UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 188 AND 186

TO THE COMBINED LICENSE NOS. NPF-91 AND NPF-92, RESPECTIVELY

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MEAG POWER SPVM, LLC

MEAG POWER SPVJ, LLC

MEAG POWER SPVP, LLC

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT UNITS 3 AND 4

DOCKET NOS. 52-025 AND 52-026

1.0 INTRODUCTION

By letter dated August 24, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21236A305), Southern Nuclear Operating Company, Inc. (SNC or the licensee) requested that the U.S. Nuclear Regulatory Commission (NRC) amend the Vogtle Electric Generating Plant (VEGP) Units 3 and 4, Combined License (COL) Nos. NPF-91 and NPF-92, respectively. The request, License Amendment Request (LAR) 21-001, "Clarification of ITAAC Regarding Invessel Components," proposed changes that would revise Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Index Nos. 68 (2.1.03.01), 75 (2.1.03.06.i), 515 (2.5.01.03e), 565 (2.5.05.02.i), and 570 (2.5.05.03b) in COL Appendix C and plant-specific design control document (PS-DCD) Tier 1 information to (1) eliminate ITAAC requirements regarding verification of the location of certain equipment, (2) eliminate the requirement that certain inspections be performed on the "as-built" components, and (3) make other related changes. In part, the changes address certain components that cannot be installed in their final location until after fuel load.

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 52.103(g), the Commission must find prior to operation that the acceptance criteria in the combined license are met (except for those acceptance criteria that the Commission found were met under § 52.97(a)(2)), where operation includes the licensee loading the initial core into the reactor. As noted in the VEGP Units 3 and 4 Updated Final Safety Analysis Report (UFSAR), Subsection 14.3.2.2, “Inspections, Tests, Analyses, and Acceptance Criteria,” one of the selection criteria for ITAAC is that “the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load.” The ITAAC that are the subject of this request relate to specific equipment, and include “as-built” components (for example, invessel components) that cannot be placed in their final operational location until after the 10 CFR 52.103(g) finding. As explained below, ITAAC required to be performed on as-built structures, systems and components (SSCs) may be completed only after the SSCs in question have been installed in their final operational location. Thus, those SSCs that cannot be installed in their final operational location until after the 10 CFR 52.103(g) finding should not have been subject to the “as-built” ITAAC requirement in accordance with 10 CFR 52.103(g) and the selection criteria identified above.

The VEGP, Units 3 and 4, COL Appendix C, Section 1.1, “Definitions,” defines “as-built” to mean the “physical properties of a structure, system, or component following the completion of its 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 structure, system, or component may be based on measurements, inspections, or tests that occur prior to installation, provided that subsequent fabrication, handling, installation, and testing does not alter the properties.” SNC stated that the ITAAC identified in LAR 21-001 cannot be completed as they currently exist in the COL, considering the COL Appendix C definition of “as-built” that establishes that “as-built” structures, systems and components (SSCs) must be in their final operational location upon ITAAC completion and prior to ITAAC Closure Notification (ICN) submittal, because certain components will not be placed in their final location until after the NRC finds that the acceptance criteria for all ITAAC have been met as required by 10 CFR 52.103(g). SNC, in the LAR, proposed changes to certain ITAAC acceptance criteria to address the fact that specific equipment cannot be verified “as-built.” However, the proposed changes, in part, also affect SSCs that can be installed prior to initial fuel load.

Pursuant to Section 10 CFR 52.63(b)(1) and 52.98(f), SNC also requested an exemption from the provisions of 10 CFR Part 52, Appendix D, “Design Certification Rule for the AP1000 Design,” Section III.B, “Scope and Contents.” The requested exemption would allow a departure from the corresponding portions of the certified information in Tier 1 of the generic Westinghouse AP1000 DCD[1]. In order to modify the PS-DCD Tier 1 information, the NRC must find SNC’s exemption request included in its submittal for the LAR to be acceptable. The staff’s review of the exemption request, as well as the LAR, is included in this safety evaluation.

The staff notes that the proposed markups in Enclosure 3 of LAR 21-001, specifically for ITAAC Index No. 68 and ITAAC Index No. 565, item 2.i), for Units 3 and 4, contain non-standard language for this type of change. The issued pages for Units 3 and 4 were edited by the NRC to reflect the typical language for similar changes. That is, for ITAAC Index No. 68, the NRC revised the text in the Design Commitment column to show the text, for Unit 3, “Not used per Amendment No. 188” (and “Not used per Amendment No. 186” for Unit 4), and deleted the “Not used” in both the Inspections, Tests, Analyses, and the Acceptance Criteria columns. For ITAAC Index No. 565, Item 2.i), the staff replaced “Not Used” in the ITA and AC columns with “Not used per Amendment No. 188” for Unit 3 (and “Not used per Amendment No. 186” for Unit

[1] While the licensee describes the requested exemption as being from Section III.B of 10 CFR Part 52, Appendix D, the entirety of the exemption pertains to proposed departures from Tier 1 information in the generic DCD. In the remainder of this evaluation, the NRC will refer to the exemption as an exemption from Tier 1 information to match the language of Section VIII.A.4 of 10 CFR Part 52, Appendix D, which specifically governs the granting of exemptions from Tier 1 information.

4). The staff views this as an editorial change that is not in any way substantively different from what was proposed, and the change is being made only to conform with standard usage to eliminate the confusion that may be created if the text remained as shown in the LAR markup for ITAAC Index No. 68 and ITAAC Index No. 565, item 2.i).

The staff notes that the LAR's proposed markups for ITAAC Index No. 75, item 6.iii), for both Units 3 and 4 do not contain the same language that was issued by the NRC in this license amendment. The issued text for ITAAC Index No. 75, item 6.iii) in this amendment applies the requested change to a more limited set of components than the markup in LAR Enclosure 3 to be consistent with the justification provided in the LAR and the LAR's discussion of SNC's planned actions. The basis for this limitation is discussed in the evaluation below. On October 13, 2021, SNC sent revised markup pages consistent with this limitation in the staff's approval (ADAMS Accession No. ML21287A258).

2.0 REGULATORY EVALUATION

LAR 21-001 proposes changes to the COL, Appendix C, and Tier 1 PS-DCD for both VEGP Units 3 and 4, in a way that, if approved, would allow completion of the identified ITAAC prior to fuel load consistent with the existing facility design. The staff considered the following regulatory requirements in reviewing the LAR that included the proposed changes:

Appendix D, Section VIII.A.4 to 10 CFR Part 52 states that exemptions from Tier 1 information are governed by the requirements in 10 CFR 52.63(b)(1) and 10 CFR 52.98(f). It also states that the Commission will deny a request for an exemption from Tier 1 if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design.

10 CFR 52.63(b)(1) allows the licensee who references a design certification rule to request NRC approval for an exemption from one or more elements of the certification information. The Commission may only grant such a request if it determines that the exemption will comply with the requirements of 10 CFR 52.7, which, in turn, points to the requirements listed in 10 CFR 50.12 for specific exemptions. In addition to the factors listed in 10 CFR 52.7, the Commission shall consider whether the special circumstances, as required by 10 CFR 52.7 and 50.12, outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. Therefore, any exemption from the Tier 1 information certified by Appendix D to 10 CFR Part 52 must meet the requirements of 10 CFR 50.12, 52.7, and 52.63(b)(1).

10 CFR 52.97(b) requires the NRC to "identify within the combined license the inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that, if met, 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 of 1954, as amended (AEA)], and the Commission's rules and regulations."

10 CFR 52.98(f) requires NRC approval for any modification to, addition to, or deletion from, the terms and conditions of a COL.

10 CFR 50.49 requires that each holder of a combined license establish a program for environmental qualification (EQ) of electric equipment important to safety and maintain documentation of that qualification.

10 CFR Part 50.55a, “Codes and standards,” paragraphs (a)(2)(iii)-(iv) incorporate by reference the requirements of Institute of Electrical and Electronics Engineers (IEEE) 603-1991 and the correction sheet dated January 30, 1995, into the rule. The rule also states, in part, in section 50.55a(h)(3) that “Applications filed on or after May 13, 1999, for construction permits and operating licenses under this part, and for design approvals, design certifications, and combined licenses under 10 CFR Part 52 of this chapter, must meet the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.”

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2, as it relates to protection against natural phenomena states the following: “Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and (3) the importance of the safety functions to be performed.”

10 CFR Part 50, Appendix A, GDC 4, as it relates to the environmental and dynamic effects design bases states the following: “Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.”

10 CFR Part 50 Appendix A, GDC 24, as it relates to separation of protection and control systems states, in part, the following: “The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system.”

Regulatory guidance referred to in this evaluation includes the following:

- Nuclear Energy Institute (NEI) 08-01, Revision 5 – Corrected, “Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52,” June 2014 (ADAMS Accession No. ML14182A158);

- Regulatory Guide (RG) 1.215, Revision 2, “Guidance for ITAAC Closure Under 10 CFR Part 52,” July 2015 (ADAMS Accession No. ML15105A447); and

- RG 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," June 1984 (ADAMS Accession No. ML003740271).

3.0 TECHNICAL EVALUATION

LAR 21-001 proposes changes that would revise ITAAC Index No. 68 (2.1.03.01), No. 75 (2.1.03.06.i), No. 515 (2.5.01.03e), No. 565 (2.5.05.02.i), and No. 570 (2.5.05.03b) to (1) eliminate ITAAC requirements regarding verification of the location of certain equipment, (2) eliminate the requirement that certain inspections be performed on the "as-built" components, and (3) make other related changes. In part, the changes address certain components that cannot be installed in their final location until after fuel load. As such, SNC states that these ITAAC conflict with 10 CFR 52.103(g), which requires that all ITAAC must be completed prior to loading the initial core, and statements of what constitutes an acceptable ITAAC in the VEGP UFSAR.

3.1 EVALUATION OF THE REQUESTED CHANGES

The NRC staff evaluated the information presented by the licensee in Enclosures 1 – 3 of the August 24, 2021 submittal in determining the acceptability of LAR 21-001. The NRC staff did not evaluate the information provided in Enclosure 4, "Draft Revised Uncompleted Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Notices (UINs) Pending NRC Approval of LAR and Exemption," because it was submitted "for information only." The staff will evaluate UINs and ITAAC Closure Notifications (ICNs) when formally submitted on the docket by the licensee.

3.1.1 ITAAC Index No. 68 (2.1.03.01)

As described in LAR 21-001, SNC proposed to remove the ITAAC Index No. 68 requirement to verify the reactor system (RXS) functional arrangement. Appendix C of the VEGP Units 3 and 4 COLs defines "functional arrangement" (for a system) as "the physical arrangement of systems and components to provide the service for which the system is intended, and which is described in the system design description."

The purpose and scope of functional arrangement ITAAC are discussed in NEI 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 5 – Corrected (ADAMS Accession No. ML14182A158), which is approved for use by RG 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52," Revision 2 (ADAMS Accession No. ML15105A447), with certain exceptions and additional guidance not relevant to this LAR. The functional arrangement ITAAC for a system is limited to the components identified in the Tier 1 Design Description for the system, including any referenced tables and figures.

In LAR 21-001, SNC requested that ITAAC Index No. 68 be deleted based on the following statements:

> Because certain RXS components will not be in their final operational location until after fuel is loaded, ITAAC Index No. 68 cannot be closed in conformance with the interpretation and understanding that "as-built" SSCs must be installed in their final operational location prior to ITAAC closure. Furthermore, for those certain RXS components, ITAAC Index No. 68 with this interpretation of "as-built" does not meet the UFSAR Subsection 14.3.2.2 selection criteria of "the ITAAC do not include any

inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load."

…Other ITAAC also demonstrate that the reactor system has been constructed in accordance with the design to the extent possible prior to fuel load. For example, ITAAC 2.1.03.02a (also referred to as ITAAC Index No. 69) demonstrates that the as-built RXS accommodates the fuel assembly and control rod drive mechanism pattern shown in Figure 2.1.3-1 and the control assemblies (rod cluster and gray rod) and drive rod arrangement shown in Figure 2.1.3-2. ITAAC 2.1.03.02c (also referred to as ITAAC Index No. 71) demonstrates that the as-built RXS accommodates the reactor vessel arrangement shown in Figure 2.1.3-3. ITAAC 2.1.03.03 (also referred to as ITAAC Index No. 72) demonstrates that inspections are performed and that the specified RXS components and pressure boundary welds meet [American Society of Mechanical Engineers] ASME Code Section III requirements; additionally, a hydrostatic test is performed as required by the ASME Code Section III. ITAAC 2.1.03.06.i (also referred to as ITAAC Index No. 75) demonstrates that the specified equipment has been adequately qualified under seismic conditions and harsh environments per design requirements. ITAAC 2.1.03.13 (also referred to as ITAAC Index No. 88) demonstrates that the fuel assemblies and rod cluster control assemblies intended for the initial core load and listed in Table 2.1.3-1 have been designed and constructed in accordance with the principal design requirements. Finally, ITAAC 2.1.03.14 (also referred to as ITAAC Index No. 89) demonstrates the acceptability of the reactor vessel head top surface and penetration nozzles through a preservice visual examination.

The staff reviewed the information in Enclosure 1 of LAR 21-001 and finds that the fuel assemblies, the rod control cluster assemblies (RCCAs), the gray rod cluster assemblies (GRCAs), and the incore instrument QuickLoc assemblies cannot be installed in their final location to support plant operations until after the 10 CFR 52.103(g) finding has been made and therefore, the as-built requirement for installed location cannot be met. Additionally, the subsequent reactor vessel assembly and fuel load is adequately controlled and verified during the initial test program. Furthermore, the staff finds that the ITAAC Index No. 68 requirement for these components does not meet the UFSAR Subsection 14.3.2.2 selection criteria of "the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load." The staff notes that functional arrangement, described in COL Appendix C, Section 2.1.3, includes Table 2.1.3-3, that also specifies the location of the fuel assemblies, RCCAs, and GRCAs prior to fuel load is in the auxiliary building, indicating there was no expectation that these components would be installed in their final as-built location prior to fuel loading.

Additionally, the staff finds the functional arrangement for the remainder of the equipment listed in Table 2.1.3-1, (i.e., the reactor vessel (RV), the reactor upper and lower internals, the control rod drive mechanisms, and source, intermediate, and power range detectors), which can be installed prior to the 10 CFR 52.103(g) finding, is adequately verified via other ITAAC. In addition to those ITAAC identified by the licensee, the staff finds ITAAC Index No. 78 performs both pre- and post- flow test visual inspections of the as-built reactor internals and verifies the lower internals are equipped with holders for material surveillance capsules. ITAAC Index No. 80 verifies the flow area of the RV direct vessel injection (DVI) nozzles. Finally, the reactor coolant system (RCS) ITAAC Index No. 41 verifies RCS flow rate through the RV.

The staff finds that SNC's proposed changes do not modify the design of equipment, delete any technical requirements, or impact the ability of an SSC to perform its function and that the remaining ITAAC continue to provide reasonable assurance that the facility has been constructed and will be operated in conformity with the license, the AEA, and NRC rules and regulations. Therefore, within the scope of this license amendment request, the NRC concludes that 10 CFR 52.97(b) is satisfied and the proposed changes to Table 2.1.3-2 for ITAAC Index No. 68 (2.1.03.01), shown below, are acceptable.

COL Appendix C Table 2.1.3-2 (for ITAAC Index No. 68) is revised as follows for Unit 3 (the Unit 4 COL is revised identically, except that the Amendment No. is 186):

Table 2.1.3-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
68	2.1.03.01	Not used per Amendment No. 188~~1. The functional arrangement of the RXS is as described in the Design Description of this Section 2.1.3.~~	~~Inspection of the as-built system will be performed.~~	~~The as-built RXS conforms with the functional arrangement as described in the Design Description of this Section 2.1.3.~~

3.1.2 ITAAC Index No. 75 (2.1.03.06.i)

Following license amendment 85 and 84 for VEGP, Units 3 and 4, respectively (ADAMS Accession No. ML17216A064), the amended ITAAC Index No. 75 represented a consolidation of former ITAAC Index Nos. 75 (2.1.03.06.i), 76 (2.1.03.06.ii), 77 (2.1.03.06.iii), 81 (2.1.03.09a.i) and 82 (2.1.03.09a.ii). Thus, three ITAAC verification activities are required to demonstrate satisfaction of design commitment item 6 in Table 2.1.3-2, which states, "The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function." These three ITAAC verification activities are: (1) verification that specific seismic Category I equipment are located on the Nuclear Island (ITAAC item 6.i)), (2) seismic qualification of the components (ITAAC item 6.ii)), and (3) verification that the as-built equipment including anchorage are bounded by the seismic qualification (ITAAC item 6.iii)). SNC's proposed changes to ITAAC Index No. 75 affect ITAAC items 6.i) and 6.iii). No changes were proposed for design commitment item 6 or ITAAC item 6.ii).

ITAAC Index No. 75, Item 6.i) currently states, "[i]nspection will be performed to verify that the seismic Category I equipment identified in Table 2.1.3-1 is located on the Nuclear Island." The acceptance criterion states, "[t]he seismic Category I equipment identified in Table 2.1.3-1 is located on the Nuclear Island."

SNC requested item 6.i) be revised to exclude components that may not be located on the nuclear island prior to the 10 CFR 52.103(g) finding based on the following information:

> ITAAC Index No. 75, item 6.i), requires that the seismic Category I equipment identified in Table 2.1.3-1 is located on the Nuclear Island. The seismic Category I equipment identified in Table 2.1.3-1 includes components which are stored in protected environments off the nuclear island until it is time for loading the initial core, namely the fuel assemblies, the rod cluster control assemblies, the gray rod cluster assemblies, and the incore instrument QuickLoc assemblies.
>
> …Fuel loading and operation of the RXS will not and cannot occur until the equipment is "on the Nuclear Island." Thus, confirming this equipment is "on the Nuclear Island" is an

> unnecessary requirement that does not serve the underlying purpose of this ITAAC which is to confirm completion of the [seismic qualification] activities that can be completed prior to the initial fuel load.

The staff reviewed the information in Enclosure 1 of LAR 21-001, for the change requested for ITAAC Index No. 75, Item 6.i). The staff finds the fuel assemblies, RCCAs, GRCAs, and incore instrument QuickLoc assemblies cannot be installed in their final locations until after the 10 CFR 52.103(g) finding. Removing the components from their protective storage environment, prior to installation, provides neither a technical or safety benefit and would only serve to subject the components to potential damage. Furthermore, the future installed location of the components is determined by the location of the reactor vessel, which is verified by this ITAAC to be on the nuclear island. The staff finds that SNC's proposed changes do not delete any technical requirements, impact the ability of an SSC to perform its function, or impact safety and the reasonable assurance finding.

The staff reviewed the information in Enclosure 1 of LAR 21-001, for the change requested for ITAAC Index No. 75, Item 6.iii). ITAAC Index No. 75, Item 6.iii) states, "Inspection will be performed for the existence of a report verifying that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions." The acceptance criterion states, "[a] report exists and concludes that the as-built equipment including anchorage is seismically bounded by the tested or analyzed conditions."

SNC, in LAR 21-001, requested ITAAC Index No. 75, Item 6.iii) be revised to remove the "as-built" attribute from the inspection requirement for the equipment listed in VEGP, Units 3 and 4, COL Table 2.1.3-1 and acceptance criteria based on the following:

> The seismic Category I equipment identified in Table 2.1.3-1 includes the reactor vessel, the upper and lower internals assemblies, the fuel assemblies, rod assemblies [such as RCCAs and GRCAs], control rod drive mechanisms, incore instrument QuickLoc assemblies, and the source, intermediate and power range detectors. Per the COL license conditions governing testing and fuel loading and UFSAR Section 4.2 describing the fuel system design, the fuel assemblies, the rod cluster control assemblies, the gray rod cluster assemblies, and the incore instrument QuickLoc assemblies will not be installed in their final operational locations until after the initial core is loaded with fuel assemblies in the reactor vessel. Accordingly, these portions of ITAAC Index No. 75 cannot be closed in conformance with the interpretation and understanding that "as-built" SSCs must be installed in their final operational location prior to ITAAC closure.
>
> …Therefore, the inspection of these invessel components listed in Table 2.1.3-1 with this interpretation of "as-built" do not meet the UFSAR Subsection 14.3.2.2 selection criteria for ITAAC that "the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load" and should be excluded from ITAAC Index No. 75.
>
> To the extent that the installation prior to fuel load is possible for other seismically-qualified components, an inspection is conducted to confirm the satisfactory installation of the seismically qualified components identified in the table. The inspection verifies the equipment make/model/serial number, as-designed equipment mounting orientation, anchorage and clearances, and electrical and other interfaces. For components not installed prior to fuel load, the pre-fuel-load inspection is accomplished by verifying a quality assurance data package exists that concludes that the equipment was constructed as per design. Additional verifications are performed following 52.103(g) as addressed in UFSAR Subsection 14.2.10.

> The inspection conducted for each component in the table [Table 2.1.3-1] considers the critical seismic attributes identified in the associated Equipment Qualification Report for that component. The inspection confirms that the equipment, including anchorage, is seismically bounded by the tested or analyzed conditions.

The staff reviewed the information in Enclosure 1 of LAR 21-001, for the changes requested for ITAAC Index No. 75, Item 6.iii). SNC's proposed revision to item 6.iii) is to remove the phrase "as-built" from both the ITA and AC of the ITAAC. This change effectively removes the "as-built" requirement for all of the equipment listed in Table 2.1.3-1. However, the staff understands based on the LAR that eight types of components listed in this table can be verified in their "as-built" locations. The staff has divided the equipment listed in Table 2.1.3-1 into two evaluation sections below, based on these groups.

Fuel Assemblies, RCCAs, GRCAs, and Incore Instrument Quickloc Assemblies

The staff reviewed UFSAR Section 4.2 and the fuel system design and finds that the fuel assemblies, the RCCAs, GRCAs, and the incore instrument QuickLoc assemblies cannot be installed in their final operational locations until after the 10 CFR 52.103(g) finding and the initial core is loaded in the reactor vessel, and that applying the "as-built" requirement in this ITAAC to these components does not meet the UFSAR Subsection 14.3.2.2 selection criteria for ITAAC. Also, the seismic qualification of these components can still be demonstrated without the components being installed. UFSAR Section 3.10.1.2, "Performance Requirements for Seismic Qualification," states that an equipment qualification data package is developed for the instrumentation and electrical equipment classified as seismic Category I. Each equipment qualification data package establishes the safety-related functional requirements of the equipment to be demonstrated during and after a seismic event. UFSAR Section 3.10.2 identifies that the methods and procedures for qualification of seismic Category I electrical equipment, instrumentation, and mechanical equipment is by analysis and testing to demonstrate structural integrity and operability. UFSAR Appendix 3D, Attachment E, Section E.6, Qualification by Analysis, states analysis is used to demonstrate the structural adequacy of the passive mechanical equipment being qualified by showing that the calculated stresses do not exceed the design allowable stresses specified in ASME Code, Section III. UFSAR Section 3.10.2.2, "Seismic and Operability Qualification of Active Mechanical equipment," states that active mechanical equipment is qualified for both structural integrity and operability for its intended service conditions by a combination of test and analysis. The test and analysis methods utilized in qualification of these components provide adequate confidence of operability under required plant conditions because the qualification methods explicitly consider the installation and component's location. With the removal of the "as-built" requirement for these components, ITAAC item 6.iii) still requires inspection and verification that equipment in Table 2.1.3-1 is seismically bounded by the tested or analyzed conditions including anchorage. Additional verifications are performed following the 10 CFR 52.103(g) finding, as addressed in UFSAR Subsection 14.2.10, "Startup Test Procedures." Therefore, the subsequent reactor vessel assembly and fuel load is adequately controlled and verified during the initial test program. And in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," the licensee must follow quality assurance requirements for installation to ensure that the final installed configuration is bounded by the analysis and test used for qualification. The staff is approving the proposed change to remove "as-built" from the ITAAC item 6.iii) for only these four types of components.

Reactor Vessel, Reactor Upper Internals Assembly, Reactor Lower Internals Assembly, CRDMs, Source Range Detectors, Intermediate Range Detectors, Upper and Lower Power Range Detectors

SNC did not provide, for these eight types of components, a basis for how removing “as-built” from item 6.iii) meets the requirement in 10 CFR 52.97(b) that ITAAC verify that the facility “has been constructed” as required. Instead, the LAR indicates that SNC intends to complete the ITAAC for these components after their installation. Since a justification for removing the “as-built” requirement for these eight types of components was not provided, and the LAR indicates that SNC, in fact, plans to satisfy the “as-built” requirement for these components, the staff is not approving the removal of the “as-built” requirement for these eight types of components.

Therefore, the staff’s approval of the change to ITAAC Index no. 75, item 6.iii) is limited to the four types of components previously discussed. For the remaining eight types of components, removal of the “as-built” requirement is not approved. For that reason, the NRC is adding a sentence to SNC’s proposed revision of the ITA in ITAAC Index No. 75 item 6.iii) as follows:

> Inspection will be performed for the existence of a report verifying that the equipment including anchorage is seismically bounded by the tested or analyzed conditions. This inspection must be performed on the as-built equipment except for the fuel assemblies, rod cluster control assemblies, gray rod cluster assemblies, and incore instrument QuickLoc assemblies.

The staff provided its analysis of SNC’s proposed change to exclude certain equipment from coverage by ITAAC Index No. 75, item 6.i) and remove “as-built” from ITAAC Index No. 75, Item 6.iii), discussed above. The staff finds that SNC’s proposed changes, as limited by the NRC, do not delete any technical requirements, do not impact the ability of an SSC to perform its function, or impact safety and the reasonable assurance finding. Based on these findings the staff concludes that with the proposed changes, as modified by NRC, the ITAAC will be sufficient to verify that the facility has been constructed and will operate in accordance with the license, the AEA, and the Commission’s rules and regulations. Therefore, within the scope of this license amendment, the NRC concludes that 10 CFR 52.97(b) is satisfied. Thus, SNC’s proposed changes, as modified, are acceptable.

COL Appendix C Table 2.1.3-2 (for ITAAC Index No. 75) is revised as follows:

Table 2.1.3-2 Inspections, Tests, Analyses, and Acceptance Criteria				
No.	**ITAAC No.**	**Design Commitment**	**Inspections, Tests, Analyses**	**Acceptance Criteria**
75	2.1.03.06.i	6. The seismic Category I equipment identified in Table 2.1.3-1 can withstand seismic design basis loads without loss of safety function.	i) Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.1.3-1 (except fuel assemblies, rod cluster control assemblies, gray rod cluster assemblies, and incore instrument QuickLoc assemblies) is located on the Nuclear Island. ii) (no changes) iii) Inspection will be performed for the existence of a report verifying that the ~~as-built~~ equipment including anchorage is seismically bounded by the tested or analyzed conditions. This inspection must be performed on the as-built equipment except for the fuel assemblies, rod cluster control assemblies, gray rod cluster assemblies, and incore instrument QuickLoc assemblies.	The seismic Category I equipment identified in Table 2.1.3-1 (except fuel assemblies, rod cluster control assemblies, gray rod cluster assemblies, and incore instrument QuickLoc assemblies) is located on the Nuclear Island. ii) (no changes) iii) A report exists and concludes that the ~~as-built~~ equipment including anchorage is seismically bounded by the tested or analyzed conditions.

3.1.3 ITAAC Index No. 515 (2.5.01.03e)

For ITAAC Index No. 515, the Design Commitment is "[t]he sensors identified on Table 2.5.1-3 are used for DAS input and are separate from those being used by the PMS and plant control system," with an Acceptance Criteria of, "[t]he sensors identified on Table 2.5.1-3 are used by DAS and are separate from those being used by PMS and plant control system." The inspection to be performed to support confirmation of the diverse actuation system (DAS) input sensors states that "[i]nspection of the as-built system will be performed."

The DAS is a non-safety-related system that provides a diverse backup to the safety-related Protection and Safety Monitoring System (PMS) in the AP1000 design. The DAS is designed to meet the applicable requirements and protective functions established by 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants." 10 CFR 50.62 provides, in part, that ATWS equipment (e.g., DAS) must be independent and diverse (from sensor output to the final actuation device) from the existing reactor trip system (i.e., PMS). The non-safety-related DAS is not required to prevent or mitigate the effects of design basis accidents or to provide or perform safety-related functions or safety-related features to mitigate or prevent against the effects of design basis accidents. As such, the DAS must meet applicable criteria in 10 CFR Part 50 Appendix A, GDC 24, "Separation of protection and control systems," which requires, in part, "The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system."

In LAR 21-001 SNC requested ITAAC Index No. 515 be revised to remove the "as-built" location attribute for these inspections only for the core exit temperature (CET) sensors on the following basis:

> [T]he inspection requirement is of the "as-built" system. The sensors identified in Table 2.5.1-3 include the core exit temperature sensors. However, pursuant to UFSAR Subsection 4.4.6.1, the sensors are to be installed within the core and thus, cannot be installed in their final operational location prior to constitution of a core by the initial fuel load. Accordingly, as currently written, ITAAC Index No. 515 cannot be closed in conformance with the interpretation and understanding that "as-built" SSCs must be installed in their final operational location prior to ITAAC closure. Furthermore, the inspection of these sensors with this interpretation of "as-built" does not meet the UFSAR Subsection 14.3.2.2 selection criteria for ITAAC that, "the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load," and should be excluded from ITAAC Index No. 515.
>
> …Construction drawings illustrate the DAS sensor flow and indication architecture. An inspection of Quality Release and Certificate of Conformance document, construction drawings, and completed construction records is performed to confirm that the sensors identified in the table were installed per the DAS sensor input requirements and are separate from those being used by the PMS and plant control system with the exception of the core exit temperature sensor installation.

The licensee also described measures, other than ITAAC 515, used to confirm that the CET sensors used by the DAS are separate than those being used by the PMS and the plant control system. For example, UFSAR Section 14.2.9.1.13, "Incore Instrumentation System Testing," describes provisions to ensure DAS CET sensors display properly in the main control room. That testing involves the DAS CET sensors and sensor flow data along with other testing to verify the DAS sensors are operating properly.

The staff reviewed the information in Enclosure 1 of LAR 21-001 and finds that the incore instrument thimble assemblies (IITAs), which contain the CET sensors identified in COL Appendix C, Table 2.5.13 (tag number IIS-009, IIS-013, IIS-030, and IIS-034), cannot be installed in their final as-built location to support plant operation until after the 10 CFR 52.103(g) finding has been made and initial fuel load has been completed. Therefore, applying the "as-built" requirement in this ITAAC to these components does not meet the UFSAR Subsection 14.3.2.2 selection criteria for ITAAC. The proposed change does not remove the DAS CET sensors from verification by ITAAC; the DAS CET sensors are still part of the system to be inspected and are within the scope of, and must meet, the acceptance criteria. The proposed change only means that the inspection may be completed without the DAS CET sensors being installed. Verification that the DAS CET sensors are separate from those provided to PMS and PLS based on review of design, construction, and installation documents is reasonable. Additionally the subsequent reactor vessel assembly and fuel load is adequately controlled and verified during the initial test program; this addresses installation of the DAS CET sensors. The staff finds that SNC's proposed change is acceptable. The proposed change does not delete any technical requirements or impact the ability of an SSC to perform its function. Based on these findings, the staff concludes that the LAR's proposed revisions to ITAAC Index No. 515 will be sufficient to verify that the facility has been constructed and will operate in accordance with the license, the provisions of the AEA, and the Commission's rules and regulations. Therefore, the NRC concludes that 10 CFR 52.97(b) is satisfied and SNC's proposed changes to be acceptable.

COL Appendix C Table 2.5.1-4 is revised (for ITAAC Index No. 515) as follows:

Table 2.5.1-4
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
515	2.5.01.03e	3.e) The sensors identified on Table 2.5.1-3 are used for DAS input and are separate from those being used by the PMS and plant control system.	Inspection of the as-built system will be performed except for the core exit temperature sensor installation.	The sensors identified on Table 2.5.1-3 are used by DAS and are separate from those being used by the PMS and plant control system.

3.1.4 ITAAC Index No. 565 (2.5.05.02.i)

Following license amendments 85 and 84 for VEGP Units 3 and 4, respectively, ITAAC Index No. 565 represents a consolidation of former ITAAC Index Nos. 565 (2.5.05.02.i), 566 (2.5.05.02.ii), 567 (2.5.05.02.iii), 568 (2.5.05.03a.i), and 569 (2.5.05.03a.ii). Current ITAAC Index No. 565 verifies the seismic and environmental qualification of components identified in COL, Appendix C, Table 2.5.5-1. The only components listed in Table 2.5.5-1 are the IITAs, which are labeled "Incore Thimble Assemblies" in Table 2.5.5-1.

As described in UFSAR Subsection 4.4.6.1, "Incore Instrumentation," the in-core instrumentation system (IIS) consists, in part, of 42 in-core instrument thimble assemblies (IITAs) which are installed within various fuel assemblies within the core. Each IITA is composed of multiple self-powered detectors (SPDs) and one core exit thermocouple assembly, contained within individual inner sheaths and collectively an outer sheath. The IITAs provide Class 1E core exit thermocouple inputs to the Protection and Safety Monitoring System (PMS), non-Class 1E core exit thermocouple inputs to the DAS, and non-Class 1E SPD signals to the on-line power distribution monitoring system (OPDMS).

SNC's current ITAAC Index No. 565 is described below.

There are two design commitments in ITAAC Index No. 565 and a set of ITAAC activities related to each design commitment. The first design commitment, item 2, states: "The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis dynamic loads without loss of safety function." Three ITAAC verification activities are required to demonstrate that this design commitment is met: (1) verification the installed components are located on the Nuclear Island (ITAAC item 2.i)), (2) seismic qualification of the components (ITAAC item 2.ii)), and (3) verification that the as-built components are bounded by the seismic qualification (ITAAC item 2.iii)).

The second design commitment, item 3.a), states: "The Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function." Two ITAAC verification activities are required to demonstrate that this design commitment is met: (1) qualification of the Class 1E equipment for a harsh environment (item 3.a.i)) and (2) verification that the as-built Class 1E components are bounded by the qualification (item 3.a.ii)).

SNC in LAR 21-001 requested ITAAC Index No. 565, item 2.i) be deleted and that items 2.iii) and 3.a.ii) be revised to remove the "as-built" attribute from the inspection, test, and analysis requirements and the acceptance criteria based on the following discussion:

> ITAAC Index No. 565, Acceptance Criteria item 2.i), requires that the seismic Category I equipment identified in Table 2.5.5-1 is located on the Nuclear Island. The seismic Category I equipment identified in Table 2.5.5-1 is the Incore Thimble Assemblies (at least three assemblies in each core quadrant). The incore thimble assemblies are stored in protected environments off the nuclear island until it is time for loading the initial core. These include the core exit temperature sensors of ITAAC Index No. 515. Storing such sensitive equipment (on the Nuclear Island) would unnecessarily subject it to potential damage. Further, fuel loading and loading of the incore thimble assemblies will not and cannot occur until the equipment is "on the Nuclear Island." Thus, this is an unnecessary requirement that does not serve the underlying purpose of ITAAC which are required to be completed prior to the initial fuel load.
>
> …The seismic Category I equipment identified in Table 2.5.5-1 is the Incore Thimble Assemblies (at least three assemblies in each core quadrant). As discussed above, this equipment will not be installed in their final operational locations until after the initial core is loaded with fuel assemblies in the reactor vessel. Accordingly, these portions of ITAAC Index No. 565 cannot be closed in conformance with the interpretation and understanding that "as-built" SSCs must be installed in their final operational location prior to ITAAC closure. Furthermore, the inspection of the Incore Thimble Assemblies with this interpretation of "as-built" do not meet the UFSAR Subsection 14.3.2.2 selection criteria for ITAAC that "the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load." Prior to fuel load, these attributes are verified by vendor inspections and testing; following the 52.103(g) finding, associated fuel loading and precritical testing will be performed in accordance with COL License Condition 2.D.(3).
>
> …An inspection is conducted to confirm that the seismic category I equipment identified in Table 2.5.5-1, the Class 1E Incore Thimble Assemblies, were manufactured per the qualified design. The inspection verifies the equipment make/model/serial number, as well as the as-designed anchorage point to the integrated grid assembly. An EQ

Reconciliation Report (EQRR) is completed to verify the seismic Category I equipment listed in the Table, including anchorage, is seismically bounded by the tested or analyzed conditions, IEEE Standard 344-1987, and NRC Regulatory Guide (RG) 1.100, Revision 2, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants.”

UFSAR Subsection 3.11.5 identifies that the environmental qualification (EQ) files developed by the reactor vendor are maintained as applicable for equipment and certain post-accident monitoring devices that are subject to a harsh environment. The contents of the qualification files are discussed in Section 3D.7. Appendix 3D, Subsection 3D.7.2.1 indicates that equipment is identified “by manufacturer, model or model series, and reference to other documents describing or depicting its construction, configuration, and modifications that are uniquely necessary after manufacture to its application in the AP1000 plant design.” Subsection 3D.7.2.2 goes on to address installation requirements, noting “So that the qualification represents the in-plant condition, the method of installation, as specified in Section 1.2 of Attachment A, is in accordance with the supplier's installation instructions. Differences unique to safety-related applications in the AP1000 design are included, with appropriate reference to drawings, technical manual supplements, or mandatory modification packages.”

In addition, Quality Control reviews the work package completion to confirm that the equipment is installed in a manner that is consistent with the as-tested/as-analyzed configuration.

SNC’s proposed change deletes ITAAC Index No. 565, item 2.i) because the “Incore Thimble Assemblies,” listed in COL, Appendix C, Table 2.5.5-1, are not located on the nuclear island until after fuel loading is authorized. This change proposes to remove the ITAAC requirement to inspect and verify that the location of the Incore Thimble Assemblies is on the nuclear island prior to the NRCs 10 CFR 52.103(g) finding, which aligns with the UFSAR 14.3.2.2 criteria, as previously mentioned, and its removal does not change any seismic qualifications for, or verification of, the Incore Thimble Assemblies. The Incore Thimble Assemblies will be located on the nuclear island following authorization to load fuel and will constitute part of the reactor system prior to operation. Thus, the deletion of ITAAC Index No. 565, item 2.i), is acceptable.

The staff reviewed the information in Enclosure 1 of LAR 21-001 and finds that the Incore Thimble Assemblies cannot be installed in their final location to support plant operation until after the 10 CFR 52.103(g) finding has been made and the initial fuel load of the reactor vessel. Thus, ITAAC Index No. 565, item 2.iii), cannot be completed in the IITAs’ “as-built” condition, and the “as-built” requirement should not have been included for this ITAAC. With the proposed revisions, the licensee would still be required to verify that the IITAs, including anchorage, are seismically bounded by the seismic qualification performed under ITAAC Index No. 565, item 2.ii). This seismic qualification explicitly considers the IITA installation method and location. Consequently, the ITAAC as revised provide sufficient assurance that the IITAs can withstand seismic design basis loads without loss of safety function. Additionally, the subsequent reactor vessel assembly and fuel load is adequately controlled and verified during the initial test program; this addresses installation of the IITAs. For these reasons, the licensee’s proposal to remove “as-built” from item 2.iii) is acceptable.

SNC’s proposed changes to ITAAC Index No. 565, item 3.a.ii) maintain the design functions of these systems and do not substantively change what the existing ITAAC is intended to verify, except that the ITAAC would be completed without the IITAs having been installed. ITAAC

Index No. 565, item 3.a.ii), cannot be completed in the IITAs' "as-built" condition, and the "as-built" requirement should not have been included for this ITAAC. With the proposed revisions, the licensee would still be required to verify that the IITAs, and the associated wiring, cables, and terminations, are bounded by the harsh environment qualification performed under ITAAC Index No. 565, item 3.a.i). As stated in the LAR and UFSAR, the qualification represents the in-plant condition and the method of installation. Consequently, the ITAAC as revised provide sufficient assurance that the IITAs can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function. Additionally, the subsequent reactor vessel assembly and fuel load is adequately controlled and verified during the initial test program; this addresses installation of the IITAs. The staff also finds that SNC's proposed changes do not delete any technical requirements, impact the ability of an SSC to perform its function, or impact safety and the reasonable assurance finding with respect to qualification of the components. Therefore, the EQ requirements defined in 10 CFR 50.49, GDC 2, and GDC 4 continue to be met. For these reasons, the licensee's proposal to remove "as-built" from item 3.a.ii) is acceptable.

Based on these findings the staff concludes that the LAR's proposed revisions to ITAAC Index No. 565 will continue to be sufficient to verify that the facility has been constructed and will operate in accordance with the license, the provisions of the AEA, and the Commission's rules and regulations. Therefore, within the scope of this license amendment, the NRC concludes that 10 CFR 52.97(b) is satisfied. Thus, SNC's proposed changes shown below are acceptable.

COL Appendix C Table 2.5.5-2 is revised (for ITAAC Index No. 565) as follows for Unit 3 (the Unit 4 COL is revised identically, except that the Amendment No. is 186):

Table 2.5.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

565	2.5.05.02.i	2. The seismic Category I equipment identified in Table 2.5.5-1 can withstand seismic design basis dynamic loads without loss of safety function.	i) Not used per Amendment No. 188 ~~Inspection will be performed to verify that the seismic Category I equipment identified in Table 2.5.5-1 is located on the Nuclear Island.~~ ii) (no changes) iii) Inspection will be performed for the existence of a report verifying that the ~~as-built~~ equipment including anchorage is seismically bounded by the tested or analyzed conditions.	i) Not used per Amendment No. 188 ~~The seismic Category I equipment identified in Table 2.5.5-1 is located on the Nuclear Island.~~ ii) (no changes) iii) A report exists and concludes that the ~~as-built~~ equipment including anchorage is seismically bounded by the tested or analyzed conditions.
		3.a) The Class 1E equipment identified in Table 2.5.5-1 as being qualified for a harsh environment can withstand the environmental conditions that would exist before, during, and following a design basis accident without loss of safety function, for the time required to perform the safety function.	i) (no changes) ii) Inspection will be performed of the ~~as-built~~ Class 1E equipment and the associated wiring, cables, and terminations located in a harsh environment.	i) (no changes) ii) A report exists and concludes that the ~~as-built~~ Class 1E equipment and the associated wiring, cables, and terminations identified in Table 2.5.5-1 as being qualified for a harsh environment are bounded by type tests, analyses, or a combination of type tests and analyses.

3.1.5 ITAAC Index No. 570 (2.5.05.03b)

COL Appendix C, Section 2.5.5, In-Core Instrumentation System, Design Description 3.b) states: [t]he Class 1E cables between the Incore Thermocouple elements and the connector boxes located on the integrated head package have sheaths." The design requirement is verified in ITAAC Index No. 570. For ITAAC 570, the design commitment is "[t]he Class 1E cables between the Incore Thermocouple elements and the connector boxes located on the integrated head package have sheaths," with an acceptance criterion of "[t]he as-built Class 1E cables between the Incore Thermocouple elements and the connector boxes located on the integrated head package have sheaths." To verify that the acceptance criterion is met, ITAAC Index No. 570 requires that "[i]nspection of the as-built system will be performed."

SNC in LAR 21-001 requested that Design Description 3.b) and ITAAC Index No. 570 (2.5.05.03b) be revised for clarity and to remove "as-built" from the ITAAC based on the following information:

> (page 6 of 19)…The incore instrumentation system consists of incore instrument thimble assemblies, which house fixed incore detectors, core exit thermocouple assemblies contained within an inner and outer sheath assembly, and associated signal processing and data processing equipment. There are 42 incore instrument thimble assemblies: each is composed of multiple fixed incore detectors and one thermocouple.
>
> (page 13 of 19) The core exit temperature sensor is located in the Incore Instrument Thimble Assembly (IITA) which is inserted into the core. The other end of the IITA is outside the reactor vessel head at the QuickLoc. The IITA connects to the head area cable assembly which is then routed through and around the Integrated Head Package (IHP), across the cable bridge, through the IHP cable rack assembly and connects with its matching cable mounted on the connector plate, which is part of the Operating Deck Connector Panel.
>
> …To give effect to the scope of this ITAAC expressed by NRC staff, the clarity changes include replacing the phrase "Incore Thermocouple elements" with "Core Exit Temperature sensors" for consistency with other ITAAC nomenclature, removing the "location" information and clarifying the scope by replacing "connector boxes" with "connector plates" consistent with terms used in the FSAR and replacing "system" with "Class 1E cables" so that the inspection language is consistent with the Design Commitment and the Acceptance Criteria.
>
> The ITAAC inspections for equipment which cannot be inspected in its final location include review of documentation such as the system design specifications, records of inspection of the components performed by the manufacturer prior to shipment to the plant site, Quality Release and Certificate of Conformance documentation, construction drawings, and completed construction records, including those performed on-site, to the extent possible given that the facility design does not allow certain components to be installed in their final operational configuration until after fuel is installed in the vessel.
>
> Design specifications require internal metallic sheaths that surround and separate the individual Class 1E thermocouple wires from non-Class 1E detector wires, which are contained within an external spiral wound sheath. The design specifications also include performance tests for overvoltage, insulation resistance, and continuity. Successful test results indicate that the sheaths protect against credible single faults between the Class 1E and non-Class 1E signals.

> The Quality Release and Certificate of Conformance document verifies the head area cable assembly acceptance test results. The Field Service Report document verifies the head area cable assemblies were installed on the integrated head package in accordance with design drawings and installation specifications issued for construction, and work package requirements. An additional Quality Release and Certificate of Conformance document verifies that the invessel Class 1E cables were installed within the incore thimble assemblies in accordance with design drawings and installation specifications and contains the incore thimble assemblies cable acceptance test results.
>
> The Class 1E cables between the Core Exit Temperature sensors (these sensors are located within the incore thimble assemblies) and the connector plates (as described above, these connectors are beyond the integrated head package on the operating deck connector panel) are inspected to verify that the design specification and installation specifications are satisfied, to enable each cable to convey the safety-related core exit thermocouple signals to the PMS.
>
> The inspections are performed and documented in accordance with manufacturer and vendor quality verification programs. The results of the inspections are documented in support of the ITAAC 2.5.05.03b Completion Packages. The inspections confirm that the Class 1E cables between the Core Exit Temperature sensors and the connector plates have sheaths.

The staff reviewed the information in Enclosure 1 of LAR 21-001 for the changes proposed to clarify Design Description 3.b) and ITAAC Index No. 570 and to remove “as-built” from the ITAAC’s required inspection and corresponding acceptance criteria.

As described in the LAR and in UFSAR Subsections 4.4.6.1, “Incore Instrumentation,” 9.1.4.2.2.2, “Phase II – Reactor Disassembly,” and 3.9.7.2, “Design Description,” the CET sensors are located within the IITAs located in the in-core thermocouple elements, and the head area cables are connected at the IHP connector “plates” and not “boxes.” Therefore, the staff finds the revised nomenclature is consistent with the terms used in the plant UFSAR and the proposed nomenclature changes are acceptable.

The LAR states that installation of the IITAs, which include the CET sensors and cables, requires fuel be installed in the reactor vessel. This means that the “as-built” requirement in the ITAAC, as applied to the CET sensors and associated Class 1E cables connecting to the IHP connector plates, is not consistent with the requirement in 10 CFR 52.103(g) that the NRC find that the acceptance criteria are met prior to operation (which includes fuel load), nor with the UFSAR Subsection 14.3.2.2 selection criteria for ITAAC that “the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load.” However, as described in the LAR, the IHP cable from the operating deck connector panel to the incore instrument QuikLoc assemblies, has already been inspected and verified in its “as-built” location (on the IHP), so the staff finds the proposed change does not materially affect the verification required of this cable. The staff also noted that the ITAAC’s method of completing acceptance criteria is a visual inspection of the Class 1E cables sheaths in the IITAs, which cannot be accomplished once the components have been installed. Therefore, applying the “as-built” provision to the ITAAC does not provide any substantial benefit or added assurance that is not already obtained by the inspections and testing performed at the vendor facilities for the IITAs.

Furthermore, the ITAAC process intended to verify the design requirements, including the separation provision between Class 1E and non-Class 1E components, is addressed in NUREG-1793, Revision 0, “Final Safety Evaluation Report Related to Certification of the

AP1000 Standard Design," dated September 2004. WCAP-17226, "Assessment of Potential Interaction between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000TM In-core Instrumentation System," Revision 2, addressed by analysis separation concerns of Class 1E and non-Class 1E cables within both the IITAs and head area cable assemblies, and was accepted by staff in NUREG-1793. This LAR does not affect the staff's evaluation in NUREG-1793 since the physical routing of those cables are not impacted. In addition, the LAR indicates that design specifications will be reviewed which contain performance tests for overvoltage, insulation resistance, and continuity to ensure presence of the required sheaths and that no credible faults can occur between the Class 1E and non-Class 1E cables in both the IITAs and the head area cable assembly.

Based on the above discussion, the staff finds that SNC's proposed changes do not delete or affect any applicable technical requirements, the method of verifying the sheaths on the Class 1E cables, or their design function. The staff also finds that the proposed changes do not impact the ability of the Class 1E cables to perform their safety functions, and that ITAAC Index No. 570 still verifies that the cables have sheaths. Specifically, the revised ITAAC will still be completed by inspection to verify that specified sheaths are present on the cables by confirming design specifications requiring performance tests for overvoltage, insulation resistance, and continuity to ensure that specified sheaths protect against credible single faults between the Class 1E and non-Class 1E cables. Additionally, the subsequent reactor vessel assembly and fuel load is adequately controlled and verified during the initial test program; this addresses installation of the IITAs. Based on these above findings, the staff concludes that SNC's proposed changes to ITAAC Index No. 570 will continue to be sufficient to verify that the facility has been constructed and will operate in accordance with the license, the provisions of the AEA, and the Commission's rules and regulations. Therefore, within the scope of this license amendment, the NRC concludes that 10 CFR 52.97(b) is satisfied. Thus, SNC's proposed changes to Design Description 3.b) and ITAAC Index No. 570 are acceptable.

COL Appendix C Table 2.5.5-2 is revised (for ITAAC Index No. 570) as follows:

Table 2.5.5-2
Inspections, Tests, Analyses, and Acceptance Criteria

No.	ITAAC No.	Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
570	2.5.05.03b	3.b) The Class 1E cables between the Core Exit Temperature sensors ~~Incore Thermocouple elements~~ and the connector plates ~~boxes located on the integrated head package~~ have sheaths.	Inspection of the ~~as-built system~~ Class 1E cables will be performed.	The ~~as-built~~ Class 1E cables between the Core Exit Temperature sensors ~~Incore Thermocouple elements~~ and the connector plates ~~boxes located on the integrated head package~~ have sheaths.

3.2 EVALUATION OF EXEMPTION

INTRODUCTION

The regulations in Section III.B of Appendix D to 10 CFR Part 52 require a holder of a COL referencing Appendix D to 10 CFR Part 52 to incorporate by reference and comply with the requirements of Appendix D, including certified information in Tier 1 of the generic AP1000 DCD. Exemptions from Tier 1 information are governed by the change process in Section

VIII.A.4 of Appendix D of 10 CFR Part 52. Because SNC has identified changes to plant-specific Tier 1 information, with corresponding changes to the associated COL Appendix C information resulting in the need for a departure, an exemption from the certified design information within plant-specific Tier 1 material is required to implement the LAR.

The Tier 1 information for which a plant-specific departure and exemption was requested is described above. The result of this exemption would be that SNC could implement the requested modifications to Tier 1 information, with corresponding changes to COL Appendix C. Pursuant to the provisions of 10 CFR 52.63(b)(1), an exemption from elements of the design as certified in the 10 CFR Part 52, Appendix D, design certification rule is requested for the specified Tier 1 change described and justified in LAR 21-001. This is a permanent exemption limited in scope to the particular Tier 1 information specified.

As stated in Section VIII.A.4 of Appendix D to 10 CFR Part 52, an exemption from Tier 1 information is governed by the requirements of 10 CFR 52.63(b)(1) and 52.98(f). Additionally, Section VIII.A.4 of Appendix D to 10 CFR Part 52 provides that the Commission will deny a request for an exemption from Tier 1 if it finds that the design change will result in a significant decrease in the level of safety otherwise provided by the design. Pursuant to 10 CFR 52.63(b)(1), the Commission may, upon application by an applicant or licensee referencing a certified design, grant exemptions from one or more elements of the certification information, so long as the criteria given in 10 CFR 52.7, which in turn, references 10 CFR 50.12, are met. Also, the Commission must consider whether the special circumstances which are defined by 10 CFR 50.12(a)(2) outweigh any potential decrease in safety due to reduced standardization caused by the exemption.

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR Part 52. 10 CFR 52.7 further states that the Commission's consideration will be governed by 10 CFR 50.12. In accordance with 10 CFR 50.12, an exemption may be granted when: (1) the exemption is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security; and (2) special circumstances are present. 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for NRC to consider granting an exemption request. The licensee stated that the requested exemption meets the special circumstances of 10 CFR 50.12(a)(2)(ii). That subsection defines special circumstance when "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." The staff's analysis of each of the required exemption criteria is presented below.

3.2.1 AUTHORIZED BY LAW

This exemption would allow the licensee to implement the amendment described above. This is a permanent exemption limited in scope to particular Tier 1 information. Subsequent changes to this plant-specific Tier 1 information, and corresponding changes to Appendix C, or any other Tier 1 information would be subject to the exemption process specified in Section VIII.A.4 of Appendix D to 10 CFR Part 52 and the requirements of 10 CFR 52.63(b)(1). As stated above, 10 CFR Part 52, Appendix D, Section VIII.A.4 allows the NRC to grant exemptions from one or more elements of the Tier 1 information. The NRC staff has determined that granting the licensee's proposed exemption will not result in a violation of the AEA or the Commission's regulations. Therefore, as required by 10 CFR 50.12(a)(1), the exemption is authorized by law.

3.2.2 NO UNDUE RISK TO PUBLIC HEALTH AND SAFETY

As discussed above in the technical evaluation, the proposed changes comply with the NRC's substantive safety regulations. Therefore, there is no undue risk to the public health and safety.

3.2.3 CONSISTENT WITH COMMON DEFENSE AND SECURITY

The proposed exemption would allow changes as described above in the technical evaluation, thereby departing from the AP1000 certified (Tier 1) design information. The changes do not alter or impede the design, function, or operation of any plant structure, system, or component (SSC) associated with the facility's physical or cyber security, and therefore does not affect any plant equipment that is necessary to maintain a safe and secure plant status. In addition, the changes have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security is not impacted by this exemption.

3.2.4 SPECIAL CIRCUMSTANCES

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present, in part, whenever application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

Special circumstances are present in the particular circumstances discussed in LAR 21-001 because the application of the specified Tier 1 information does not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. As discussed above, the changes to ITAAC Index Nos. 75, 515, 565, and 575 regarding the location of components on the nuclear island and the performance of certain ITAAC verifications in the "as-built" condition only have a practical effect for components that cannot be installed before fuel load. Because these requirements are intended to reflect the installed condition of the components and because ITAAC must be satisfied before fuel load, the application of the requirements to those components would not serve the underlying purpose of these requirements. In addition, the remaining ITAAC verifications combined with post-fuel load verifications serve the underlying purpose of the ITAAC to verify that the installed components meet the design requirement. The proposed changes also eliminate an unnecessary requirement by deleting a functional arrangement ITAAC (Index No. 68) that is either duplicated by other ITAAC (for those components that can be installed before fuel load) or cannot be completed as stated and will be verified by other means (for those components that cannot be installed before fuel load). The proposed changes also revise the nomenclature for certain components to align with the current licensing basis. The proposed changes do not adversely affect any function or feature used for the prevention and mitigation of accidents or their safety analyses. The changes described above do not impact the ability of any SSC to perform its function or negatively impact safety. The revisions to the Tier 1 information and corresponding changes to Appendix C will continue to meet applicable regulatory requirements. Therefore, for the above reasons, the staff finds that the special circumstances required by 10 CFR 50.12(a)(2)(ii) for the granting of an exemption from the Tier 1 information exist.

3.2.5 SPECIAL CIRCUMSTANCES OUTWEIGH REDUCED STANDARDIZATION

This exemption would allow the implementation of changes to COL Appendix C and corresponding Tier 1 information based on the LAR. The effect of the changes in LAR 21-001 is to (1) remove requirements to complete certain ITAAC in the "as-built" location for components that cannot be installed prior to the 10 CFR 52.103(g) finding, (2) eliminate a functional arrangement ITAAC that is duplicated by other ITAAC, with the exception of those components that cannot be installed before the 10 CFR 52.103(g) finding, (3) eliminate a requirement to verify equipment is located on the "nuclear island," when that equipment cannot be installed before the 10 CFR 52.103(g) finding, and (4) revise nomenclature used in the ITAAC to improve clarity. Also, for those components that cannot be installed before the 10 CFR 52.103(g) finding, there are post-fuel load requirements that adequately address verifications regarding the locations and installed condition of these components. These changes, as modified by NRC, do not alter any technical requirements or impact the ability of an SSC to perform its function. The design functions of the system associated with this request will continue to be maintained because the associated revisions to the Tier 1 information support the design function of the associated SSCs. The only practical effect of the changes is to briefly delay until after fuel load the verification of the location and installed condition of components that cannot be installed before fuel load. Therefore, there is no meaningful reduction in standardization, and the safety impact that may result from any reduction in standardization is minimized, because the proposed change does not result in a reduction in the level of safety. In addition, the special circumstances discussed above outweigh any potential decrease in safety from reduced standardization because the only practical effect of the changes is to eliminate a contradiction between the ITAAC as written and the requirement to complete ITAAC before scheduled fuel load. Thus, the staff finds that 10 CFR Part 52.63(b)(1) is satisfied.

3.2.6 NO SIGNIFICANT REDUCTION IN SAFETY

The exemption request proposes to depart from the certified design by allowing changes discussed above in the technical evaluation. These changes, as modified by NRC, do not alter any technical requirements or impact the ability of an SSC to perform its function. The changes will not modify the design or modify the operation of any systems or equipment, there are no new failure modes introduced by these changes, and the level of safety provided by the current structures, systems, and components will be unchanged. As discussed above, the only practical effect of the changes is to briefly delay until after fuel load the verification of the location and installed condition of components that cannot be installed before fuel load. Therefore, based on the foregoing reasons and as required by 10 CFR Part 52, Appendix D, Section VIII.A.4, the staff finds that granting the exemption would not result in a significant decrease in the level of safety otherwise provided by the design.

For the reasons given above, the standards for an exemption from the specified Tier 1 information have been satisfied.

3.3 SUMMARY

In LAR 21-001, SNC proposed to make changes that would affect the COL Appendix C and corresponding PS-DCD Tier 1 information. None of the above proposed changes represent changes to the design or operation of the plant. Therefore, the requirements defined in 10 CFR 50.49, 10 CFR 50.55a, GDC 2, GDC 4, and GDC 24 continue to be met. The staff finds that with the changes, as modified by the NRC, the ITAAC continue to be sufficient to verify that the facility has been constructed and will be operated in accordance with the license, the AEA, and NRC rules and regulations. Therefore, within the scope of this license amendment, the NRC finds that 10 CFR 52.97(b) is satisfied. The NRC documented its review of the above changes in Section 3.1 of this safety evaluation and finds the changes acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC staff published its proposed no significant hazards consideration determination in the *Federal Register* on September 3, 2021 (86 FR 49572). Under its regulations, the Commission may issue an amendment and make it immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, where it has made a final determination that no significant hazards consideration is involved.

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

An evaluation of the issue of no significant hazards consideration is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed revisions have been found to continue to provide the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The affected system is not an initiator of any accident analyzed in the Updated Final Safety Analysis Report (UFSAR), nor do the changes involve an interface with any SSC accident initiator or initiating sequence of events, and thus, the probabilities of the accidents evaluated in the UFSAR are not affected. The proposed changes do not involve a change to any mitigation sequence or the predicted radiological releases due to postulated accident conditions; thus, the consequences of the accidents evaluated in the UFSAR are not affected.

   The UFSAR describes the analyses of various design basis transients and accidents to demonstrate compliance of the design with the acceptance criteria for these events. The acceptance criteria for the various events are based on meeting the relevant regulations and general design criteria and are a function of the anticipated frequency of occurrence of the event and potential radiological consequences to the public. The revised ITAAC maintains the plant conditions, and thus, maintains the frequency designation and consequence level as previously evaluated.

   Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed revisions have been found to continue to confirm the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The proposed revisions do not change the function of the related systems, and thus, the changes do not introduce a new failure mode, malfunction or sequence of events that could adversely affect safety or safety-related equipment.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed revisions have been found to continue to provide the required functional capability of the safety systems for previously evaluated accidents and anticipated operational occurrences. The proposed revisions do not change the function of the related systems nor significantly affect the margins provided by the systems. No safety analysis or design basis acceptance limit/criterion is challenged or exceeded by the requested changes.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above evaluation, the staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations in 10 CFR 50.91(b)(2), on September 10, 2021, the Georgia State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, "Standards for Protection Against Radiation." The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding as published in the *Federal Register* on September 3, 2021 (86 FR 49572). And as stated above, the NRC is making a final no significant hazards consideration determination for this license amendment. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

Because the exemption is necessary to allow the changes proposed in the license amendment, and because the exemption does not authorize any activities other than those proposed in the license amendment, the environmental consideration for the exemption is identical to that of the license amendment. Accordingly, the exemption meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the exemption.

7.0 CONCLUSION

The staff has determined that pursuant to Section VIII.A.4 of Appendix D to 10 CFR Part 52, the exemption (1) is authorized by law, (2) presents no undue risk to the public health and safety, (3) is consistent with the common defense and security, (4) presents special circumstances, and (5) does not significantly reduce the level of safety at the licensee's facility. Also, the staff has determined that the special circumstances for the exemption outweigh any decrease in safety that may result from a reduction in standardization caused by the exemption. Therefore, the staff grants the licensee an exemption from the Tier 1 information requested by the licensee.

The staff has concluded, based on the considerations discussed in Section 3.1 that there is reasonable assurance that: (1) the health and safety of the public will not be endangered by the proposed changes, (2) the changes are in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, the staff finds the changes proposed in this license amendment acceptable.

8.0 REFERENCES

1. Southern Nuclear Operating Company, Vogtle Electric Generating Plant Units 3 and 4, "Request for License Amendment and Exemption: Clarification of ITAAC Regarding Invessel Components (LAR 21-001)," August 24, 2021 (ADAMS Accession No. ML21236A305).

2. AP1000 Design Control Document, Revision 19, June 13, 2011 (ADAMS Accession No. ML11171A500).

3. Combined License NPF-91 for Vogtle Electric Generating Plant Unit 3, Southern Nuclear Operating Company (ADAMS Accession No. ML14100A106).

4. Combined License NPF-92 for Vogtle Electric Generating Plant Unit 4, Southern Nuclear Operating Company (ADAMS Accession No. ML14100A135).

5. Vogtle Electric Generating Plant Units 3 and 4, Updated Final Safety Analysis Report, Revision 10, June 14, 2021 (ADAMS Accession No. ML21179A130).

6. Nuclear Energy Institute (NEI) 08-01, "Industry Guideline for the ITAAC Closure Process Under 10 CFR Part 52," Revision 5 – Corrected, dated June 30, 2014 (ADAMS Accession No. ML14182A158).

7. Regulatory Guide 1.215, "Guidance for ITAAC Closure Under 10 CFR Part 52," Revision 2, July 20, 2015 (ADAMS Accession No. ML15105A447).

8. WCAP-17226, Revision 2, “Assessment of Potential Interaction between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000TM In-core Instrumentation System,” dated August 25, 2010 (ADAMS Accession No. ML102390521).

9. NUREG-1793, Supplement 2, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design Docket No. 52-006,” dated August 5, 2011 (ADAMS Accession No. ML112061231).

10. IEEE 603-1991, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations.”