ITAAC verifies the independence of Class 1E current-carrying circuits by physical separation. An ITAAC inspection is performed of physical separation of the NMS Class 1E current-carrying circuits. The physical separation ITAAC inspection results verify that the following physical separation criteria are met: i. Physical separation between redundant divisions of the NMS Class 1E I&C current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers; and such physical separation satisfies the criteria of RG 1.75. The configuration of each as-built barrier agrees with its associated as-built drawing. ii. Physical separation between the NMS Class 1E I&C current-carrying circuits and non-Class 1E I&C current-carrying circuits is provided by a minimum separation distance, or by barriers (where the minimum separation distances cannot be maintained), or by a combination of separation distance and barriers. The configuration of each as-built barrier agrees with its associated as-built drawing.
02.06.03	FSAR Section 7.1.2, Independence, discusses the independence of the NMS Class 1E I&C circuits per the criteria of RG 1.75, which endorses IEEE Std. 384-1992. Electrical isolation is provided between the redundant divisions of the NMS Class 1E I&C circuits, and between Class 1E I&C circuits and non-Class 1E I&C circuits by Class 1E isolation devices so a failure in an I&C circuit does not prevent safety-related function completion in a different Class 1E I&C circuit. An ITAAC inspection is performed to verify the following electrical isolation criteria are met: i. Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between redundant divisions of the NM system Class 1E I&C circuits. ii. Class 1E electrical isolation devices that satisfy the criteria of RG 1.75 are installed between the NMS Class 1E I&C circuits and non-Class 1E I&C circuits.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 2.6-3: Neutron Monitoring System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
02.06.01	X					
02.06.02	X					
02.06.03	X					

2.7 Radiation Monitoring - Module-Specific

2.7.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is automatic actions of various systems based on radiation monitoring. The components actuated by these automatic radiation monitoring functions are contained in module-specific systems.

Design Commitments

- The CES automatically responds to the CES and radiation monitor high radiation signals listed in Table 2.7-3 to mitigate a release of radioactivity.
- The CVCS automatically responds to the CVCS high radiation signals listed in Table 2.7-3 to mitigate a release of radioactivity.
- The auxiliary boiler system (ABS) automatically responds to the ABS high radiation signals listed in Table 2.7-3 to mitigate a release of radioactivity.
- The MSS automatically responds to the MSS high radiation signals listed in Table 2.7-3 to mitigate a release of radioactivity.

2.7.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7-1 contains the ITAAC for the radiation monitoring - module-specific automatic actions.

Table 2.7-1: Radiation Monitoring - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.07.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The CES automatically responds to the CES and radiation monitor high radiation signals listed in Table 2.7-3 to mitigate a release of radioactivity.	A test will be performed of the CES and radiation monitor high radiation signals listed in Table 2.7-3.	Upon initiation of the real or simulated CES and radiation monitor high radiation signals listed in Table 2.7-3, the CES automatically aligns/actuates the identified components to the positions identified in the table.
02.	The CVCS automatically responds to the CVCS high radiation signal listed in Table 2.7-3 to mitigate a release of radioactivity.	A test will be performed of the CVCS high radiation signal listed in Table 2.7-3.	Upon initiation of the real or simulated CVCS high radiation signal listed in Table 2.7-3, the CVCS automatically aligns/actuates the identified component to the position identified in the table.
03.	The ABS automatically responds to the ABS high radiation signals listed in Table 2.7-3 to mitigate a release of radioactivity.	A test will be performed of the ABS high radiation signals listed in Table 2.7-3.	Upon initiation of the real or simulated ABS high radiation signals listed in Table 2.7-3, the ABS automatically aligns/actuates the identified component to the position identified in the table.
04.	The MSS automatically responds to the MSS high radiation signals listed in Table 2.7-3 to mitigate a release of radioactivity.	A test will be performed of the MSS high radiation signals listed in Table 2.7-3.	Upon initiation of the real or simulated MSS high radiation signals listed in Table 2.7-3, the MSS automatically aligns/actuates the identified components to the position identified in the table.

Table 2.7-2: Radiation Monitoring - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
02.07.01	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For the CES and radiation monitor high radiation signals listed in Table 2.7-3, the CES automatically aligns the components identified in Table 2.7-3 to the required positions identified in the table. In accordance with FSAR Table 14.2-36, a preoperational test demonstrates the CES automatically aligns the components identified in Table 2.7-3 to the required positions identified in the table upon initiation of a real or simulated CES high radiation signal from CES-RIT-1011. In accordance with FSAR Table 14.2-36, a preoperational test demonstrates the CES automatically aligns the component identified in Table 2.7-3 to the required positions identified in the table upon initiation of a real or simulated radiation monitor high radiation signal from 00-RM-SKD-RW070-02.
02.07.02	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For the CVCS high radiation signal listed in Table 2.7-3, the CVCS automatically aligns the component identified in Table 2.7-3 to the required position identified in the table. In accordance with FSAR Table 14.2-33, a preoperational test demonstrates the CVCS automatically aligns the component identified in Table 2.7-3 to the required position identified in the table upon initiation of a real or simulated CVCS high radiation signal from CVC-RIT-1004.
02.07.03	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each ABS high radiation signal listed in Table 2.7-3, the ABS automatically aligns the component identified in Table 2.7-3 to the required position identified in the table. In accordance with the information presented in FSAR Table 14.2-6, a preoperational test demonstrates the ABS automatically aligns the component identified in Table 2.7-3 to the required position identified in the table upon initiation of a real or simulated ABS high radiation signal from 00-AB-RT-1010, 00-AB-RT-1011, 00-AB-RT-1012, 00-AB-RT-1014, and 00-AB-RT-1015.
02.07.04	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each MSS high radiation signal listed in Table 2.7-3, the MSS automatically aligns the components identified in Table 2.7-3 to the required position identified in the table. In accordance with the information presented in FSAR Table 14.2-24, a preoperational test demonstrates the MSS automatically aligns the components identified in Table 2.7-3 to the required position identified in the table upon initiation of a real or simulated MSS high radiation signal from MS-RIT-1002 and MS-RIT-2002.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 2.7-3: Radiation Monitoring - Module-Specific Automatic Actions

Radiation Monitor ID(s)	Variable(s) Monitored	Actuated Component(s)	Component ID(s)	Component Actions(s)
CES-RIT-1011	CES vacuum pump discharge	1. CES charcoal filter unit inlet isolation valve 2. CES discharge flow to gaseous radioactive waste system (GRWS) isolation valve 3. CES effluent to process sample panel isolation valve 4. CES process sample panel return isolation valve 5. CES service air connection isolation valve 6. CES purge air solenoid valve to CES vacuum pump A 7. CES purge air solenoid valve to CES vacuum pump B 8. CES vacuum pump A and B isolation valves	1. CE-A0V-0104 2. CE-A0V-0040 3. CE-SV-0037 4. CE-SV-0038 5. CE-AOV-0020 6. CE-SV-0026A 7. CE-SV-0026B 8. CE-AOV-0023A/ CE-AOV-0023B	1. Close 2. Open 3. Close 4. Close 5. Close 6. Close 7. Close 8. Close
00-RM-SKD-RW070-02 (radiation monitor)	GRWS cubicle area	CES to GRWS vapor condenser isolation valve	CE-AOV-0040	Close
CVC-RIT-1004	RCS discharge to regenerative heat exchanger	RCS discharge to process sampling system isolation valve	CVC-AOV-0059	Close
1. 00-AB-RT-1010 2. 00-AB-RT-1011 3. 00-AB-RT-1012 4. 00-AB-RT-1014 5. 00-AB-RT-1015	1. Auxiliary boiler skid vent 2. Auxiliary boiler skid and ABS superheater skid drain 3. ABS superheater skid inlet vent 4. ABS header drain 5. Turbine Generator Building auxiliary steam header	Module main steam to ABS steam header isolation valve	AB-AOV-0108	Close
1. MS-RIT-1002 2. MS-RIT-2002	1. SG #1 main steam line radiation 2. SG #2 main steam line radiation	1. Main steam common steam header drain pot control valve 2. SG #1 drain pot control valve 3. SG #2 drain pot control valve	1. MS-AOV-0020 2. MS-AOV-0112 3. MS-AOV-0212	1, Close 2. Close 3. Close

Table 2.7-4: Radiation Monitoring - Module-Specific Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
02.07.01	X					
02.07.02	X					
02.07.03	X					
02.07.04	X					

3.0 Shared Structures, Systems, and Components and Non-Structures, Systems, and Components Based Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Design Descriptions and ITAAC

This chapter provides the structures, systems, and components ITAAC Design Descriptions and ITAAC for those SSC that are common or shared by multiple NPMs. This chapter also includes non-SSC based ITAAC Design Descriptions and ITAAC that are common or shared by multiple NPMs. 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 3.16 shall only be completed once in conjunction with the ITAAC in Chapter 2 for the lead NPM.

Table 3.0-1: Shared Systems Subject to Inspections, Tests, Analyses, and Acceptance Criteria

Shared System	NPMs Supported
Balance-of-plant drain system	1 system per 6 modules
Containment flooding and drain system	1 system per 6 modules
Normal control room heating ventilation and air conditioning system	1 system per 6 modules
Control room habitability system	1 system per 6 modules
Reactor Building heating ventilation and air conditioning system	1 system per 6 modules
Fuel handling equipment system	1 system per 6 modules
Fuel storage system	1 system per 6 modules
Ultimate heat sink	1 system per 6 modules
Fire protection system	1 system per 6 modules
Plant lighting system	1 system per 6 modules
Gaseous radioactive waste system	1 system per 6 modules
Liquid radioactive waste system	1 system per 6 modules
Auxiliary boiler system	1 system per 6 modules
Radioactive waste drain system	1 system per 6 modules
Overhead heavy load handling system	1 system per 6 modules
Reactor Building and Reactor Building components	1 system per 6 modules
Radioactive Waste Building	1 system per 6 modules
Control Building	1 system per 6 modules
Physical security system	1 system per 6 modules
Site cooling water system	1 system per 6 modules
Demineralized water system	1 system per 6 modules

3.1 Control Room Habitability

3.1.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the control room habitability system (CRHS), which is described in FSAR Section 6.4. The CRHS supports up to six NPMs.

The CRHS performs the following nonsafety-related system functions that are verified by ITAAC. The CRHS supports the Control Building (CRB) by providing clean breathing air to the MCR and maintains a positive control room pressure during high radiation or loss of normal AC power conditions.

Design Commitments

- The air exfiltration out of the control room envelope (CRE) is less than or equal to the assumptions used to size the CRHS inventory and the supply flow rate.
- The CRHS valves listed in Table 3.1-3 change position under design basis temperature, differential pressure, and flow conditions.
- The CRHS solenoid-operated valves listed in Table 3.1-3 perform their function to fail open on loss of motive power under design basis temperature, differential pressure, and flow conditions.
- The CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a DBA.
- The CRHS maintains a positive pressure in the MCR relative to the adjacent areas.

3.1.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.1-1 contains the ITAAC for the CRHS.

Table 3.1-1: Control Room Habitability System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.01.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The air exfiltration out of the CRE is less than or equal to the assumptions used to size the CRHS inventory and the supply flow rate.	A test will be performed of the CRE.	The air exfiltration measured by tracer gas testing is less than or equal to the CRE air infiltration rate assumed in the dose analysis.
02.	The CRHS valves listed in Table 3.1-3 change position under design basis temperature, differential pressure, and flow conditions.	A test will be performed of the CRHS valves listed in Table 3.1-3 under preoperational temperature, differential pressure, and flow conditions.	Each CRHS valve listed in Table 3.1-3 strokes fully open and fully closed by remote operation under preoperational temperature, differential pressure, and flow conditions.
03.	The CRHS solenoid-operated valves listed in Table 3.1-3 perform their function to fail open on loss of motive power under design basis temperature, differential pressure, and flow conditions.	A test will be performed of the CRHS solenoid-operated valves listed in Table 3.1-3 under preoperational temperature, differential pressure and flow conditions.	Each CRHS solenoid-operated valve listed in Table 3.1-3 performs its function to fail open on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
04.	The CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a DBA.	An analysis will be performed of the as-built CRE heat sinks.	A report exists and concludes that the CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a DBA.
05.	The CRHS maintains a positive pressure in the MCR relative to adjacent areas.	A test will be performed of the CRHS.	The CRHS maintains a positive pressure of greater than or equal to 1/8 inches water gauge in the CRE relative to adjacent areas, while operating in DBA alignment.

Table 3.1-2: Control Room Habitability System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
03.01.01	Testing is performed on the CRE in accordance with RG 1.197, “Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors,” Revision 0, to demonstrate that air exfiltration from the CRE is controlled. RG 1.197 allows two options for CRE testing; either integrated testing (tracer gas testing) or component testing. FSAR Section 6.4, Control Room Habitability, describes the testing requirements for the CRE habitability program and provides the maximum air exfiltration allowed from the CRE. In accordance with FSAR Table 14.2-15, a preoperational test using the tracer gas test method demonstrates that the air exfiltration from the CRE does not exceed the assumed unfiltered leakage rate for the dose analysis. Tracer gas testing in accordance with ASTM E741 will be performed to measure the unfiltered in-leakage into the CRE with the CRHS operating.
03.01.02	The CRHS valves are tested by remote operation to demonstrate the capability to perform their function to transfer open and transfer closed under preoperational temperature, differential pressure, and flow conditions. In accordance with FSAR Table 14.2-15, a preoperational test demonstrates that each CRHS valve listed in Table 3.1-3 strokes fully open and fully closed by remote operation under preoperational test conditions. Preoperational test conditions are established that approximate design-basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
03.01.03	The CRHS solenoid-operated valves are tested to demonstrate the capability to perform their function to fail open on loss of motive power under preoperational temperature, differential pressure, and flow conditions. In accordance with FSAR Table 14.2-15 a preoperational test demonstrates that each CRHS solenoid-operated valve listed in Table 3.1-3 repositions to the open position on loss of motive power (electric power to the valve actuating solenoid(s) is lost, or pneumatic pressure to the valve(s) is lost). Preoperational test conditions are established that approximate design basis temperature, differential pressure, and flow conditions to the extent practicable, consistent with preoperational test limitations.
03.01.04	FSAR Section 6.4.4, Design Evaluation, discusses the thermal mass of the CRB and its contents limit the temperature increase as shown in Table 6.4-2 within the CRE within an acceptable range for the first 72 hours following a DBA. An analysis confirms that the CRE bulk average air temperature is acceptable on a loss of active cooling for the first 72 hours following a DBA.
03.01.05	FSAR Section 6.4.3.2, System Operation, discusses the operation of the CRHS, which maintains a positive pressure in the MCR relative to the adjacent areas. FSAR Table 6.4-1, Control Room Habitability System Design Parameters, provides the required positive pressure in the MCR relative to the adjacent areas. In accordance with FSAR Table 14.2-15, a preoperational test demonstrates that the CRHS maintains a positive pressure of greater than or equal to 1/8 inches water gauge in the MCR relative to adjacent areas, while operating in a DBA alignment.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.1-3: Control Room Habitability System Mechanical Equipment

Equipment Name	Equipment Identifier	Failure Position
Air supply isolation solenoid valves	00-CRH-SV-0007A 00-CRH-SV-0007B	Open
CRE pressure relief isolation valves	00-CRH-SV-0028A 00-CRH-SV-0028B	Open

Table 3.1-4: Control Room Habitability System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.01.01			X			
03.01.02			X			
03.01.03			X			
03.01.04	X					
03.01.05			X			

3.2 Normal Control Room Heating Ventilation and Air Conditioning System

3.2.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the normal control room HVAC system (CRVS), which is described in FSAR Section 9.4.1. The CRVS supports up to six NPMs.

The CRVS performs the following nonsafety-related system functions that are verified by ITAAC:

- The CRVS supports the CRB by providing isolation of the CRE from the surrounding areas and outside environment via isolation dampers.
- The CRVS supports the CRB by maintaining the CRB at a positive pressure relative to the RXB and the outside atmosphere to control the ingress of potentially airborne radioactivity from the RXB or the outside atmosphere to the CRB.
- The CRVS supports the EDAS by providing ventilation to maintain airborne hydrogen concentrations below the allowable limits.
- The CRVS supports the normal DC power system by providing ventilation to maintain airborne hydrogen concentrations below allowable limits.

Design Commitments

- The CRVS air-operated CRE isolation dampers listed in Table 3.2-3 perform their function to fail to the closed position on loss of motive power under design basis temperature, differential pressure, and flow conditions.
- The CRVS maintains a positive pressure in the CRB relative to the outside environment.
- The CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.

3.2.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.2-1 contains the ITAAC for the CRVS.

Table 3.2-1: Normal Control Room Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.02.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The CRVS air-operated CRE isolation dampers listed in Table 3.2-3 perform their function to fail to the closed position on loss of motive power under design basis temperature, differential pressure, and flow conditions.	A test will be performed of the air-operated CRE isolation dampers listed in Table 3.2-3 under preoperational temperature, differential pressure and flow conditions.	Each CRVS air-operated CRE isolation damper listed in Table 3.2-3 performs its function to fail to the closed position on loss of motive power under preoperational temperature, differential pressure, and flow conditions.
02.	The CRVS maintains a positive pressure in the CRB relative to the outside environment.	A test will be performed of the CRVS while operating in the normal operating alignment.	The CRVS maintains a positive pressure of greater than or equal to 1/8 inches water gauge in the CRB relative to the outside environment, while operating in the normal operating alignment.
03.	The CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.	A test will be performed of the CRVS while operating in the normal operating alignment.	The airflow capability of the CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.

Table 3.2-2: Normal Control Room Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
03.02.01	The CRVS control room envelope isolation dampers are tested to demonstrate the capability to perform their function to fail to the closed position on loss of motive power. In accordance with FSAR Table 14.2-16, a preoperational test demonstrates that each CRVS air-operated CRE isolation damper listed in Table 3.2-3 repositions to the closed position on loss of motive power (electric power to the valve actuating solenoid is lost, or pneumatic pressure to the damper is lost). Preoperational test conditions are established that approximate design differential pressure conditions to the extent practicable, consistent with preoperational test limitations. A manual signal, actual automatic signal, or simulated automatic signal may be used to operate the valves because the control logic of the valves is not being verified by this ITAAC.
03.02.02	FSAR Section 9.4.1.2, System Description, discusses the operation of the CRVS, which maintains a positive pressure in the CRB relative to the outside environment. This is consistent with the requirements of 10 CFR Part 20, Subparts E and H and 10 CFR Part 50, Appendix I. In accordance with FSAR Table 14.2-16 a preoperational test demonstrates that the CRVS will maintain a positive pressure of greater than or equal to 1/8 inches water gauge in the CRB relative to the outside environment, while operating in a normal operating alignment.
03.02.03	FSAR Section 9.4.1, Control Room Area Ventilation System, provides a discussion of how the CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume. In accordance with FSAR Table 14.2-16, a preoperational test demonstrates that the airflow capability of the CRVS maintains the hydrogen concentration levels in the CRB battery rooms containing batteries below one percent by volume.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.2-3: Normal Control Room HVAC System Mechanical Equipment

Equipment Name	Equipment Identifier	Actuator Type
CRE isolation dampers	00-CRV-MBD-0026 00-CRV-MBD-0027 00-CRV-MBD-0038 00-CRV-MBD-0039 00-CRV-MBD-0092 00-CRV-MBD-0093	Motor-operated

Table 3.2-4: Normal Control Room Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.02.01			X			
03.02.02			X			
03.02.03					X	

3.3 Reactor Building Heating Ventilation and Air Conditioning System

3.3.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the Reactor Building HVAC system (RBVS), which is described in FSAR Section 9.4.2. The RBVS supports up to six NPMs.

The RBVS performs the following nonsafety-related system functions that are verified by ITAAC:

- The RBVS supports the RXB by maintaining the RXB at a negative pressure relative to the outside atmosphere to control the movement of potentially airborne radioactivity from the RXB to the environment.
- The RBVS supports the Radioactive Waste Building (RWB) by maintaining the RWB at a negative ambient pressure relative to the outside atmosphere to control the movement of potentially airborne radioactivity from the RWB to the environment.
- The RBVS supports the EDAS by providing ventilation to maintain airborne hydrogen concentrations below allowable limits.
- The RBVS supports the normal DC power system by providing ventilation to maintain airborne hydrogen concentrations below allowable limits.

Design Commitments

- The RBVS maintains a negative pressure in the RXB relative to the outside environment.
- The RBVS maintains a negative pressure in the RWB relative to the outside environment.
- The RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.

3.3.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.3-1 contains the ITAAC for the RBVS.

Table 3.3-1: Reactor Building HVAC System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.03.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The RBVS maintains a negative pressure in the RXB relative to the outside environment.	A test will be performed of the RBVS while operating in the normal operating alignment.	The RBVS maintains a negative pressure in the RXB relative to the outside environment, while operating in the normal operating alignment.
02.	The RBVS maintains a negative pressure in the RWB relative to the outside environment.	A test will be performed of the RBVS while operating in the normal operating alignment.	The RBVS maintains a negative pressure in the RWB relative to the outside environment, while operating in the normal operating alignment.
03.	The RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.	A test will be performed of the RBVS while operating in the normal operating alignment.	The airflow capability of the RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.

Table 3.3-2: Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
03.03.01	FSAR Sections 9.4.2, Reactor Building and Spent Fuel Pool Area Ventilation System, discusses the operation of the RBVS that maintains a negative pressure in the RXB relative to the outside environment. This is consistent with the requirements of 10 CFR Part 20, Subparts E and H and 10 CFR Part 50, Appendix I. In accordance with FSAR Table 14.2-17, a preoperational test demonstrates that the RBVS will maintain a negative pressure in the RXB relative to the outside environment, while operating in a normal operating alignment.
03.03.02	FSAR Sections 9.4.2, Reactor Building and Spent Fuel Pool Area Ventilation System, discusses the operation of the RBVS that maintains a negative pressure in the RWB relative to the outside environment. This is consistent with the requirements of 10 CFR Part 20, Subparts E and H and 10 CFR Part 50, Appendix I. In accordance with FSAR Table 14.2-17, a preoperational test demonstrates that the RBVS will maintain a negative pressure in the RWB relative to the outside environment, while operating in a normal operating alignment.
03.03.03	FSAR Section 9.4.2, Reactor Building and Spent Fuel Pool Area Ventilation System, provides a discussion of how the RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume. In accordance with FSAR Table 14.2-17, a preoperational test demonstrates that the airflow capability of the RBVS maintains the hydrogen concentration levels in the RXB battery rooms containing batteries below one percent by volume.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.3-3: Reactor Building Heating Ventilation and Air Conditioning System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.03.01			X			
03.03.02			X			
03.03.03					X	

3.4 Fuel Handling Equipment System

3.4.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the fuel handling equipment (FHE) system, which is described in FSAR Section 9.1.4. The FHE system supports up to six NPMs.

The FHE system performs the following nonsafety-related system functions that are verified by ITAAC:

- The FHE system supports the reactor fuel assembly by providing structural support and handling capabilities of irradiated and spent fuel assemblies during refueling activities.
- The FHE system supports the reactor fuel assembly by providing structural support and handling capabilities of new fuel assemblies during refueling activities.
- The FHE system supports the reactor fuel assembly by providing the ability to visually inspect and repair irradiated fuel assemblies.

Design Commitments

- The FHE system equipment listed in Table 3.4-3 is single-failure-proof in accordance with the approved design.
- The FHE system equipment listed in Table 3.4-3 is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.
- All weld joints of the FHE system equipment listed in Table 3.4-3 whose failure could result in the drop of a critical load comply with the ASME NOG-1 Code or ASME NUM-1 Code listed in Table 3.4-3.
- The fuel handling machine (FHM) travel is limited to maintain a water inventory for personnel shielding with the pool level at the lower limit of the normal operating low water level.
- The new fuel elevator (NFE) travel is limited to maintain a water inventory for personnel shielding with the pool level at the lower limit of the normal operating low water level.

3.4.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.4-1 contains the ITAAC for the FHE system.

Table 3.4-1: Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.04.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The FHE system equipment listed in Table 3.4-3 is single-failure-proof in accordance with the approved design.	An inspection will be performed of the as-built FHE system equipment listed in Table 3.4-3.	A report exists and concludes that the FHE system equipment listed in Table 3.4-3 is single-failure-proof in accordance with the approved design.
02.	The FHE system equipment listed in Table 3.4-3 is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.	A rated load test will be performed of the FHE system equipment listed in Table 3.4-3.	The FHE system equipment listed in Table 3.4-3 lifts, supports, holds with the brakes, and transports a load of at least 125 percent of the manufacturer's rated capacity.
03.	All weld joints of the FHE system equipment listed in Table 3.4-3 whose failure could result in the drop of a critical load comply with the ASME NOG-1 Code or ASME NUM-1 Code listed in Table 3.4-3.	An inspection will be performed of the FHE system equipment listed in Table 3.4-3 as-built weld joints whose failure could result in the drop of a critical load.	The results of the non-destructive examination of the FHE system equipment listed in Table 3.4-3 weld joints whose failure could result in the drop of a critical load comply with the ASME NOG-1 Code or ASME NUM-1 Code listed in Table 3.4-3.
04.	The FHM travel is limited to maintain a water inventory for personnel shielding with the pool level at the lower limit of the normal operating low water level.	A test will be performed of the FHM vertical travel limit.	The FHM 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.
05.	The NFE travel is limited to maintain a water inventory for personnel shielding with the pool level at the lower limit of the normal operating low water level.	A test will be performed of the NFE vertical travel limit.	The NFE 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.

Table 3.4-2: Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
03.04.01	ITAACFSAR Section 9.1.4, Fuel Handling Equipment, describes that the FHE system equipment listed in Table 3.4-3 is classified as Type I equipment as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) and ASME NUM-1 Code, Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type). An ITAAC inspection is performed of the FHM main and auxiliary hoists machinery arrangement to verify the existence of the following single-failure proof features: 1) Non-redundant structural components (bridge, trolley, wire rope drum, and hook) are designed to appropriate standards, constructed from base material demonstrated to meet appropriate material properties 2) Redundant design features to stop and hold the load following: a. specified component failures (e.g., wire rope, drive train, and control system) b. operator errors (e.g., two-blocking and overload) An ITAAC inspection is performed of the new fuel jib crane and new fuel elevator machinery arrangements to verify the existence of single-failure-proof features so that any credible failure of a single component will not result in the loss of capability to stop and hold the critical load. This ITAAC inspection may be performed any time after manufacture of the FHE system equipment (at the factory or later).
03.04.02	FSAR Section 9.1.4, Fuel Handling Equipment, describes that the FHE system equipment listed in Table 3.4-3 is classified as Type I equipment as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) and ASME NUM-1 Code, Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type). The FHE system equipment listed in Table 3.4-3 is tested in accordance with the applicable requirements of ASME NOG-1 and ASME NUM-1. A site acceptance test demonstrates that each FHE system equipment listed in Table 3.4-3 is rated load tested at 125% (+5%, -0%) of the manufacturer's rating.
03.04.03	FSAR Section 9.1.4, Fuel Handling Equipment, describes that the FHE system equipment listed in Table 3.4-3 is classified as Type I equipment as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) and ASME NUM-1 Code, Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type). An ITAAC inspection is performed to verify that the ASME Type I FHE system equipment listed in Table 3.4-3 as-built welds are nondestructively examined in accordance with the standards of ASME NOG-1, ASME NUM-1, and the purchase specification. This ITAAC inspection may be performed any time after manufacture of the FHE system equipment listed in Table 3.4-3 (at the factory or later).
03.04.04	FSAR Section 9.1.4, Fuel Handling Equipment, describes that the FHE system has provisions to limit maximum height to maintain sufficient water inventory above the top of the fuel assembly. FHM mechanical and electrical interlocks limit FHM maximum lift height of a fuel assembly to ensure a shielding water depth of at least 10 feet. In accordance with FSAR Table 14.2-44, a preoperational test demonstrates that the FHM 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.

Table 3.4-2: Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1] (Continued)

ITAAC No.	Discussion
03.04.05	FSAR Section 9.1.4, Fuel Handling Equipment, describes that the FHE system has provisions to limit maximum height to maintain sufficient water inventory above the top of the fuel assembly. The NFE provides the ability to lift irradiated fuel assemblies for inspections and repairs with the spent fuel operating mode, and a vertical travel limit ensures a shielding water depth of at least 10 feet. A site acceptance test or preoperational test demonstrates that the NFE 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.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.4-3: Fuel Handling Equipment System Equipment

Equipment Identifier	Equipment Name	Design Code
00-FHE-CRN-0001	New fuel jib crane	ASME NUM-1, Type IA
00-FHE-CRN-0002	New fuel elevator hoist	ASME NUM-1, Type IA
00-FHE-CRN-0003	Fuel handling machine mast rotate and hoist, bridge, and trolley	ASME NOG-1, Type I
	Fuel handling machine auxiliary hoist	ASME NUM-1, Type IA

Table 3.4-4: Fuel Handling Equipment System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal/ External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.04.01	X					
03.04.02	X					
03.04.03	X					
03.04.04	X		X			
03.04.05	X		X			

3.5 Fuel Storage System

3.5.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the fuel storage system, which is described in FSAR Sections 9.1.1 and 9.1.2. The fuel storage system supports up to six NPMs.

The fuel storage system performs the following nonsafety-related system functions that are verified by ITAAC:

- The fuel storage system supports the reactor fuel assembly system by providing neutron absorption to ensure subcriticality during storage of new and spent fuel.

Design Commitments

- The fuel storage racks maintain an effective neutron multiplication factor (k-effective) within the following limits at a 95 percent probability, 95 percent confidence level when loaded with fuel of the maximum reactivity to assure subcriticality during plant life, including normal operations and postulated accident conditions:
  - k-effective must not exceed 0.95 if flooded with borated water
  - k-effective must not exceed 1.0 if flooded with unborated water

3.5.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.5-1 contains the ITAAC for the fuel storage system.

Table 3.5-1: Fuel Storage System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.05.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	Not Used		
02.	The fuel storage racks maintain an effective neutron multiplication factor (k-effective) within the following limits at a 95 percent probability, 95 percent confidence level when loaded with fuel of the maximum reactivity to assure subcriticality during plant life, including normal operations and postulated accident conditions: • k-effective must not exceed 0.95 if flooded with borated water • k-effective must not exceed 1.0 if flooded with unborated water	An inspection will be performed of the as-built fuel storage racks, their configuration in the SFP, and the associated documentation.	The as-built fuel storage racks, including any neutron absorbers, and their configuration within the SFP conform to the design values for materials and dimensions and their tolerances, as shown to be acceptable in the fuel storage criticality analysis described in the UFSAR.

Table 3.5-2: Fuel Storage System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[(1)]

ITAAC No.	Discussion
03.05.01	Not Used
03.05.02	FSAR Section 9.1.2, New and Spent Fuel Storage, discusses the fuel storage racks. An ITAAC inspection is performed to verify that the as-built fuel storage racks, including any neutron absorbers, conform to the design values for materials and dimensions and their tolerances, as presented in the approved criticality analysis.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.5-3: Fuel Storage System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.05.01						
03.05.02	X					

3.6 Ultimate Heat Sink

3.6.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the ultimate heat sink (UHS), which is described in Section 9.2.5. The UHS supports up to six NPMs.

The configuration of the UHS includes the combined volume of water in the reactor pool, refueling pool (RFP), and SFP. The pool areas are open to each other with a weir wall partially separating the SFP from the RFP. The dry dock area is not considered part of the UHS volume.

The structural components of the reactor pool, RFP, and SFP (i.e., structural walls, weir wall, and floor) and associated pool liners are components of the RXB structure. The design commitments for the RXB are provided in Section 3.11.

The UHS performs the following safety-related system functions that are verified by ITAAC:

- The UHS supports the CNTS by providing the removal of heat via direct water contact with the CNV.
- The UHS supports the DHRS by accepting the heat from the decay heat removal heat exchanger.
- The UHS supports the fuel storage system by providing the removal of decay heat from the spent fuel via direct water contact with the spent fuel assemblies.

The UHS performs the following nonsafety-related system functions that are verified by ITAAC:

- The UHS supports the CNTS by providing the radiation shielding for the NPMs via the water surrounding the components.
- The UHS supports the fuel storage system by providing radiation shielding for spent fuel via the water surrounding the components.

Design Commitment

- Drain down of the SFP, RFP, and reactor pool below the minimum safety water level is prevented by
  - i locating piping penetrations through the SFP, RFP, and reactor pool walls at or above the minimum safety water level, and
  - ii equipping each suction and discharge piping with an open end that extends below the minimum safety water level within the SFP, RFP, and reactor pool with an anti-siphon device that is located at or above the minimum safety water level.

3.6.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.6-1 contains the ITAAC for the UHS.

Table 3.6-1: Ultimate Heat Sink Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.06.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	Drain down of the SFP, RFP, and reactor pool below the minimum safety water level is prevented by i. locating piping penetrations through the SFP, RFP, and reactor pool walls at or above the minimum safety water level, and ii. equipping each suction and discharge piping with an open end that extends below the minimum safety water level within the SFP, RFP, and reactor pool with an anti-siphon device that is located at or above the minimum safety water level.	An inspection will be performed of i. each piping penetration through the SFP, RFP, and reactor pool walls, and ii. each suction and discharge piping with an open end within the SFP, RFP, and reactor pool.	i. Each piping penetration through the SFP, RFP, and reactor pool walls is located at or above the 75.5 ft building elevation (49.5 ft pool level as measured from the bottom of the SFP, RFP, and reactor pool). ii. Each suction and discharge piping with an open end that extends below the 75.5 ft building elevation within the SFP, RFP, and reactor pool is equipped with an anti-siphon device located at or above the 75.5 ft building elevation (49.5 ft pool level as measured from the bottom of the SFP, RFP, and reactor pool).

Table 3.6-2: Ultimate Heat Sink Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
03.06.01	FSAR Section 9.1.3, Pool Cooling and Cleanup System, discusses the pool cooling subsystem interface with the SFP, RFP, and reactor pool. Shielding is maintained by design of the penetrations of the inlet and outlet piping and the presence of anti-siphoning devices, both of which are at or above the 49.5 ft pool water level as seen in FSAR Figure 9.1.3-2. The design does not allow UHS pool water level to drain below the minimum level needed to support plant safety analyses and is above the level needed for adequate shielding of the spent fuel assemblies. An ITAAC inspection is performed to verify that each piping penetration through the SFP, RFP, and reactor pool walls is located at or above the 75.5 ft building elevation (49.5 ft pool level as measured from the bottom of the SFP, RFP, and reactor pool), and each suction and discharge piping with an open end that extends below the 75.5 ft building elevation in the SFP, RFP, and reactor pool is equipped with an anti-siphon device located at or above the 75.5 ft building elevation (49.5 ft pool level as measured from the bottom of the SFP, RFP, and reactor pool). This inspection is performed by physical measurements in the as-built SFP, RFP, and reactor pool.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.6-3: Ultimate Heat Sink Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.06.01	X					

3.7 Fire Protection System

3.7.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the fire protection system (FPS), which is described in FSAR Section 9.5.1. The FPS supports up to six NPMs.

The FPS performs the following nonsafety-related system functions that are verified by ITAAC:

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

Design Commitments

- Two separate firewater storage tanks provide a dedicated volume of water for firefighting.
- The FPS has a sufficient number of fire pumps to provide 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.
- 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.
- 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.
- An alternative shutdown capability that is physically and electrically independent of the MCR exists.
- A plant fire hazards analysis (FHA) considers potential fire hazards and ensures the fire protection features in each fire area are suitable for the hazards.

3.7.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.7-1 contains the ITAAC for the FPS.

Table 3.7-1: Fire Protection System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.07.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	Two separate firewater storage tanks provide a dedicated volume of water for firefighting.	An inspection will be performed of the as-built firewater storage tanks.	Each firewater storage tank provides a usable water volume dedicated for firefighting that is greater than or equal to 300,000 gallons.
02.	The FPS has a sufficient number of fire pumps to provide 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.	i. An analysis will be performed of the as-built fire pumps. ii. A test will be performed of the fire pumps.	i. A report exists and concludes that the fire pumps can provide 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. ii. Each fire pump delivers the design flow to the FPS.
03.	i. 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. ii. 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. iii. An alternative shutdown capability that is physically and electrically independent of the MCR exists.	A safe-shutdown analysis of the as-built plant will be performed, including a post-fire safe-shutdown (FSSD) circuit analysis.	A safe-shutdown analysis report exists and concludes that: i. 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 and that reentry into the fire area for repairs and operator actions is not possible. ii. 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. iii. The I&C equipment rooms within the RXB used as the alternative shutdown capability are physically and electrically independent of the MCR.
04.	A plant FHA considers potential fire hazards and ensures the fire protection features in each fire area are suitable for the hazards.	An FHA of the as-built plant will be performed.	An FHA report exists and concludes that: i. Combustible loads and ignition sources are accounted for. ii. Fire protection features are suitable for the hazards they are intended to protect against.

Table 3.7-2: Fire Protection System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[(1)]

ITAAC No.	Discussion
03.07.01	FSAR Section 9.5.1, Fire Protection Program, discusses how the FPS meets the guidance provided by RG 1.189 and applicable NFPA standards. Two separate dedicated 100 percent capacity freshwater storage tanks are provided. An ITAAC inspection is performed to verify that the minimum usable water volume of each firewater storage tank is greater than or equal to 300,000 gallons. If the storage tanks are also used as backup water sources for other non-fire emergencies, the ITAAC inspection verifies that the non-fire emergencies cannot drain the tank below the minimum dedicated useable water volume of 300,000 gallons required for firefighting.
03.07.02	FSAR Section 9.5.1, Fire Protection Program, discusses how the capacity of each FPS pump is adequate to supply the total flow demand at the pressure required at the pump discharge and provides the design flow of the fire pumps. An analysis confirms that the as-built fire pumps provide 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. In accordance with FSAR Table 14.2-22, a preoperational test demonstrates that each fire pump delivers the design flow to the FPS.
03.07.03	FSAR Section 9.5.1 discusses that (a) safe-shutdown can be achieved assuming that all equipment in any one fire area (except for the MCR and the area under the bioshield) is rendered inoperable by fire and that reentry into the fire area for repairs and operator actions is not possible, (b) that 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, and (c) an independent alternative shutdown capability that is physically and electrically independent of the MCR exists. A safe shutdown analysis of the as-built plant will be performed, including a post-FSSD circuit analysis performed in accordance with RG 1.189 and NEI 00-01 for all possible fire-induced failures that could affect the safe shutdown success path, including multiple spurious actuations. The safe shutdown analysis will verify that • safe shutdown can be achieved assuming that all equipment in any one fire area (except for the MCR and the area under the bioshield) is rendered inoperable by fire and that reentry into the fire area for repairs and operator actions is not possible. • 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. • I&C equipment rooms within the RXB used as the alternative shutdown capability are physically and electrically independent of the MCR.
03.07.04	FSAR Appendix 9A, Fire Hazards Analysis, discusses the methodology and presents the FHA for each fire area. The FHA must reflect the as-built configuration of the plant. The FHA is an analysis of the fire hazards, including combustible loading and ignition sources, and analysis of the fire protection features required to mitigate each postulated fire. An FHA of the as-built plant will be performed in accordance with RG 1.189, as described in FSAR Appendix 9A. The FHA will verify (1) combustible loads and ignition sources are accounted for, and (2) fire protection features are suitable for the hazards for which they are intended.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.7-3: Fire Protection System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.07.01					X	
03.07.02					X	
03.07.03					X	
03.07.04					X	

3.8 Plant Lighting System

3.8.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the plant lighting system (PLS), which is described in FSAR Section 9.5.3. The PLS supports up to six NPMs.

The PLS performs the following nonsafety-related system function that is verified by ITAAC. The PLS supports the CRB by providing emergency lighting in the MCR.

Design Commitment

- The PLS provides emergency illumination of the operator workstations in the MCR.

3.8.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.8-1 contains the ITAAC for the PLS.

Table 3.8-1: Plant Lighting System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.08.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The PLS provides emergency illumination of the operator workstations in the MCR.	A test will be performed of the MCR operator workstations illumination.	The PLS provides at least 10 foot-candles of illumination at the MCR operator workstations when it is the only MCR lighting system in operation.

Table 3.8-2: Plant Lighting System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
03.08.01	FSAR Section 9.5.3 discusses the PLS that provides emergency illumination of the operator workstations in the MCR. In accordance with FSAR Table 14.2-53, a preoperational test demonstrates that the PLS provides at least 10 foot-candles of illumination at the MCR operator workstations.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.8-3: Plant Lighting System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.08.01	X			X		

3.9 Radiation Monitoring - Shared-Systems

3.9.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is automatic actions of various systems based on radiation monitoring. The systems actuated by these automatic radiation monitoring functions are shared by NPMs 1 through 6.

Design Commitments

- The CRVS automatically responds to the CRVS high-radiation signals upstream of the CRVS filter unit listed in Table 3.9-3 to mitigate a release of radioactivity.
- The CRVS and the CRHS automatically respond to the CRVS high-radiation signals downstream of the CRVS filter unit listed in Table 3.9-3 to mitigate a release of radioactivity.
- The RBVS automatically responds to the RBVS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.
- The GRWS automatically responds to the GRWS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.
- The CFDS automatically responds to the CFDS high-radiation signal listed in Table 3.9-3 to mitigate a release of radioactivity.
- The balance-of-plant drain system automatically responds to the BPDS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.
- The liquid radioactive waste system (LRWS) automatically responds to the LRWS and radiation monitor high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.
- The ABS automatically responds to the ABS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.
- The DWS automatically responds to the DWS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.
- The radioactive waste drain system (RWDS) automatically responds to the RWDS high-radiation signal listed in Table 3.9-3 to mitigate a release of radioactivity.
- The site cooling water system (SCWS) automatically responds to the SCWS high-radiation signal listed in Table 3.9-3 to mitigate a release of radioactivity.

3.9.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.9-1 contains the ITAAC for radiation monitoring - shared-systems.

Table 3.9-1: Radiation Monitoring - Shared-Systems Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.09.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The CRVS automatically responds to the CRVS high-radiation signals upstream of the CRVS filter unit listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the CRVS high-radiation signals listed in Table 3.9-3.	Upon initiation of the real or simulated CRVS high-radiation signals upstream of the CRVS filter unit listed in Table 3.9-3, the CRVS automatically aligns/actuates the identified components to the positions identified in the table.
02.	The CRVS and the CRHS automatically respond to the CRVS high-radiation signals downstream of the CRVS filter unit listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the CRVS high-radiation signals listed in Table 3.9-3.	Upon initiation of the real or simulated CRVS high-radiation signals downstream of the CRVS filter unit listed in Table 3.9-3, the CRVS and the CRHS automatically align/actuate the identified components to the positions identified in the table.
03.	The RBVS automatically responds to the RBVS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the RBVS high-radiation signals listed in Table 3.9-3.	Upon initiation of the real or simulated RBVS high-radiation signals listed in Table 3.9-3, the RBVS automatically aligns/actuates the identified components to the positions identified in the table.
04.	The GRWS automatically responds to the GRWS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the GRWS high-radiation signals listed in Table 3.9-3.	Upon initiation of the real or simulated GRWS high-radiation signals listed in Table 3.9-3, the GRWS automatically aligns/actuates the identified components to the positions identified in the table.
05.	The CFDS automatically responds to the CFDS high-radiation signal listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the CFDS high-radiation signal listed in Table 3.9-3.	Upon initiation of a real or simulated CFDS high-radiation signal listed in Table 3.9-3, the CFDS automatically aligns/actuates the identified components to the positions identified in the table.
06.	The BPDS automatically responds to the BPDS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the BPDS high-radiation signals listed in Table 3.9-3.	Upon initiation of the real or simulated BPDS high-radiation signals listed in Table 3.9-3, the BPDS automatically aligns/actuates the identified components to the positions identified in the table.
07.	The LRWS automatically responds to the LRWS and radiation monitor high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the LRWS and radiation monitor high-radiation signals listed in Table 3.9-3.	Upon initiation of the real or simulated LRWS and radiation monitor high-radiation signals listed in Table 3.9-3, the LRWS automatically aligns/actuates the identified components to the positions identified in the table.
08.	The ABS automatically responds to the ABS high-radiation signals listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the ABS high-radiation signals listed in Table 3.9-3.	Upon initiation of the real or simulated ABS high-radiation signals listed in Table 3.9-3, the ABS automatically aligns/actuates the identified components to the positions identified in the table.

Table 3.9-1: Radiation Monitoring - Shared-Systems Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.09.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
09.	The DWS automatically responds to the DWS high-radiation signal listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the DWS high-radiation signal listed in Table 3.9-3.	Upon initiation of the real or simulated DWS high-radiation signal listed in Table 3.9-3, the DWS automatically aligns/actuates the identified components to the positions identified in the table.
10.	The RWDS automatically responds to the RWDS high-radiation signal listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the RWDS high-radiation signal listed in Table 3.9-3.	Upon initiation of the real or simulated RWDS high-radiation signal listed in Table 3.9-3, the RWDS automatically aligns/actuates the identified component to the positions identified in the table.
11.	The SCWS automatically responds to the SCWS high-radiation signal listed in Table 3.9-3 to mitigate a release of radioactivity.	A test will be performed of the SCWS high-radiation signal listed in Table 3.9-3.	Upon initiation of the real or simulated SCWS high-radiation signal listed in Table 3.9-3, the SCWS automatically aligns/actuates the identified component to the positions identified in the table.

Table 3.9-2: Radiation Monitoring - Shared-Systems Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[(1)]

ITAAC No.	Discussion
03.09.01	FSAR Section 11.5 discusses the CRVS radiation monitoring instrumentation and automatic response to high radiation conditions. FSAR Section 9.4.1 describes the CRVS. For each CRVS high radiation signal listed in Table 3.9-3, the CRVS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with FSAR Table 14.2-16, a preoperational test demonstrates the CRVS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated CRVS high radiation signal from 00-CRV-RIT-1003 and 00-CRV-RIT-1004.
03.09.02	FSAR Section 11.5 discusses the CRVS radiation monitoring instrumentation and automatic response to high radiation conditions, including alignment to the CRHS. The CRVS and CRHS are described in FSAR Section 9.4.1 and FSAR Section 6.4, respectively. For each CRVS high radiation signal listed in Table 3.9-3, the CRVS and the CRHS automatically align the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with FSAR Table 14.2-15, a preoperational test demonstrates the CRVS and the CRHS automatically align the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated CRVS high radiation signal from 00-CRV-RIT-1010-1 and 00-CRV-RIT-1011-2.
03.09.03	FSAR Section 11.5 discusses the RBVS radiation monitoring instrumentation and automatic response to high radiation conditions. FSAR Section 9.4.2 describes the RBV. For each RBVS high radiation signal listed in Table 3.9-3, the RBVS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with FSAR Table 14.2-17, a preoperational test demonstrates the RBVS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated RBVS high radiation signal from 00-RBV-RIT-1011 and 00-RBV-RIT-1164.
03.09.04	FSAR Section 11.5 discusses the GRWS radiation monitoring instrumentation and automatic response to high radiation conditions. FSAR Section 11.3 describes the GRWS. For each GRWS high radiation signal listed in Table 3.9-3, the GRWS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with FSAR Table 14.2-31, a preoperational test demonstrates the GRWS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated GRWS high radiation signal from 00-GRW-RIT-1021A, GRW-RIT-1021B, and GRW-RIT-1026.
03.09.05	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each CFDS high radiation signal listed in Table 3.9-3, the CFDS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with the information presented in FSAR Table 14.2-37, a preoperational test demonstrates the CFDS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated CFDS high radiation signal from 00-CFD-RT-1007.
03.09.06	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each BPDS high radiation signal listed in Table 3.9-3, the BPDS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with the information presented in FSAR Table 14.2-21, a preoperational test demonstrates the BPDS automatically aligns the components identified in Table 3.9-3 the required positions identified in the table upon initiation of a real or simulated BPDS high radiation signal from 00-BPD-RIT-1010, 00-BPD-RIT-1001, and 00-BPD-RIT-1034.

Table 3.9-2: Radiation Monitoring - Shared-Systems Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
03.09.07	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each LRWS and ARM high radiation signal listed in Table 3.9-3, the LRWS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with FSAR Table 14.2-30, a preoperational test demonstrates the LRWS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated LRWS or radiation monitor high radiation signal from 00-LRW-RIT-1021, 00-LRW-RIT-1022, and 00-RM-SKD-RW070-02.
03.09.08	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each ABS high radiation signal listed in Table 3.9-3, the ABS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with FSAR Table 14.2-6, a preoperational test demonstrates the ABS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated ABS high radiation signal from 00-AB-RT-1010, 00-AB-RT-1011, 00-AB-RT-1012, 00-AB-RT-1014, and 00-AB-RT-1015.
03.09.09	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each DWS high radiation signal listed in Table 3.9-3, the DWS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with the information presented in FSAR Table 14.2-11, a preoperational test demonstrates the DWS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated DWS high radiation signal from 00-DW-RT-1011 and 00-DW-RT-1012.
03.09.10	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each RWDS high radiation signal listed in Table 3.9-3, the RWDS automatically aligns the component identified in Table 3.9-3 to the required position identified in the table. In accordance with the information presented in FSAR Table 14.2-20, a preoperational test demonstrates the RWDS automatically aligns the component identified in Table 3.9-3 to the required position identified in the table upon initiation of a real or simulated RWDS high radiation signal from 00-RWD-RIT-1041.
03.09.11	FSAR Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions describes the system responses to high radiation conditions. For each SCWS high radiation signal listed in Table 3.9-3, the SCWS automatically aligns the components identified in Table 3.9-3 to the required positions identified in the table. In accordance with the information presented in FSAR Table 14.2-8, a preoperational test demonstrates the SCWS automatically aligns the component identified in Table 3.9-3 to the required positions identified in the table upon initiation of a real or simulated SCWS high radiation signal from 00-SCW-RIT-1096.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.9-3: Radiation Monitoring - Shared-Systems Automatic Actions

Radiation Monitoring ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
00-CRV-RIT-1003 00-CRV-RIT-1004	CRVS outside air upstream of CRVS filter unit	1. CRVS filter unit bypass damper 2. CRVS filter unit bypass damper 3. CRVS filter unit inlet isolation damper 4. CRVS filter unit outlet isolation damper 5. CRVS filtration unit	1. 00-CRV-MBD-0110 2. 00-CRV-MBD-0111 3. 00-CRV-MBD-0003 4. 00-CRV-MBD-0004 5. 00-CRV-FLT-0067	1. Close 2. Close 3. Open 4. Open 5. Start
00-CRV-RIT-1010-1 00-CRV-RIT-1011-2	CRVS outside air downstream of CRVS filter unit	1. CRVS filtration unit 2. CRVS filtration unit fan 3. CRVS filter unit bypass damper 4. CRVS filter unit bypass damper 5. CRVS filter unit inlet isolation damper 6. CRVS filter unit outlet isolation damper 7. CRVS control room envelope supply damper 8. CRVS control room envelope supply damper 9. CRVS control room envelope return damper 10. CRVS control room envelope return damper 11. CRVS control room envelope exhaust damper 12. CRVS control room envelope exhaust damper 13. CRVS general exhaust fan 14. CRVS battery exhaust fan A 15. CRVS battery exhaust fan B 16. CRHS air supply isolation valve 17. CRHS air supply isolation valve 19. CRHS pressure relief vent valve 19. CRHS pressure relief vent valve 20. CRVS supply air handling units	1. 00-CRV-FLT-0067 2. 00-CRV-FAN-0067 3. 00-CRV-MBD-0110 4. 00-CRV-MBD-0111 5. 00-CRV-MBD-0003 6. 00-CRV-MBD-0004 7. 00-CRV-MBD-0026 8. 00-CRV-MBD-0027 9. 00-CRV-MBD-0092 10. 00-CRV-MBD-0093 11. 00-CRV-MBD-0038 12. 00-CRV-MBD-0039 13. 00-CRV-FAN-0055 14. 00-CRV-FAN-0084A 15. 00-CRV-FAN-0084B 16. 00-CRH-SV-0007A 17. 00-CRH-SV-0007B 18. 00-CRH-SV-0028A 19. 00-CRH-SV-0028B 20. 00-CRV-AHU-0010A/B	1. Stop 2. Stop 3. Close 4. Close 5. Close 6. Close 7. Close 8. Close 9. Close 10. Close 11. Close 12. Close 13. Stop 14. Stop 15. Stop 16. Open 17. Open 18. Open 19. Open 20. Stop
00-RBV-RIT-1011 00-RBV-RIT-1164	RBVS SFP exhaust filter inlet	1. RBVS module 01-03 exhaust isolation damper 2. RBVS module 04-06 exhaust isolation damper 3. RBVS dry dock exhaust isolation damper 4. RBVS FLT-0044A bypass bubble tight isolation damper 5. RBVS FLT-0044A inlet isolation damper 6. RBVS FLT-0044A internal isolation damper 7. RBVS FLT-0044B bypass bubble tight isolation damper 8. RBVS FLT-0044B inlet isolation damper 9. RBVS FLT-0044B internal isolation damper	1. 00-RBV-MBD-0316 2. 00-RBV-MBD-0315 3. 00-RBV-MBD-0336 4. 00-RBV-MBD-0043A 5. 00-RBV-MBD-0042A 6. 00-RBV-MBD-0045A 7. 00-RBV-MBD-0043B 8. 00-RBV-MBD-0042B 9. 00-RBV-MBD-0045B	1. Close 2. Close 3. Close 4. Close 5. Open 6. Open 7. Close 8. Open 9. Open

Table 3.9-3: Radiation Monitoring - Shared-Systems Automatic Actions (Continued)

Radiation Monitoring ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
00-GRW-RIT-1021A	GRWS charcoal decay bed skid A outlet	1. GRWS charcoal decay bed skid A outlet isolation valve 2. GRWS charcoal decay bed skid A inlet isolation valve 3. GRWS to Radioactive Waste Building HVAC system (RWBVS) exhaust upstream isolation valve 4. GRWS to RWBVS exhaust downstream isolation valve	1. 00-GRW-AOV-0019A 2. 00-GRW-AOV-0014A 3. 00-GRW-AOV-0023 4. 00-GRW-AOV-0024	1. Close 2. Close 3. Close 4. Close
00-GRW-RIT-1021B	GRWS charcoal decay bed skid B outlet	1. GRWS charcoal decay bed skid B outlet isolation valve 2. GRWS charcoal decay bed skid B inlet isolation valve 3. GRWS to RWBVS exhaust upstream isolation valve 4. GRWS to RWBVS exhaust downstream isolation valve	1. 00-GRW-AOV-0019B 2. 00-GRW-AOV-0014B 3. 00-GRW-AOV-0023 4. 00-GRW-AOV-0024	1. Close 2. Close 3. Close 4. Close
00-GRW-RIT-1026	GRWS effluent to RBVS	1. GRWS to RWBVS exhaust upstream isolation valve 2. GRWS to RWBVS exhaust downstream isolation valve	1. 00-GRW-AOV-0023 2. 00-GRW-AOV-0024	1. Close 2. Close
00-CFD-RIT-1007	CFDS containment drain separator gaseous discharge to RBVS	1. CFDS pump A 2. CFDS pump B 3. CFDS drain separator gas discharge to RBVS isolation valve 4. CFDS drain separator water discharge to PCWS flow control valve	1. 00-CFD-P-0004A 2. 00-CFD-P-0004B 3. 00-CFD-AOV-0017 4. 00-CFD-AOV-0018	1. Stop 2. Stop 3. Close 4. Close
00-BPD-RIT-1010	Condensate resin regeneration skid waste effluent	1. Chemical waste collection tank sump pump A 2. Chemical waste collection tank sump pump B 3. Chemical waste collection tank to BPDS collection tank flow control valve 4. Chemical waste collection tank to LRWS isolation valve	1. 00-BPD-P-0027A 2. 00-BPD-P-0027B 3. 00-BPD-AOV-0030 4. 00-BPD-AOV-0051	1. Stop 2. Stop 3. Close 4. Close
1. 00-BPD-RIT-1001 2. 00-BPD-RIT-1034	BPDS Turbine Generator Building floor drains BPDS auxiliary boiler blowdown	1. Wastewater collection tank sump pump A 2. Wastewater collection tank sump pump B 3. Wastewater collection tank sump to BPDS collection tank flow control valve 4. Wastewater collection tank sump to LRWS isolation valve 5. Wastewater collection tank firewater removal pump	1. 00-BPD-P-0002A 2. 00-BPD-P-0002B 3. 00-BPD-AOV-0009 4. 00-BPD-AOV-0050 5. 00-BPD-P-0003	1. Stop 2. Stop 3. Close 4. Close 5. Stop
00-LRW-RIT-1021 00-LRW-RIT-1022	LRWS discharge to utility water system (UWS)	1. LRWS to UWS isolation valve 2. LRWS to UWS isolation valve	1. 00-LRW-AOV-0059 2. 00-LRW-AOV-0060	1. Close 2. Close
00-RM-SKD-RW070-02 (radiation monitor)	GRWS cubicle area	LRWS to GRWS discharge isolation valve	00-LRW-AOV-0083	Close

Table 3.9-3: Radiation Monitoring - Shared-Systems Automatic Actions (Continued)

Radiation Monitoring ID(s)	Variable Monitored	Actuated Component(s)	Component ID(s)	Component Action(s)
1. 00-AB-RT-1010 2. 00-AB-RT-1011 3. 00-AB-RT-1012 4. 00-AB-RT-1014 5. 00-AB-RT-1015	1. Auxiliary boiler skid vent 2. Auxiliary boiler skid and ABS superheater skid drain 3. ABS superheater skid inlet vent 4. ABS header drain 5. Turbine Generator Building auxiliary steam header	1. Auxiliary boiler to ABS superheater skid isolation valve 2. ABS superheater skid outlet isolation valve	1. 00-AB-AOV-0110 2. 00-AB-AOV-0111	1. Close 2. Close
00-DW-RT-1011	North RXB demineralized water distribution header	North RXB demineralized water distribution header isolation valve	00-DW-AOV-0023	Close
00-DW-RT-1012	South RXB demineralized water distribution header	South RXB demineralized water distribution header isolation valve	00-DW-AOV-0024	Close
00-RWD-RIT-1041	Reactor component cooling (RCCW) water drain tank	RWDS to RCCWS expansion tank isolation valve	00-RWD-AOV-0126	Close
00-SCW-RIT-1096	SCWS blowdown line	SCWS blowdown isolation valve	00-SCW-AOV-0167	Close

Table 3.9-4: Radiation Monitoring - Shared-Systems Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.09.01			X			
03.09.02			X			
03.09.03			X			
03.09.04			X			
03.09.05			X			
03.09.06			X			
03.09.07			X			
03.09.08			X			
03.09.09			X			
03.09.10			X			
03.09.11			X			

3.10 Overhead Heavy Load Handling System

3.10.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

The scope of this section is the Overhead Heavy Load Handling System (OHLHS), which is described in FSAR Section 9.1.5. The OHLHS supports up to six NPMs.

The OHLHS performs the following risk-significant system functions that are verified by ITAAC:

- The RBC main hoist and lower block assembly support the NPM by providing structural support and mobility while moving from refueling, inspection and operating bay.

The OHLHS performs the following nonsafety-related system function that is verified by ITAAC. The OHLHS supports the RXB by providing movement for lifts using overhead handling equipment performed in an area where uncontrolled motion of lifted load can result in the loss of an essential safety function.

<u>Design Commitments</u>

- The OHLHS equipment listed in Table 3.10-3 is single-failure-proof in accordance with the approved design.
- The OHLHS equipment listed in Table 3.10-3 is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.
- All weld joints of the OHLHS equipment listed in Table 3.10-3 whose failure could result in the drop of a critical load comply with the ASME NOG-1 Code or ASME NUM-1 Code listed in Table 3.10-3.

3.10.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.10-1 contains the ITAAC for the OHLHS.

Table 3.10-1: Overhead Heavy Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.10.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The OHLHS equipment listed in Table 3.10-3 is single-failure-proof in accordance with the approved design.	An inspection will be performed of the as-built OHLHS equipment listed in Table 3.10-3.	A report exists and concludes that the OHLHS equipment listed in Table 3.10-3 is single-failure-proof in accordance with the approved design.
02.	The OHLHS equipment listed in Table 3.10-3 is capable of lifting and supporting its rated load, holding the rated load, and transporting the rated load.	A rated load test will be performed of the OHLHS equipment listed in Table 3.10-3.	The OHLHS equipment listed in Table 3.10-3 lifts, supports, holds with the brakes, and transports a load of at least 125 percent of the manufacturer's rated capacity
03.	All weld joints of the OHLHS equipment listed in Table 3.10-3 whose failure could result in the drop of a critical load comply with the ASME NOG-1 Code or ASME NUM-1 Code listed in Table 3.10-3.	An inspection will be performed of the OHLHS equipment listed in Table 3.10-3 as-built weld joints whose failure could result in the drop of a critical load.	The results of the non-destructive examination of the OHLHS equipment listed in Table 3.10-3 weld joints whose failure could result in the drop of a critical load comply with the ASME NOG-1 Code or ASME NUM-1 Code listed in Table 3.10-3.

Table 3.10-2: Overhead Heavy Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
03.10.01	FSAR Section 9.1.5, Overhead Heavy Load Handling Systems, describes that the OHLHS equipment listed in Table 3.10-3 is classified as a Type I equipment as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) and ASME NUM-1 Code, Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type). An ITAAC inspection is performed of the OHLHS equipment listed in Table 3.10-3 to verify the existence of the single-failure-proof features required by the ASME NOG-1 Code or ASME NUM-1 Code listed for each OHLHS equipment listed in Table 3.10-3. This ITAAC inspection may be performed any time after manufacture of the OHLHS equipment listed in Table 3.10-3 (at the factory or later).
03.10.02	FSAR Section 9.1.5, Overhead Heavy Load Handling Systems, describes that the OHLHS equipment listed in Table 3.10-3 is classified as a Type I equipment as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) and ASME NUM-1 Code, Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type). In accordance with ASME NOG-1 and ASME NUM-1, each OHLHS equipment listed in Table 3.10-3 is full load tested at a maximum of 100% of the manufacturer's rating. After the full load test is completed, and prior to use of the equipment to handle loads, each OHLHS equipment listed in Table 3.10-3 is rated load tested at 125% (+5%, -0%) of the manufacturer's rating in accordance with the ASME NOG-1 Code or ASME NUM-1 Code listed for each OHLHS equipment listed in Table 3.10-3. A site acceptance test demonstrates that the OHLHS equipment listed in Table 3.10-3 is rated load tested at 125% (+5%, -0%) of the manufacturer's rating.
03.10.03	FSAR Section 9.1.5, Overhead Heavy Load Handling Systems, describes that the OHLHS equipment listed in Table 3.10-3 is classified as a Type I equipment as defined by the ASME NOG-1 Code, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) and ASME NUM-1 Code, Rules for Construction of Cranes, Monorails, and Hoists (With Bridge or Trolley or Hoist of the Underhung Type). An ITAAC inspection is performed to verify that the ASME Type I OHLHS equipment listed in Table 3.10-3 as-built welds are nondestructively examined in accordance with the standards of ASME NOG-1, ASME NUM-1, and the purchase specifications. This ITAAC inspection may be performed any time after manufacture of the OHLHS equipment listed in Table 3.10-3 (at the factory or later).

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.10-3: Overhead Heavy Load Handling System Equipment

Equipment Identifier	Equipment Description	Design Code
00-RBC-CRN-0001	RBC main hoist and lower block assembly	ASME NOG-1, Type I (Note 1)
	RBC sister hook	ASME NOG-1, Type I
	RBC auxiliary hoists (2 total)	ASME NUM-1, Type IA
00-RBC-CRN-0006	Articulating traveling jib crane	ASME NUM-1, Type IA
00-RBC-CRN-0007	Dry dock jib crane	ASME NUM-1, Type IA
00-RBC-MHE-0002	Auxiliary wet hoist	ASME NUM-1, Type IA
00-RBC-CRN-0008	Module access platform jib crane	ASME NUM-1, Type IA
00-RBC-CRN-0005	Traveling jib crane hoist	ASME NUM-1, Type IA

Note 1: Refer to FSAR Table 9.1.5-1 for exceptions.

Table 3.10-4: Overhead Heavy Load Handling System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal/ External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.10.01				X		
03.10.02				X		
03.10.03				X		

3.11 Reactor Building

3.11.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the RXB, which is described in FSAR Section 3.8.4. The RXB supports up to six NPMs.

The RXB performs the following safety-related system function that is verified by ITAAC:

- The RXB supports the following systems by housing and providing structural support:
  - NPM
  - CVCS
  - UHS
  - MPS
  - NMS

The RXB performs the following nonsafety-related, risk-significant system function that is verified by ITAAC:

- The RXB supports the OHLHS by housing and providing structural support.

Design Commitments

- Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the RXB fire area of origin.
- Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the RXB flooding area of origin.
- The Seismic Category I RXB is protected against external flooding in order to prevent flooding of safety-related SSC within the structure.
- The RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding.
- The RXB is Seismic Category I and maintains its structural integrity under the design basis loads.
- Non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE.
- Safety-related SSC are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.

3.11.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.11-1 contains the ITAAC for the RXB.

Table 3.11-1: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.11.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the RXB fire area of origin.	An inspection will be performed of the RXB as-built fire and smoke barriers.	The following RXB fire and smoke barriers exist in accordance with the FHA, and have been qualified for the fire rating specified in the FHA: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • fire-rated walls, floors, and ceilings • smoke barriers
02.	Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the RXB flooding area of origin.	An inspection will be performed of the RXB as-built internal flooding barriers.	The following RXB internal flooding barriers exist in accordance with the internal flooding analysis report and have been qualified as specified in the internal flooding analysis report: • flood resistant doors • curbs and sills • walls • water tight penetration seals • National Electrical Manufacturer's Association enclosures
03.	The Seismic Category I RXB is protected against external flooding in order to prevent flooding of safety-related SSC within the structure.	An inspection will be performed of the RXB as-built floor elevation at ground entrances.	The RXB floor elevation at ground entrances is higher than the maximum external flood elevation.
04.	The RXB includes radiation shielding barriers for normal operation and post-accident radiation shielding.	An inspection and analysis will be performed of the as-built RXB radiation shielding barriers.	A report exists and concludes the radiation attenuation capability of RXB radiation shielding barriers is greater than or equal to the required attenuation capability of the approved design.
05.	Not Used		
06.	The RXB is Seismic Category I and maintains its structural integrity under the design basis loads.	A reconciliation analysis will be performed of the as-built RXB under the actual design basis loads.	A design summary report exists and concludes that the as-built RXB maintains its structural integrity in accordance with the approved design under the actual design basis loads, and the in-structure responses for the as-built RXB are enveloped by those in the approved design.

Table 3.11-1: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.11.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
07.	Non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE.	An inspection and analysis will be performed of the as-built non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB.	A report exists and concludes that the non-Seismic Category I SSC located where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE as demonstrated by one or more of the following criteria: • Seismic Category I SSC are isolated from non-Seismic Category I SSC, so that interaction does not occur. • Seismic Category I SSC are analyzed to confirm that the ability to perform their safety functions is not impaired as a result of impact from non- Seismic Category I SSC. • A non-Seismic Category I restraint system designed to Seismic Category I requirements is used to ensure that no interaction occurs between Seismic Category I SSC and non-Seismic Category I SSC.
08.	Safety-related SSC are protected against the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.	An inspection and analysis will be performed of the as-built high- and moderate-energy piping systems and protective features for the safety-related SSC located in the RXB outside the Reactor Pool Bay.	Protective features are installed in accordance with the as-built Pipe Break Hazard Analysis Report and safety-related SSC are protected against or qualified to withstand the dynamic and environmental effects associated with postulated failures in high- and moderate-energy piping systems.

Table 3.11-2: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
03.11.01	FSAR Section 9.5.1, Fire Protection Program, discusses that fire and smoke barriers separate: (1) safety-related systems from any potential fires in nonsafety-related areas that could affect the ability of safety-related systems to perform their safety-related function, (2) redundant divisions or trains of safety-related systems from each other to prevent damage that could adversely affect a safe shutdown function from a single fire, (3) equipment within a single safety-related electrical division that present a fire hazard to equipment in another safety-related division, and (4) electrical circuits (safety-related and nonsafety-related) whose fire-induced failure could cause a spurious actuation that could adversely affect a safe shutdown function. An ITAAC inspection is performed to verify that the following RXB as-built fire barriers and smoke barriers are installed in accordance with the FHA and are qualified for the fire rating specified in the FHA: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • smoke barriers • fire-rated walls, floors, and ceilings The objective of the inspection is to verify that the fire and smoke barriers meet the design requirements, meet the location requirements, and are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.
03.11.02	FSAR Section 3.4.1, Internal Flood Protection for Onsite Equipment Failures, discusses the features used to mitigate the consequences of internal flooding, which include structural enclosures, barriers, curbs, sills, and watertight seals. An ITAAC inspection is performed to verify that the following RXB as-built internal flooding barriers are installed in accordance with the internal flooding analysis report and are qualified as specified in the internal flooding analysis report: • flood resistant doors • curbs and sills • walls • water tight penetration seals • NEMA enclosures The objective of the inspection is to verify that the flooding barriers meet the design requirements, meet the location requirements, and are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.
03.11.03	FSAR Section 2.4.2, Floods, discusses that the maximum flood elevation (including wind-induced wave run-up) is one foot below baseline plant elevation. FSAR Section 3.4.2.1, Probable Maximum Flood, states that the design basis flood level (including wave action) is one foot below the baseline top of concrete elevation at the ground level floor. An ITAAC inspection is performed to verify that the RXB as-built floor elevation at ground entrances is located above the maximum external flood elevation to protect the RXB from external flooding. The inspection will compare the maximum external flood elevation against the RXB as-built design drawings to verify that the floor elevation at ground entrances is a minimum of one foot above the maximum external flood elevation.

Table 3.11-2: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[(1)] (Continued)

ITAAC No.	Discussion
03.11.04	FSAR Section 12.3, Radiation Protection Design Features, provides the design bases for radiation shielding, including type, form and material properties utilized in specific locations. Radiation shielding is provided to meet the radiation zone and access requirements for normal operation and post-accident conditions, and to demonstrate compliance with 10 CFR 50.49, GDC 4, PDC 19, GDC 61, 10 CFR 50.34(f)(2)(vii), and other relevant requirements. Compartment walls, ceilings, and floors, or other barriers provide shielding. An ITAAC inspection is performed of the RXB radiation barriers to verify wall materials and thicknesses. The required thicknesses are specified in FSAR Table 12.3-5. Attenuation capabilities are determined based on wall materials and thicknesses, and an analysis and report will conclude that attenuation capabilities are greater than or equal to the approved design.
03.11.05	Not Used
03.11.06	FSAR Section 3.8.4 and Appendix 3B provide descriptive information, including plans and sections of each Seismic Category I structure, to establish that there is sufficient information to define the primary structural aspects and elements relied upon for the structure to perform the intended safety functions. Critical dimensions are identified in FSAR Appendix 3B. The RXB and its design basis loads are discussed in FSAR Section 3.8.4. Critical sections are the subcomponents of individual Seismic Category I structures (i.e., shear walls, floor slabs and roofs, structure-to-structure connections) that are analytically representative of an essentially complete design. Design basis load combinations are shown in FSAR Tables 3.8.4-1 and 3.8.4-2. A reconciliation analysis of the as-built RXB is performed to ensure the RXB maintains its structural integrity in accordance with the approved design under the actual design basis loads, and the in-structure responses for the RXB are enveloped by those in the approved design. The design summary report provides criteria for the reconciliation between design and as-built conditions, as described in FSAR Section 3.8.4.
03.11.07	FSAR Section 3.2.1, Seismic Classification, discusses that per RG 1.29, some SSC that perform no safety-related functions could, if they failed under seismic loading, prevent or reduce the functioning of Seismic Category I SSC. An ITAAC inspection and analysis is performed to verify that the as-built non-Seismic Category I SSC where there is a potential for adverse interaction with the RXB or a Seismic Category I SSC in the RXB will not impair the ability of Seismic Category I SSC to perform their safety functions as demonstrated by one or more of the following criteria: i. Seismic Category I SSC are isolated from non-Seismic Category I SSC so that interaction does not occur. ii. Seismic Category I SSC are analyzed to confirm that the ability to perform their safety functions is not impaired as a result of impact from non-Seismic Category I SSC. A non-Seismic Category I restraint system designed to Seismic Category I requirements is used to ensure that no interaction occurs between Seismic Category I SSC and non-Seismic Category I SSC.

Table 3.11-2: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1] (Continued)

ITAAC No.	Discussion
03.11.08	FSAR Section 3.6, Protection against Dynamic Effects Associated with Postulated Rupture of Piping, provides the design bases and criteria for the analysis required to demonstrate that safety-related SSC are not impacted by the adverse effects of a high-and moderate-energy pipe failure within the plant. An ITAAC inspection is performed to verify that the as-built protective features located in the RXB outside the reactor pool bay credited in the reconciled Pipe Break Hazards Analysis Report (such as pipe whip restraints, pipe whip or jet impingement barriers, jet impingement shields, or guard pipe) have been installed in accordance with design drawings of sufficient detail to show the existence and location of the protective hardware. The as-built inspection is intended to verify that changes to postulated pipe failure locations and protective features or protected equipment made during construction do not adversely affect the safety-related functions of the protected equipment.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.11-3: Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.11.01		X			X	
03.11.02		X				
03.11.03		X				
03.11.04		X				
03.11.05						
03.11.06	X					
03.11.07	X					
03.11.08	X	X				

3.12 Radioactive Waste Building

3.12.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the RWB, which is described in FSAR Section 3.8.4. The RWB supports up to six NPMs.

Design Commitments

- The RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding.
- The RWB includes the radiation shield doors listed in FSAR Table 12.3-7 for normal operation and for post-accident radiation shielding. These doors have a radiation shielding capability that meets or exceeds the approved design.
- The below grade portions of the RWB and above grade portions used for storage or processing of radioactive waste is an RW-IIa structure and maintains its structural integrity under the design basis loads.

3.12.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.12-1 contains the ITAAC for the RWB.

Table 3.12-1: Radioactive Waste Building Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.12.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding.	An inspection and analysis will be performed of the as-built RWB radiation shielding barriers.	A report exists and concludes the radiation attenuation capability of RWB radiation shielding barriers is greater than or equal to the required attenuation capability of the approved design.
02.	The RWB includes the radiation shield doors listed in FSAR Table 12.3-7 for normal operation and for post-accident radiation shielding. These doors have a radiation shielding capability that meets or exceeds the approved design.	An inspection and analysis will be performed of the as-built RWB radiation shield doors listed in FSAR Table 12.3-7.	A report exists and concludes each RWB radiation shield door is installed in its design location and has a radiation shielding capability that meets or exceeds its door shielding value in FSAR Table 12.3-7.
03.	The below grade portions of the RWB and above grade portions used for storage or processing of radioactive waste is an RW-IIa structure and maintains its structural integrity under the design basis loads.	A reconciliation analysis will be performed for the RW-IIa portions of the as-built RWB under the actual design basis loads.	A design summary report exists and concludes that the RW-IIa portions of the as-built RWB maintain structural integrity in accordance with the approved design under the design basis loads.

Table 3.12-2: Radioactive Waste Building Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
03.12.01	FSAR Section 12.3, Radiation Protection Design Features, provides the design bases for radiation shielding, including type, form and material properties utilized in specific locations. Radiation shielding is provided to meet the radiation zone requirement for normal operation and post-accident conditions and to demonstrate conformance with GDC 61, RG 4.21, RG 8.8, and other relevant requirements. Compartment walls, ceilings, and floors, or other barriers provide shielding. An ITAAC inspection is performed of the RWB radiation barriers to verify wall materials and thicknesses. The required thicknesses are specified in FSAR Table 12.3-6. Attenuation capabilities are determined based on wall materials and thicknesses, and an analysis and report will conclude that attenuation capabilities are greater than or equal to the approved design.
03.12.02	FSAR Section 12.3 provides the design bases for radiation shielding. Radiation shielding is provided to meet the radiation zone requirements for normal operation and post-accident conditions, and to demonstrate conformance to RG 4.21 and RG 8.8. FSAR Table 12.3-7 lists the RWB radiation shield doors and identifies the radiation shielding capability of each door as a percentage of the effective shielding provided by the wall in which the door is located. An ITAAC inspection is performed to verify the RWB radiation shield doors listed in FSAR Table 12.3-7 are installed in their design location. An analysis and report will conclude each RWB radiation shield door has a radiation shielding capability that meets or exceeds its door shielding value in FSAR Table 12.3-7.
03.12.03	The RW-IIa RWB and its design basis loads are discussed in FSAR Section 3.8.4. Design basis loads for RW-IIa structures are listed in RG 1.143. A reconciliation analysis of the RW-IIa portions of the RWB is performed to ensure that deviations between the drawings used for construction and the as-built RW-IIa RWB are reconciled and the RW-IIa RWB maintains its structural integrity under the design basis loads in accordance with the approved design under the actual design basis loads.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.12-3: Radioactive Waste Building Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.12.01			X			
03.12.02			X			
03.12.03	X		X			

3.13 Control Building

3.13.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the CRB, which is described in FSAR Section 3.8.4. The CRB supports up to six NPMs.

The CRB performs the following safety-related system function that is verified by ITAAC:

- The CRB supports the MPS by housing and providing structural support.

The CRB performs the following nonsafety-related system function that is verified by ITAAC:

- The CRB supports the CRVS by providing a portion of the CRE.

Design Commitments

- Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the CRB fire area of origin.
- Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the CRB flooding area of origin.
- The Seismic Category I CRB is protected against external flooding in order to prevent flooding of safety-related SSC within the structure.
- The CRB area housing the MCR and associated facilities is Seismic Category I and maintains its structural integrity under the design basis loads.
- Non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE.

3.13.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.13-1 contains the ITAAC for the CRB.

Table 3.13-1: Control Building Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.13.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	Fire and smoke barriers provide confinement so that the impact from internal fires, smoke, hot gases, or fire suppressants is contained within the CRB fire area of origin.	An inspection will be performed of the CRB as-built fire and smoke barriers.	The following CRB fire and smoke barriers exist in accordance with the FHA, and have been qualified for the fire rating specified in the FHA: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • fire-rated walls, floors, and ceilings • smoke barriers
02.	Internal flooding barriers provide confinement so that the impact from internal flooding is contained within the CRB flooding area of origin.	An inspection will be performed of the CRB as-built internal flooding barriers.	The following CRB internal flooding barriers exist in accordance with the internal flooding analysis report and have been qualified as specified in the internal flooding analysis report: • flood resistant doors • walls • water tight penetration seals • National Electrical Manufacturer's Association (NEMA) enclosures
03.	The Seismic Category I CRB is protected against external flooding in order to prevent flooding of safety-related SSC within the structure.	An inspection will be performed of the CRB as-built floor elevation at ground entrances.	The CRB floor elevation at ground entrances is higher than the maximum external flood elevation.

Table 3.13-1: Control Building Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.13.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
04.	The CRB area housing the MCR and associated facilities is Seismic Category I and maintains its structural integrity under the design basis loads.	A reconciliation analysis will be performed of the as-built CRB area housing the MCR and associated facilities under the actual design basis loads.	A design summary report exists and concludes that (1) the as-built CRB area housing the MCR and associated facilities maintains its structural integrity in accordance with the approved design under the actual design basis loads, and (2) the in-structure responses for the as-built CRB area housing the MCR and associated facilities are enveloped by those in the approved design.
05.	Non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE.	An inspection and analysis will be performed of the as-built non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB.	A report exists and concludes that the non-Seismic Category I SSC located where there is a potential for adverse interaction with the CRB or a Seismic Category I SSC in the CRB will not impair the ability of Seismic Category I SSC to perform their safety functions during or following an SSE as demonstrated by one or more of the following criteria: • Seismic Category I SSC are isolated from non-Seismic Category I SSC, so that interaction does not occur. • Seismic Category I SSC are analyzed to confirm that the ability to perform their safety functions is not impaired as a result of impact from non-Seismic Category I SSC. • A non-Seismic Category I restraint system designed to Seismic Category I requirements is used to ensure that no interaction occurs between Seismic Category I SSC and non-Seismic Category I SSC.

Table 3.13-2: Control Building Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
03.13.01	FSAR Section 9.5.1, Fire Protection Program, discusses that fire and smoke barriers separate: (1) safety-related systems from any potential fires in nonsafety-related areas that could affect the ability of safety-related systems to perform their safety-related function., (2) redundant divisions or trains of safety-related systems from each other to prevent damage that could adversely affect a safe shutdown function from a single fire., (3) equipment within a single safety-related electrical division that present a fire hazard to equipment in another safety-related division, and (4) electrical circuits (safety-related and nonsafety-related) whose fire-induced failure could cause a spurious actuation that could adversely affect a safe shutdown function. An ITAAC inspection is performed to verify that the following CRB as-built fire barriers and smoke barriers are installed in accordance with the FHA and are qualified for the fire rating specified in the FHA: • fire-rated doors • fire-rated penetration seals • fire-rated dampers • smoke barriers The objective of the inspection is to verify that the fire and smoke barriers meet the design requirements, meet the location requirements, and are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.
03.13.02	FSAR Section 3.4.1, Internal Flood Protection for Onsite Equipment Failures, discusses the features used to mitigate the consequences of internal flooding, which include structural enclosures, barriers, and watertight seals. An ITAAC inspection is performed to verify that the following CRB as-built internal flooding barriers are installed in accordance with the internal flooding analysis report and are qualified as specified in the internal flooding analysis report: • flood resistant doors • walls • water tight penetration seals • NEMA enclosures The objective of the inspection is to verify that the flooding barriers meet the design requirements, meet the location requirements, and are qualified for their intended use based upon visual inspection and review of the as-built drawings and qualification documentation.
03.13.03	FSAR Section 2.4.2, Floods, discusses that the maximum flood elevation (including wind-induced wave run-up) is one foot below baseline plant elevation. FSAR Section 3.4.2.1, Probable Maximum Flood, states that the design basis flood level (including wave action) is one foot below the baseline top of concrete elevation at the ground level floor. An ITAAC inspection is performed to verify that the CRB as-built floor elevation at ground entrances is located above the maximum external flood elevation to protect the CRB from external flooding. The inspection will compare the maximum external flood elevation against the CRB as-built design drawings to verify that the floor elevation at ground entrances is a minimum of one foot above the maximum external flood elevation.

Table 3.13-2: Control Building Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
03.13.04	FSAR Section 3.8.4 and FSAR Appendix 3B provide descriptive information, including plans and sections of each Seismic Category I structure, to establish that there is sufficient information to define the primary structural aspects and elements relied upon for the structure to perform the intended safety functions. Critical dimensions are identified in Appendix 3B. The CRB design basis loads are discussed in FSAR Section 3.8.4. Critical sections are the subcomponents of individual Seismic Category I structures (i.e., shear walls, floor slabs and roofs, structure-to-structure connections) that are analytically representative of an essentially complete design. Design basis loads load combinations are shown in FSAR Tables 3.8.4-1 and 3.8.4-2. A reconciliation analysis of the as-built Seismic Category I areas of the CRB is performed to ensure the CRB maintains its structural integrity in accordance with the approved design under the actual design basis loads, and the in-structure responses for the CRB are enveloped by those in the approved design. The design summary report provides criteria for the reconciliation between design and as-built conditions, as described in FSAR Section 3.8.4.
03.13.05	FSAR Section 3.2.1, Seismic Classification, discusses that per RG 1.29, some SSC that perform no safety-related functions could, if they failed under seismic loading, prevent or reduce the functioning of Seismic Category I SSC. An ITAAC inspection and analysis is performed to verify that the as-built non-Seismic Category I SSC located where a potential for adverse interaction with a Seismic Category I SSC exists will not impair the ability of Seismic Category I SSC to perform their safety functions as demonstrated by one or more of the following criteria: i. The collapse of the non-Seismic Category I structure will not cause the non-Seismic Category I structure to strike a Seismic Category I SSC. ii. The collapse of the non-Seismic Category I structure will not impair the integrity of Seismic Category I SSC, nor result in incapacitating injury to control room occupants. iii. The non-Seismic Category I structure will be analyzed and designed to prevent its failure under SSE conditions.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.13-3: Control Building Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.13.01		X			X	
03.13.02		X				
03.13.03		X				
03.13.04	X					
03.13.05	X					

3.14 Equipment Qualification - Shared Equipment

3.14.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is equipment qualification of equipment shared by NPMs 1 through 6.

This section applies to the safety-related RXB overpressurization vents, and a limited population of shared, nonsafety-related equipment that has augmented Seismic Category I or environmental qualification requirements. The nonsafety-related equipment in this section provides one of the following nonsafety-related functions:

- physical support of irradiated fuel (FHE, spent fuel storage racks, and RBC)
- a path for makeup water to the UHS
- containment of the UHS water
- monitors UHS water level

Additionally, this section applies to the nonsafety-related, RW-IIa components and piping used for processing gaseous radioactive waste.

Design Commitments

- The shared Seismic Category I equipment listed in Table 3.14-3, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after an SSE.
- The shared electrical equipment listed in Table 3.14-3 located in a harsh environment, including its connection assemblies, withstands the design basis harsh environmental conditions experienced during normal operations, AOO, DBA, and post-accident conditions, and performs its function for the period of time required to complete the function.
- The RW-IIa components used for processing gaseous radioactive waste listed in Table 3.14-3 are constructed to the standards of RW-IIa.

3.14.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.14-1 contains the ITAAC for equipment qualification - shared-equipment.

Table 3.14-1: Equipment Qualification - Shared Equipment Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.14.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The common Seismic Category I equipment listed in Table 3.14-3, including its associated supports and anchorages, withstands design basis seismic loads without loss of its function(s) during and after an SSE.	i. A type test, analysis, or a combination of type test and analysis will be performed of the common Seismic Category I equipment listed in Table 3.14-3, including its associated supports and anchorages.	i. A Seismic Qualification Report exists and concludes that the common Seismic Category I equipment listed in Table 3.14-3, including its associated supports and anchorages, will withstand the design basis seismic loads and perform its function during and after an SSE.
		ii. An inspection will be performed of the common Seismic Category I as- built equipment listed in Table 3.14-3, including its associated supports and anchorages.	ii. The common Seismic Category I equipment listed in Table 3.14-3, including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the equipment's Seismic Qualification Report.
02.	The common electrical equipment listed in Table 3.14-3 located in a harsh environment, including its connection assemblies, withstands the design basis harsh environmental conditions experienced during normal operations, AOOs, DBA, and post-accident conditions and performs its function for the period of time required to complete the function.	i. A type test or a combination of type test and analysis will be performed of the common electrical equipment listed in Table 3.14-3, including its connection assemblies.	i. An equipment qualification record form exists and concludes that the common electrical equipment listed in Table 3.14-3, including its connection assemblies, performs its function under the environmental conditions specified in the equipment qualification record form for the period of time required to complete the function.
		ii. An inspection will be performed of the common as-built electrical equipment listed in Table 3.14-3, including its connection assemblies.	ii. The common electrical equipment listed in Table 3.14-3, including its connection assemblies, is installed in its design location in a configuration bounded by the EQ record form.
03.	The RW-IIa components used for processing gaseous radioactive waste listed in Table 3.14-3 are constructed to the standards of RW-IIa.	An inspection and reconciliation analysis will be performed of the as-built RW-IIa components used for processing gaseous radioactive waste listed in Table 3.14-3.	A report exists and concludes that the as-built RW-IIa components used for processing gaseous radioactive waste listed in Table 3.14-3 meet the RW-IIa design criteria.

Table 3.14-2: Equipment Qualification - Shared Equipment Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1]

ITAAC No.	Discussion
03.14.01	FSAR Section 3.10, Seismic and Dynamic Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the Seismic Category I equipment, including its associated supports and anchorages, is qualified by type test, analysis, or a combination of type test and analysis to perform its function under the design basis seismic loads during and after an SSE. The qualification method employed for the Seismic Category I equipment is the same as the qualification method described for that type of equipment in FSAR Section 3.10. This method conforms to IEEE-344 and ASME QME-1 (as referenced in FSAR Section 3.10). The scope of equipment for this design commitment is the common, safety-related equipment, and the common, nonsafety-related equipment that provides one of the following nonsafety-related functions: • physical support of irradiated fuel (FHM, spent fuel storage racks, and RBC) • a path for makeup water to the UHS • containment of UHS water • monitors UHS water level The ITAAC verifies that: (1) a Seismic Qualification Report exists for each Seismic Category I component type, and (2) the seismic qualification record form concludes that the Seismic Category I equipment listed in Table 3.14-3, including its associated supports and anchorages, performs its function under the seismic design basis load conditions specified in the Seismic Qualification Report. After installation in the plant, an ITAAC inspection is performed to verify that the Seismic Category I equipment listed in Table 3.14-3, including its associated supports and anchorages, is installed in its design location in a Seismic Category I structure in a configuration bounded by the Seismic Qualification Report.
03.14.02	FSAR Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment, presents information to demonstrate that the common electrical equipment, including its connection assemblies, located in a harsh environment is qualified by type test or a combination of type test and analysis to perform its function under design basis harsh environmental conditions, experienced during normal operations, AOOs, DBAs, and post-accident conditions in accordance with 10 CFR 50.49. As defined in IEEE-Std-572-2006, IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations, a connection assembly is any connector or termination combined with related cables or wires as an assembly. The qualification method employed for the equipment is the same as the qualification method described for that type of equipment in FSAR Section 3.11. The scope of equipment for this design commitment is the nonsafety-related equipment that provides monitoring of the UHS water level and the nonsafety-related electrical equipment on the FHM and RBC used to physically support irradiated fuel. The ITAAC verifies that: (1) an equipment qualification record form exists for the common electrical equipment listed in Table 3.14-3 and addresses connection assemblies, (2) the equipment qualification record form concludes that the common electrical equipment, including its connection assemblies, performs its function under the environmental conditions specified in FSAR Section 3.11 and the equipment qualification record form, and (3) the required post-accident operability time for the common electrical equipment in the equipment qualification record form is in agreement with FSAR Section 3.11. After installation in the plant, an ITAAC inspection is performed to verify that the common electrical equipment listed in Table 3.14-3, including its connection assemblies, is installed in its design location in a configuration bounded by the equipment qualification record form.

Table 3.14-2: Equipment Qualification - Shared Equipment Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1) (Continued)

ITAAC No.	Discussion
03.14.03	The classification of SSC that contain radioactive waste in accordance with RG 1.143 is discussed in FSAR Section 3.2.1. The scope of the equipment for this design commitment is the nonsafety-related, radioactive waste components that meet both of the following criteria: (1) classified as RW-IIa in accordance with RG 1.143 and (2) designed for processing gaseous radioactive waste As described in FSAR Section 11.2.2 for the LRWS and FSAR Section 11.3.2 for the GRWS, component classification applies to components up to and including the first isolation device. Table 3.14-3 identifies the components for which this ITAAC is applicable. An ITAAC inspection and reconciliation analysis is performed of the as-built LRWS and GRWS RW-IIa components used for processing gaseous radioactive waste to ensure that deviations between the drawings used for construction and the as-built RW-IIa components are reconciled. A report concludes the as-built RW-IIa components meet the design criteria of RG 1.143, RW-IIa.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.14-3: Mechanical and Electrical / Instrumentation and Controls Shared Equipment

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category	EQ Category[1]
Module Assembly Equipment - Bolting					
00-MAEB-MHE-0007	Reactor flange tool support	N/A	N/A	I	N/A
Fuel Handling Equipment					
00-FHE-CRN-0002	NFE	Harsh	Electrical Mechanical	I	B A
00-FHE-CRN-0003	FHM	Harsh	Electrical Mechanical	I	B A
Spent Fuel Storage System					
None	Spent Fuel Storage Racks	N/A	N/A	I	N/A
Ultimate Heat Sink					
00-UHS-LE-0101A-1 00-UHS-LIT-0101A-1 00-UHS-LE-0101B-2 00-UHS-LIT-0101B-2	Pool level instruments	N/A	N/A	I	N/A
00-UHS-LE-0102A-1 00-UHS-LIT-0102A-1 00-UHS-LE-0102B-2 00-UHS-LIT-0102B-2	SFP level instruments	Harsh	Electrical	I	A
00-UHS-0001-BBD1-N	Water Makeup Line	N/A	N/A	I	N/A
00-UHS-HV-0001	Water Makeup Line Isolation Valve	N/A	N/A	I	N/A
Reactor Building Crane					
00-RBC-CRN-0001	RBC main hoist, lower block assembly, and sister hook	Harsh	Electrical Mechanical	I	B A
Reactor Building Components					
None	Seismic Category I portions of the UHS pool liner and dry dock liner	N/A	N/A	I	N/A
None	NPM support	N/A	N/A	I	N/A
00-RBCM-RPD-0001A 00-RBCM-RPD-0001B	RXB overpressurization vents	Harsh	Mechanical	I	A
None	RXB overpressurization always-open vents (2 total)	N/A	N/A	I	N/A

Table 3.14-3: Mechanical and Electrical / Instrumentation and Controls Shared Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category	EQ Category[1]
00-RBCM-RPD-0002A 00-RBCM-RPD-0002B 00-RBCM-RPD-0002C 00-RBCM-RPD-0002D 00-RBCM-RPD-0002E 00-RBCM-RPD-0002F 01-RBCM-RPD-0003 02-RBCM-RPD-0003 03-RBCM-RPD-0003 04-RBCM-RPD-0003 05-RBCM-RPD-0003 06-RBCM-RPD-0003	RXB blow off panels	Harsh	Mechanical	I	A
None	Dry dock gate supports	N/A	N/A	I	N/A
Liquid Radioactive Waste System					
00-LRW-DGS-0006A 00-LRW-DGS-0006B	Degasifiers	N/A	N/A	RW-IIa	N/A
00-LRW-CND-0073A 00-LRW-CND-0073B	Degasifier condensers	N/A	N/A	RW-IIa	N/A
00-LRW-VP-0080A 00-LRW-VP-0080A	Degasifier vacuum pumps	N/A	N/A	RW-IIa	N/A
Gaseous Radioactive Waste System					
00-GRW-TNK-0012	Charcoal guard bed	N/A	N/A	RW-IIa	N/A
00-GRW-AOV-0011 00-GRW-AOV-0013 00-GRW-AOV-0056 00-GRW-AOV-0058 00-GRW-CKV-0110 00-GRW-HV-0202 00-GRW-HV-0203A 00-GRW-HV-0203B 00-GRW-HV-9002 00-GRW-HV-9003 00-GRW-RV-0071	Charcoal guard bed valves	N/A	N/A	RW-IIa	N/A
00-GRW-SKD-0002A 00-GRW-SKD-0002B	Charcoal decay bed skids	N/A	N/A	RW-IIa	N/A

Table 3.14-3: Mechanical and Electrical / Instrumentation and Controls Shared Equipment (Continued)

Equipment Identifier	Description	EQ Environment	Qualification Program	Seismic Category	EQ Category[1]
00-GRW-AOV-0014A 00-GRW-AOV-0014B 00-GRW-AOV-0019A 00-GRW-AOV-0019B 00-GRW-AOV-0065A 00-GRW-AOV-0065B 00-GRW-AOV-0066A 00-GRW-AOV-0066B	Charcoal decay bed skid isolation valves	N/A	N/A	RW-IIa	N/A

Note:
1. EQ Categories:
    - A - Equipment that will experience the environmental conditions of DBA for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.
    - B - Equipment that will experience the environmental conditions of DBA through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand the accident environment for the time during which it must not fail with safety margin to failure.

Table 3.14-4: Equipment Qualification - Shared Equipment Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.14.01	X			X		
03.14.02	X			X		
03.14.03			X			

3.15 Human Factors Engineering

3.15.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The Human Factors Engineering (HFE) Program design process is employed to design the control rooms and the human-system interfaces (HSIs) and associated equipment while relating the high-level goal of plant safety into individual, discrete focus areas for the design.

The HFE and control room design team establish design guidelines, define program-specific design processes, and verify that the guidelines and processes are followed. The scope of the HFE Program includes the following:

- location and accessibility requirements for the control rooms and other control stations
- layout requirements of the control rooms, including requirements regarding the locations and design of individual displays and panels
- basic concepts and detailed design requirements for the information displays, controls, and alarms for HSI control stations
- coding and labeling conventions for control room components and plant displays
- HFE design requirements and guidelines for the screen-based HSI, including the actual screen layout and the standard dialogues for accessing information and controls
- requirements for the physical environment of the control rooms (e.g., lighting, acoustics, heating, ventilation, and air conditioning)
- HFE requirements and guidelines regarding the layout of operator workstations and work spaces
- corporate policies and procedures regarding the verification and validation of the design of HSI

Design Commitment

- The configuration of the main control room HSI is consistent with the design verified and validated by the integrated system validation as reconciled by the Design Implementation Implementation Plan.
- The MCR design incorporates HFE principles that reduce the potential for operator error.

3.15.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.15-1 contains the ITAAC for HFE.

Table 3.15-1: Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.15.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	The configuration of the main control room HSI is consistent with the design verified and validated by the integrated system validation as reconciled by the Design Implementation Implementation Plan.	An inspection will be performed of the as-built configuration of the main control room HSI.	A report exists and concludes the as-built configuration of the main control room HSI is consistent with the design verified and validated by the integrated system validation as reconciled by the Design Implementation Implementation Plan.
02.	The MCR design incorporates HFE principles that reduce the potential for operator error.	An integrated system validation (ISV) test is performed in accordance with the Verification and Validation Implementation Plan.	A report exists and concludes that acceptance criteria associated with each ISV test scenario are satisfied upon initial performance of the scenarios or upon remediation of failures.

Table 3.15-2: Human Factors Engineering Inspections, Tests, Analyses, and Acceptance Criteria Additional Information(1)

ITAAC No.	Discussion
03.15.01	FSAR Section 18.11, Design Implementation, describes the implementation of HFE aspects of the plant design. The Design Implementation activities verify that the final MCR is consistent with the verified and validated design resulting from the HFE design process. An ITAAC inspection is performed to verify that the as-built configuration of main control room HSI is consistent with the final as-designed HSI configuration. As used here, the final as-designed HSI configuration is the COL holder’s configuration-controlled design, which includes changes made subsequent to integrated system validation under a licensee’s configuration control process and includes resolution of human engineering discrepancies.
03.15.02	FSAR Section 18.10, Human Factors Verification and Validation, describes the ISV, which provides a comprehensive performance-based assessment of the design of the HSI resources based on their realistic operation within a simulator-driven MCR. The ISV is part of the overall HFE program. Integrated system validation is performed in accordance with the Verification and Validation Implementation Plan. The ISV uses a representative set of scenarios to assess the usability of the MCR and HSI resources and the tolerance of, or susceptibility to error. The acceptance criteria associated with each ISV test scenario identified in the Verification and Validation Results Summary Report are satisfied upon initial performance of the scenarios or upon remediation of failures.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

3.16 Physical Security System

3.16.1 Inspections, Tests, Analyses, and Acceptance Criteria Design Description

ITAAC System Description

The scope of this section is the physical security system, which is described in Section 13.6.

Design Commitments

- Vital equipment is located only within a vital area.
- Access to vital equipment requires passage through at least two physical barriers.
- The external walls, doors, ceilings, and floors in the MCR and central alarm station (CAS) are bullet-resistant.
- An access control system is installed for use by individuals who are authorized access to vital areas within the nuclear island and structures without escort.
- Unoccupied vital areas within the nuclear island and structures have locking devices and intrusion-detection devices that annunciate in the CAS.
- The CAS is located inside the protected area and the interior is not visible from the perimeter of the protected area.
- Security alarm devices in the RXB and CRB, including transmission lines to annunciators are tamper-indicating and self-checking, and alarm annunciation indicates the type of alarm and its location.
- Intrusion detection and assessment systems in the RXB and CRB provide visual display and audible annunciation of alarms in the CAS.
- Intrusion detection systems' recording equipment records 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.
- Emergency exits through the vital area boundaries within the nuclear island and structures are alarmed with intrusion detection devices and are secured by locking devices that allow prompt egress during an emergency.
- The CAS has landline telephone service with the control room and local law enforcement authorities.
- The CAS is capable of continuous communication with on-duty security force personnel.
- Non-portable communications equipment in the CAS remains operable from an independent power source in the event of the loss of normal power.

3.16.2 Inspections, Tests, Analyses, and Acceptance Criteria

Table 3.16-1 contains the ITAAC for the physical security system.

Table 3.16-1: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.16.xx)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
01.	Vital equipment is located only within a vital area.	An inspection will be performed of vital equipment.	Vital equipment is located only within a vital area.
02.	Access to vital equipment requires passage through at least two physical barriers.	An inspection will be performed of vital equipment.	Vital equipment is located within a protected area such that access to the vital equipment requires passage through at least two physical barriers.
03.	The external walls, doors, ceilings, and floors in the MCR and CAS are bullet-resistant.	A type test, analysis, or a combination of type test and analysis will be performed of the external walls, doors, ceilings, and floors in the MCR and CAS.	A report exists and concludes that the walls, doors, ceilings, and floors in the MCR and CAS are bullet-resistant.
04.	An access control system is installed for use by individuals who are authorized access to vital areas within the nuclear island and structures without escort.	A test will be performed of the access control system.	The access control system is installed and provides authorized access to vital areas within the nuclear island and structures only to those individuals with authorization for unescorted access.
05.	Unoccupied vital areas within the nuclear island and structures have locking devices and intrusion detection devices that annunciate in the CAS.	A test, inspection, or combination of test and inspection will be performed of unoccupied vital areas' intrusion detection equipment and locking devices.	Unoccupied vital areas within the nuclear island and structures are locked and alarmed and intrusion is detected and annunciated in the CAS.
06.	The CAS is located inside the protected area and the interior is not visible from the perimeter of the protected area.	An inspection will be performed of the CAS.	The CAS is located inside the protected area and the interior of the alarm station is not visible from the perimeter of the protected area.
07.	Security alarm devices in the RXB and CRB, including transmission lines to annunciators, are tamper-indicating and self-checking, and alarm annunciation indicates the type of alarm and its location.	A test will be performed of security alarm devices and transmission lines in the RXB and CRB.	Security alarm devices in the RXB and CRB, including transmission lines to annunciators, are tamper-indicating and self-checking; an automatic indication is provided when failure of the alarm system or a component thereof occurs or when the system is on standby power; and the alarm annunciation indicates the type of alarm and location.
08.	Intrusion detection and assessment systems in the RXB and CRB provide visual display and audible annunciation of alarms in the CAS.	A test will be performed of intrusion detection and assessment systems in the RXB and CRB.	The intrusion detection systems in the RXB and CRB provide a visual display and audible annunciation of alarms in the CAS.
09.	Intrusion detection systems' recording equipment records 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.	A test will be performed of the intrusion detection systems' recording equipment.	Intrusion detection systems' recording equipment is capable of recording each security alarm annunciation within the nuclear island and structures including each alarm, false alarm, alarm check, and tamper indication and the type of alarm, location, alarm circuit, date, and time.

Table 3.16-1: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 03.16.xx) (Continued)

No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
10.	Emergency exits through the vital area boundaries within the nuclear island and structures are alarmed with intrusion detection devices and are secured by locking devices that allow prompt egress during an emergency.	A test, inspection, or combination of test and inspection will be performed of emergency exits through the vital area boundaries within the nuclear island and structures.	Emergency exits through the vital area boundaries within the nuclear island and structures are alarmed with intrusion detection devices and secured by locking devices that allow prompt egress during an emergency.
11.	The CAS has landline telephone service with the control room and local law enforcement authorities.	A test, inspection, or combination of test and inspection will be performed of the CAS landline telephone service.	The CAS is equipped with landline telephone service with the control room and local law enforcement authorities.
12.	The CAS has continuous communication with on-duty security force personnel.	A test, inspection, or combination of test and inspection will be performed of the CAS continuous communication capabilities.	The CAS is capable of continuous communication with security personnel who have responsibilities within the physical protection program and during contingency response events.
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.	A test, inspection, or combination of test and inspection will be performed of the non-portable communications equipment in the CAS.	All non-portable communication devices in the CAS remain operable from an independent power source in the event of the loss of normal power.

Table 3.16-2: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[(1)]

ITAAC No.	Discussion
03.16.01	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. Vital equipment and vital areas are discussed in the report. An ITAAC inspection is performed of the as-built vital equipment to verify that the equipment is located in a vital area.
03.16.02	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. Provisions for accessing vital equipment are discussed in the report. An ITAAC inspection is performed of the as-built vital equipment location to verify that access to vital equipment within the nuclear island and structures requires passage through at least two physical barriers.
03.16.03	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. A type test, analysis, or a combination of type test and analysis are performed of the bullet-resisting barriers used in the external walls, doors, ceilings, and floors in the MCR and CAS. 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.
03.16.04	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. The access control system that limits access to vital areas within the nuclear island and structures to individuals with unescorted access authorization is discussed in the report. In accordance with FSAR Table 14.2-66, 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.
03.16.05	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. The report discusses that unoccupied vital area portals within the nuclear island and structures are equipped with locking devices and alarms that annunciate in the CAS. In accordance with FSAR Table 14.2-67, 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 CAS as described in Technical Report TR-118318, “NuScale Design of Physical Security Systems.”

Table 3.16-2: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[1] (Continued)

ITAAC No.	Discussion
03.16.06	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security related information that describes security design bases and requirements for security SSC. The CAS and its location is discussed in the report. An ITAAC inspection is performed of the as-built CAS to verify that it is located inside the protected area and the interior is not visible from the protected area perimeter.
03.16.07	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. Security alarms within the nuclear island and structures are discussed in the report. In accordance with FSAR Table 14.2-67, a preoperational test demonstrates that • security alarm devices, including transmission lines to annunciators, are tamper-indicating and self-checking. • alarm annunciation indicates the type of alarm and location. • an automatic indication is provided when failure of the alarm system or a component occurs, or when the system is on standby power.
03.16.08	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. The intrusion detection and assessment system within the nuclear island and structures is discussed in the report. In accordance with FSAR Table 14.2-67, 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 CAS.
03.16.09	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. The intrusion detection and assessment system within the nuclear island and structures is discussed in the report. In accordance with FSAR Table 14.2-67, 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.

Table 3.16-2: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria Additional Information[(1)] (Continued)

ITAAC No.	Discussion
03.16.10	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. Emergency exits at vital area boundaries within the nuclear island and structures are discussed in the report. In accordance with Table 14.2-67, 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 as described in Technical Report TR-118318.
03.16.11	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. The CAS’s landline telephone service is discussed in the report. In accordance with FSAR Table 14.2-61, a preoperational test, inspection, or a combination of test and inspection demonstrates that the CAS is equipped with conventional landline telephone service with the MCR and local law enforcement authorities as described in TR-118318.
03.16.12	FSAR Section 13.6 discusses that the physical security system 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. Technical Report TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. The CAS’s communication system is discussed in the report. In accordance with FSAR Table 14.2-61, a preoperational test, inspection, or a combination of test and inspection demonstrates that the CAS is capable of continuous communication with on-duty security force personnel as described in TR-118318.
03.16.13	FSAR Section 13.6 discusses that the physical security system 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. TR-118318, “NuScale Design of Physical Security Systems,” provides safeguards and security-related information that describes security design bases and requirements for security SSC. Non-portable communications equipment in the CAS is discussed in the report. In accordance with FSAR Table 14.2-61, a preoperational test, inspection, or a combination of test and inspection demonstrates that non-portable communications equipment in the CAS remains operable without disruption from an independent power source in the event of loss of normal power as described in TR-118318.

Note:
1) References to Tables and Figures refer to ITAAC unless the reference specifically states FSAR Tables or Figures.

Table 3.16-3: Physical Security System Inspections, Tests, Analyses, and Acceptance Criteria Top-Level Design Feature Categories

ITAAC No.	Design Basis Accident	Internal / External Hazard	Radiological	PRA & Severe Accident	Fire Protection	Physical Security
03.16.01						X
03.16.02						X
03.16.03						X
03.16.04						X
03.16.05						X
03.16.06						X
03.16.07						X
03.16.08						X
03.16.09						X
03.16.10						X
03.16.11						X
03.16.12						X
03.16.13						X