SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

RELATED TO AMENDMENT NO. 13 TO

COMBINED LICENSE NOS. NFP-91 AND NFP-92

SOUTHERN NUCLEAR OPERATING COMPANY

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNCIPAL ELECTRIC AUTHORITY OF GEORGIA

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 February 24, 2012, and as supplemented by letters listed below, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR)-12-002, to change the technical specifications (TS) for Vogtle Electric Generating Plant, Units 3 and 4 (VEGP). The initial application letter is available in the Agencywide Documents Access and Management System (ADAMS) with Accession No. ML120650172. The proposed license amendments would upgrade the VEGP TS to improve operator usability by more closely aligning the TS with the latest form and content of standard technical specifications (STS). In particular, the proposed changes would result in closer alignment with the guidance in NUREG-1430, “STS Babcock & Wilcox Plants” and NUREG-1431, “STS Westinghouse Plants,” as updated by the Nuclear Regulatory Commission (NRC) approved generic changes; and the guidance in TSTF-GG-05-01, “Writer's Guide for Plant-Specific Improved Technical Specifications,” Revision 1, dated June 2005 (the Writer’s Guide).

The following supplemental letters provided additional information including some additional changes to the originally submitted TS changes that clarified the initial application, without expanding the scope of the application as originally noticed, or affecting the NRC staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* on May 29, 2012 (77 FR 31662):

- Letter dated October 4, 2012 (ADAMS Accession No. ML12286A360)
- Letter dated December 7, 2012 (ADAMS Accession No. ML12346A053)

During the course of its review, the NRC staff held a series of telephone conference calls and meetings with SNC that were open to the public. These public meetings and conference calls served to clarify the proposed changes to the VEGP TS with respect to the guidance in the STS.

The proposed changes to the VEGP TS are based on (1) Revision 4 of NUREG-1430 and NUREG-1431, dated April 30, 2012; (2) the "Final Policy Statement on Technical Specification

Improvements for Nuclear Power Reactors" (final policy statement), published on July 22, 1993 (58 FR 39132); (3) 10 CFR 50.36, "Technical specifications"; and (4) the Writer's Guide.

Hereinafter, existing VEGP TS are referred to as current TS or CTS, VEGP TS revised as proposed are referred to as improved TS or ITS[1], and standard TS, as given in NUREG-1430 and NUREG-1431, Revision 4, are referred to as standard TS or STS. Corresponding TS bases are referred to as CTS bases, ITS bases, and STS bases, respectively. For convenience, Attachment 1 to this safety evaluation (SE) provides a list of acronyms and initializations used in this SE.

2.0 BACKGROUND

Part 52 of Title 10 of the Code of Federal Regulations (10 CFR Part 52), Subpart C—Combined Licenses, sets out the requirements and procedures applicable to Commission issuance of combined licenses for nuclear power facilities. Sections 52.79(a) and (d) set out many of the requirements for the contents of a COL application that references a standard design certification. On February 10, 2012, the first (reference) combined licenses (R-COLs) were issued by the NRC under 10 CFR Part 52, Subpart C, to SNC for VEGP. SNC's R-COL applications for VEGP referenced the Westinghouse AP1000 design certification rule, Appendix D to 10 CFR Part 52. The plant-specific TS, which were issued with the VEGP R-COLs, consist of the VEGP site-specific TS and the AP1000 generic TS (Chapter 16 of Design Control Document (DCD) Revision 19). The AP1000 generic TS were modeled after Revision 2 of NUREG-1431, dated June 30, 2001, and were incorporated by reference into the VEGP plant-specific TS.

Content of LAR-12-002

The LAR letter has five enclosures:

- Enclosure 1 Basis for Proposed Change
- Enclosure 2 Technical Specifications Markup Pages
- Enclosure 3 Clean-Typed Technical Specifications Pages
- Enclosure 4 Clean-Typed Technical Specifications Bases Pages (for information only)
- Enclosure 5 List of Regulatory Commitments

Enclosure 1 contains a summary description of the LAR and discusses SNC's motivation for proposing changes to the CTS. Namely, improving the usability of TS by control room operators by clarifying and simplifying the TS requirements to be more consistent with the most recent revision of NUREG-1431, with the exception of instrumentation requirements, which are

[1] Use of the initialization "ITS" is an editorial convenience and does not imply that this amendment is a conversion from the current plant-specific TS to the improved plant-specific TS based on the improved standard TS; the current plant-specific TS for Vogtle Electric Generating Plant, Units 3 and 4, already conform to the improved standard TS.

extensively reformatted according to the presentation in NUREG-1430. Section 3.3 of the Babcock & Wilcox STS follows the general approach of separating instrumentation Functions (i.e., sensors and associated signal processing, such as analog to digital conversion), Manual Actuation Functions (i.e., switches), and Automatic Actuation Logic Functions into separate Specification subsections. This approach allows splitting the initial actions from the default actions in the ACTIONS table of instrumentation Specification subsections that include a table listing the LCO-required Functions, and associated operability, applicability, action, and surveillance requirements.

- Initial actions are common to all of an LCO's required Functions and typically provide for bypassing or tripping an inoperable channel (or division); and bypassing one inoperable channel and tripping the other inoperable channel when there are two inoperable channels for one or more Functions.

- Upon failure to meet one of the initial actions (e.g., such as restoring the channel to operable status within the specified completion time), or when more than two of four channels (or, one of two channels) are inoperable (i.e., a loss of function condition), default actions typically direct consulting the LCO's function table for the specified follow-up actions condition that must be entered. These conditions are specified for each Function and generally specify exiting the applicability for that Function.

Enclosure 1 has six attachments. Attachments 1 through 5 include a detailed description and technical evaluation of each proposed CTS change within the scope of each designated "description of change" (DOC). The following are the five types of CTS changes:

A Administrative - Changes to the CTS that result in no new requirements and no changes in operational restrictions and flexibility. (Enclosure 1, Attachment 1)

M More Restrictive - Changes to the CTS that result in new requirements, added operational restrictions, or reduced flexibility. (Enclosure 1, Attachment 2)

R Relocation - Changes to the CTS that relocate LCO requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii) to licensee-controlled documents. (Enclosure 1, Attachment 3)

D Detailed Information Removed - Changes to the CTS that either eliminate detail or relocate detail to licensee-controlled documents. Typically, this involves details of system design and system description including design limits, or procedural details for meeting TS requirements. (Enclosure 1, Attachment 4)

L Less Restrictive - Changes to the CTS that result in elimination of requirements, reduced operational restrictions, or added flexibility. (Enclosure 1, Attachment 5)

Enclosure 1, Attachment 6 provides no significant hazards considerations (NSHC) evaluations in support of the CTS changes. Except for DOCs of type L, a single generic NSHC evaluation is provided for all the DOCs of each CTS change type. Each type L DOC includes a separate specific NSHC evaluation.

The DOCs are identified by an alpha-numeric designation formatted in accordance with change types listed above followed by a sequential number (e.g., A001, M01, R1, D01, L01, etc.) with one, two, or three digits according to the total number of DOCs proposed for each type.

Enclosure 2 provides a markup of affected CTS pages; CTS pages with no changes are not included. Enclosure 3 provides the clean-typed CTS pages as revised (i.e., ITS pages); again, CTS pages with no changes are not included.

Enclosure 4 provides (for information only) a draft of the clean-typed CTS bases pages as revised (i.e., ITS bases pages), which updates the CTS bases to be consistent with the CTS changes proposed in the LAR, and to make the rationale and intent of each ITS requirement clearer and easier to understand. Changes to the bases are controlled by Specification 5.5.6, "TS Bases Control Program," and are not within the scope of the LAR. However, the NRC staff did review the ITS bases pages to verify that they are consistent with the CTS changes. As before, CTS bases pages with no changes are not included.

Enclosure 5 provides a list of regulatory commitments associated with LAR-12-002. These regulatory commitments by SNC are discussed in Section 5.0 of this SE.

3.0 REGULATORY EVALUATION

Background Regarding the Development of Improved Standard Technical Specifications

Section 182a of the Atomic Energy Act (the Act) requires that applicants for nuclear power plant operating licenses shall state:

> [S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization . . . of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TS. In doing so, the Commission placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences. As recorded in the Statements of Consideration for Section 50.36, "Technical Specifications for Facility Licenses; Safety Analysis Reports" (33 FR 18610, December 17, 1968), the Commission noted that applicants were expected to incorporate into their TS "those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Pursuant to 10 CFR 50.36, TS for nuclear reactors are required to include items in the following categories: (1) safety limits, and limiting safety system settings; (2) limiting conditions for operation (LCOs)—includes remedial actions; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a nuclear reactor plant's TS.

On February 6, 1987, the Commission issued an interim policy statement on TS improvements, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788). During the period from 1989 to 1992, industry PWR and BWR owners groups and the NRC staff developed improved STS (e.g., NUREG-1431 and NUREG-1430) that would establish model TS based on the Commission's policy for each primary reactor type. In addition, representatives from the NRC, nuclear reactor plant licensees, and industry owner's groups developed generic administrative and editorial guidelines in the form of a Writer's Guide[2] for preparing TS, which gives appropriate consideration to human factors engineering principles and was used throughout the development of plant-specific improved TS.

In September 1992, the Commission issued NUREG-1431 and NUREG-1430, Revision 0, which were developed using the guidance and criteria contained in the Commission's interim policy statement. The improved standard TS in NUREG-1431 and NUREG-1430 were established as models for developing plant-specific improved TS in general for Westinghouse and Babcock & Wilcox plants, respectively. The improved STS reflect the results of a detailed review of the application of the interim policy statement LCO selection criteria to generic system functions. The NRC published these results—known as the split report (ADAMS accession No. ML11264A057)—in a May 9, 1988, letter from T. E. Murley (NRC) to the nuclear steam supply system (NSSS) vendor owner's groups (e.g., R. A. Newton of the Westinghouse Owners Group and W. S. Wilgus of the Babcock & Wilcox Owners Group). The split report provides the results of the NRC staff's review of the NSSS vendor owners groups' application of the Commission's interim policy statement LCO selection criteria to the LCOs in the previous versions of standard TS (e.g., NUREG-0452, "Standard TS, Westinghouse Plants," and NUREG-0103, "Standard TS, Babcock and Wilcox Plants"). The specifications and associated bases in the improved standard TS NUREGs, Volume 1 and Volume 2, respectively, also reflect the results of extensive discussions concerning various drafts of improved standard TS so that the application of the TS LCO selection criteria and the writer's guide would consistently reflect detailed system configurations and operating characteristics for all reactor designs. As such, the improved standard TS bases presented in Volume 2 of NUREG-1431 and NUREG-1430 provide an abundance of generic information regarding the extent to which the improved standard TS present requirements that are necessary to protect public health and safety.

On July 22, 1993, the Commission issued its final policy statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Act and 10 CFR 50.36. The final policy statement described the safety benefits of the improved standard TS and encouraged operating reactor licensees to use the improved standard TS as the basis for license amendments to upgrade existing plant-specific TS requirements, including complete conversions to improved plant-specific TS based on the improved standard TS. In addition, the final policy statement gave guidance for evaluating the required scope of the TS and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TS (that is, the interim policy statement's LCO selection criteria, which were updated in the final policy statement to clarify the scope of Criterion 2—to also include "design feature, or operating restriction"—and to enumerate Criterion 4). The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TS, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

> [T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition

[2] The Writer's Guide was first published as a formal document in February of 1993 by the Nuclear Management and Resources Council, Inc. (NUMARC), as NUMARC 93-03, "Writer's Guide for the Restructured Technical Specifications." The current version of the writer's guide was published in June of 2005 by the PWR and BWR owners groups' Technical Specifications Task Force (TSTF) as TSTF GG-05-01, "Writer's Guide for Plant-Specific Improved Technical Specifications," Revision 1.

> of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the final policy statement should be retained in the TS; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents. The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four LCO selection criteria (also referred to as TS screening criteria), as stated in 10 CFR 50.36(c)(2)(ii) subparagraphs (A), (B), (C), and (D), are as follows:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4 A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The proposed amendment to upgrade the CTS is consistent with the Commission's final policy statement in that it further enhances the clarity and usability of the plant-specific TS for VEGP, and more closely conforms to the latest revision of the STS.

<u>Evaluation</u>

The current plant-specific TS are a condition of the combined licenses for VEGP and consist of the generic TS and bases in Chapter 16 of the AP1000 DCD, Revision 19, and the site-specific TS and bases—information supplied by the combined license applicant in place of bracketed information placeholders in the generic TS and bases. The AP1000 DCD contains the Tier 1, Tier 2, and Tier 2* information, and the generic TS and bases that are incorporated by reference in "Appendix D to Part 52—Design Certification Rule for the AP1000 Design."

Part 52 requirements for changing information in a combined license holder's plant-specific FSAR and TS:

Section 52.98(c) states in part, "If a combined license references a certified design, then—(1) Changes to or departures from information within the scope of the referenced design certification rule are subject to the applicable change processes in that rule"; since the generic TS are within the scope of the design certification rule, changes to the plant-specific TS are governed by that rule. In Appendix D to Part 52, Section VIII, "Processes for Changes and Departures," Subsection C, "Operational Requirements," paragraph 6 states, "After Issuance of a license, the generic TS have no further effect on the plant-specific TS. Changes to the plant-specific TS will be treated as license amendments under 10 CFR 50.90."

Section 52.98(c) also states in part, "If a combined license references a certified design, then— (2) Changes that are not within the scope of the referenced design certification rule are subject to the applicable change processes in 10 CFR Part 50, unless they also involve changes to or noncompliance with information within the scope of the referenced design certification rule. In these cases, the applicable provisions of this section and the design certification rule apply." Because the technical specifications are part of the license, the Part 50 change process that applies to site-specific information in the plant-specific TS is that specified by 10 CFR 50.90.

Section 50.90 states, in part, "Whenever a holder of a license, including … a combined license … desires to amend the license, application for an amendment must be filed with the Commission, as specified in Section 50.4 and 52.3 … as applicable, fully describing the changes desired …"

Part 52 requirements for changes to information relocated from plant-specific TS to licensee-controlled documents:

<u>Scope of relocated TS requirements in the LAR</u>

The scope of the TS upgrade LAR includes relocation of selected details to the TS bases or the final safety analysis report (FSAR); the licensee asserts that the affected Specification subsections do not need these details to ensure adequate protection. The scope of the TS upgrade LAR also includes relocation of two entire Specification subsections (along with the associated TS bases) to the technical requirements manual (TRM), a document that the licensee commits to incorporate by reference into the FSAR. The licensee asserts that the relocated Specification subsections do not need to be in TS to ensure adequate protection because the specified system functions do not satisfy any of the LCO selection criteria.

<u>Control of TS requirements and details in TS requirements relocated to the FSAR or TRM</u>

The plant-specific FSAR is a licensee-controlled document containing Tier 1, Tier 2, Tier 2* and site-specific information. Changes to the FSAR are governed by the change processes specified by 10 CFR 52.98(c), as noted above. In particular:

- For changes to information in the FSAR that is within the scope of the referenced design certification (DC) rule, Appendix D to Part 52, Section VIII, "Processes for Changes and Departures," Subsection A addresses changes to Tier 1 information and Subsection B addresses changes to Tier 2 and Tier 2* information.

- For changes to information in the FSAR that is not within the scope of the referenced DC rule (site-specific information), Section 52.98(c)(2) states, "Changes that are not within the scope of the referenced design certification rule are subject to the applicable change processes in 10 CFR Part 50...” The Part 50 change process that applies to site-specific information in the FSAR is 10 CFR 50.59.

Whether within the scope of the DC or not, and depending upon the specifics of the proposed change to information in the FSAR or TRM, NRC prior approval may be required.

In its response to **RAI 16-32, Issue 1**, the licensee asserted that “the material [LCO 3.9.5 and LCO 3.9.6] being relocated [to the TRM] is considered site-specific FSAR content, and is not within the scope of the referenced design certification rule.” The NRC staff recognizes that Section 50.59 would apply to changes that strictly only affect information in the TRM. However, the relocated information from the plant-specific TS and TS bases is based on information from both the “site-specific” and DCD content of the FSAR. Therefore, to maintain consistency between the FSAR and the TRM, as required by Section 52.98(c)(2), the Part 52, Appendix D, Section VIII change process must be used for any change to the TRM that includes a change to design certification information.

More generally, before a licensee can determine the applicable change process for information (both details and entire Specifications) relocated from the plant-specific TS to the FSAR or TRM, it must have first categorized the potentially affected information in the FSAR, which is related to the relocated information in the FSAR or TRM, as either Tier 1, Tier 2, or Tier 2* (within DC scope), or site-specific (not within DC scope).

Control of details in TS requirements that are relocated to the TS bases

The TS bases are a licensee-controlled document. For changes to information (just details) relocated from the plant-specific TS to the TS bases, the applicable change control process is VEGP TS 5.5.6, “TS Bases Control Program.” The provisions of this program include reference to Section 50.59 as the applicable process for changes to the TS bases that involve changes to related information in the FSAR. By referencing Section 50.59, this program will ensure that TS bases changes affecting only site-specific information in the FSAR will be adequately controlled. However, the NRC staff believes that if the related information is not site-specific, but is either Tier 1, Tier 2 or Tier 2*, then 10 CFR 52.98(c) would determine the change process that must be applied to the TS bases change, regardless of the change process specified in the TS bases control program.

Therefore, the NRC staff believes that the Specification 5.5.6 should reference Section 52.98 in place of Section 50.59. However, resolution of this issue is beyond the scope of LAR 12-002. Also, this issue does not affect the NRC staff’s conclusion in Section F of Part 4.0 of this safety evaluation (SE) that relocated TS information will be adequately controlled by the licensee using processes specified by applicable regulations in Part 50 and Part 52, and by Specification 5.5.6.

Part 4.0 of this SE explains the NRC staff’s determination that the VEGP ITS based on STS are consistent with the VEGP current licensing basis, the guidance in the final policy statement, and the requirements in 10 CFR 50.36 and 10 CFR 50.36a.

4.0 TECHNICAL EVALUATION

In its review of the LAR for upgrading the VEGP TS, the NRC staff evaluated five kinds of CTS changes, as defined by SNC in Enclosure 1, Attachments 1 through 5, of the LAR letter. The NRC staff also confirmed that existing regulatory requirements are adequate for controlling future changes to requirements that are removed from the CTS and placed in licensee-controlled documents.

A summary description of the DOCs as presented in the LAR, are listed and described in five tables attached to this SE, one table for each change type; i.e., Tables A, M, R, D, and L. These tables provide a summary description of the proposed changes to CTS, references to the specific CTS requirements that are being changed, and the resulting specific ITS requirements that incorporate the changes. The tables summarize the changes being made to CTS, and may not provide every detail of each change. But because of the similarities between the changes in each table, the reasons why the staff finds a table's changes acceptable apply to every change listed in the table, with the exception of Table D, Table R, and Table L. In the staff's discussion of Tables D and L each category of change is addressed. In the discussion of Table R each relocated Specification subsection is addressed. All details of the actual changes are provided in the licensee's LAR letter and supplemental letters. Many of the DOCs address changes in two or more CTS and ITS sections. Since each table consists of a set of 14 or more sub-tables corresponding to ITS Sections 1.0, 2.0, 3.0, 3.1 through 3.9, 4.0, and 5.0, each table enumerates the individual changes of each DOC to facilitate their identification when referencing this SE.

The NRC staff's evaluation and additional description of the kinds of changes to the CTS requirements listed in Tables A, M, R, D, and L attached to this SE are presented in Sections A through E below, as follows:

Section A - Administrative Changes
Section B - More Restrictive Changes
Section C - Relocation Changes
Section D - Detail Removed Changes
Section E - Less Restrictive Changes

Section F describes the regulatory control of specifications, requirements, and detailed information relocated from CTS to licensee-controlled documents. Section G provides a summary of the results of the technical evaluation.

A. Administrative Changes

Administrative changes are intended to incorporate human factors principles into the form and structure of the ITS so that plant operations personnel can use the ITS more easily. These changes may be editorial in nature and may involve reformatting or reorganizing CTS requirements, but do not alter the technical content, operational restrictions, or operational flexibility of CTS provisions. Every section of the ITS reflects this type of change.

In order to ensure consistency, the NRC staff and SNC have used the guidance in the STS and the Writer's Guide in reformatting and reorganizing CTS requirements, and in making editorial improvements and other administrative changes. Among the changes proposed by SNC and found acceptable by the NRC staff are:

- Identifying plant-specific wording for system names, etc.;
- Splitting up requirements currently grouped under a single current specification and moving them to more appropriate locations in two or more specifications of the ITS;

- Combining related requirements, which are presented in separate specifications of the CTS, into a single specification of the ITS;
- Presentation changes that involve rewording or reformatting, which improves clarity (including moving an existing requirement to another location within the TS) and conformance with the Writer's Guide, but no technical changes; and
- Deletion of redundant CTS requirements that exist elsewhere in the CTS.

Table A attached to this SE lists the administrative changes being made to the VEGP CTS. Table A is organized in ITS section order and includes the following:

- DOC identifier;
- Summary description of the administrative changes;
- Change type; and
- Reference to affected CTS and ITS requirements.

During the review of the administrative changes, the NRC staff issued requests for additional information (RAIs) to seek further clarification from the licensee to ensure completeness and accuracy of the proposed changes. The NRC staff evaluated the licensee's responses to these RAIs, and found the enhanced justifications consistent with the original proposal and acceptable, except as discussed below:

> As described in DOC A018, the licensee proposed (1) splitting each of current SR 3.2.1.1 ("Verify $F_Q{}^C(Z)$ within limit.") and SR 3.2.1.2 ("Verify that $F_Q{}^W(Z)$ within limits.") into two separate SRs (one pair with the first frequency; the second pair with the second and third frequencies), and (2) revising the associated SRs table Notes as surveillance column Notes to improve their usability by plant operators.
>
> In **RAI 16-33**, the NRC staff requested that the licensee provide clarification of the intended meaning of the new surveillance column Note ("Not required to be performed if the On-Line Power Distribution Monitoring System (OPDMS) was monitoring parameters upon exceeding 75% RTP.") in new SR 3.2.1.1 and SR 3.2.1.2 because the new Note appeared to be inconsistent with existing SRs table Note 1 ("During power escalation at the beginning of each cycle, THERMAL POWER may be increased until a power level for extended operation has been achieved at which a power distribution map is obtained.") and Note 2 ("If the OPDMS becomes inoperable while in MODE 1 these surveillances must be performed within 31 days of the last verification of OPDMS parameters.") for the existing surveillances and their associated frequencies. In its response letter dated October 4, 2012, the licensee explained that the new surveillance column Note is needed to avoid confusing plant operators when implementing new SR 3.2.1.1 and SR 3.2.1.2.
>
> The NRC staff finds this response acceptable because the new surveillance column Note is consistent with and clarifies the intent of the first Frequency ("Once after each refueling prior to THERMAL POWER exceeding 75% RTP") of existing SR 3.2.1.1 and SR 3.2.1.2.
>
> However, during the review of this response, the NRC staff identified an additional concern regarding the application of the other new surveillance column Note ("Not required to be performed until 31 days after the last verification of OPDMS parameters.") to the first frequency ("Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which [$F_Q{}^C(Z)$] [$F_Q{}^W(Z)$] was last verified.") of new

SR 3.2.1.3 and SR 3.2.1.4. The licensee and the NRC staff discussed this concern during a public conference call on November 15, 2012. In a subsequent public conference call on November 29, 2012, the licensee acknowledged that current SRs table Note 2 ("If the OPDMS becomes inoperable while in MODE 1 these surveillances must be performed within 31 days of the last verification of OPDMS parameters."), which applies to current SR 3.2.1.1 and SR 3.2.1.2, and new SR 3.2.1.3 and SR 3.2.1.4 (revised as a surveillance column Note), is a "non-conservative TS requirement" with respect to the second frequency of these current SRs and the first frequency of these new SRs.

The NRC issued Administrative Letter (AL) 98-10: "Dispositioning of Technical Specifications That Are Insufficient to Assure Plant Safety," dated December 29, 1998, to address the NRC staff's expectations regarding correction of facility TS that are found to "contain nonconservative values or specify incorrect actions." In the case noted above, the NRC staff considers "incorrect actions" to include a surveillance column note that incorrectly modifies a surveillance frequency. The AL states that "the discovery of an improper or inadequate TS is considered a degraded or nonconforming condition as defined in Generic Letter (GL) 91-18, Revision 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions."[3] Imposing administrative controls in response to improper or inadequate TS is considered an acceptable short-term corrective action. The NRC staff expects that, following the imposition of administrative controls, an amendment to the VEGP plant-specific TS, with appropriate justification and schedule, will be submitted in a timely fashion.

In an e-mail message dated December 6, 2012 (ADAMS Accession No. ML12345A256), the licensee stated its plans to follow the AL 98-10 process regarding the discovery of a nonconservative TS. The staff determined that that this particular issue had no bearing on the current amendment request because the improper modification of surveillance frequencies by a SRs table Note in the CTS is retained as surveillance column Notes in the corresponding SRs of the ITS. Therefore, this particular issue is beyond the scope of the review. As such, the NRC staff accepts the licensee's proposed resolution of this concern; therefore, **RAI 16-33** is resolved.

Based on the above discussion, the NRC staff concludes that the licensee-proposed administrative and editorial changes listed in Table A are acceptable because they are consistent with the Writer's Guide and the STS, result in no changes in unit operational requirements, and comply with the VEGP licensing basis and the Commission's regulations.

B. More Restrictive Changes

The licensee proposed a number of requirements that are more restrictive than those in the CTS. The ITS requirements in this category include requirements that are either new, more conservative than corresponding requirements in the CTS, or have additional restrictions that are not in the CTS. Among the changes proposed by SNC and found acceptable by the NRC staff are changes in which:

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[3] This most recent version of this guidance is contained in Regulatory Issue Summary (RIS) 2005-20, "Revision to NRC Inspection Manual Part 9900 Technical Guidance, 'Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety (ADAMS Accession No. ML073531346),'" Revision 1, dated April 16, 2008, (ADAMS Accession No. ML073440103).

- The scope of a surveillance requirement is increased (M-1);
- A required action specifies placing the unit in a lower operational mode (M-2);
- A more restrictive plant configuration for surveillance testing is imposed (M-3);
- The scope of a required action is increased or a required action is added (M-4);
- A surveillance requirement is added (M-5);
- A surveillance requirement frequency is increased (i.e., a shorter test interval) (M-6); and
- A required action completion time is decreased (M-7).

Table M attached to this SE lists the more restrictive changes being made to the VEGP CTS. Table M is organized in ITS section order and includes the following:

- DOC identifier;
- Summary description of each more restrictive change adopted;
- Change type; and
- Reference to CTS and ITS requirements.

During the review of the more restrictive changes, the NRC staff issued RAIs to seek further clarifications from the licensee to ensure completeness and accuracy of the proposed changes. The NRC staff evaluated the licensee's responses to these RAIs, found the enhanced justifications consistent with the original proposal and acceptable, except as discussed below:

As described in DOC M06, the licensee proposed moving the following two LCO Notes regarding reactor coolant pump (RCP) starting limitations from LCO 3.4.4, "RCS Loops," LCO 3.4.8, "Minimum RCS Flow," and LCO 3.4.14, "Low Temperature Overpressure Protection (LTOP)," and incorporating them into LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

1. No reactor coolant pump (RCP) shall be started when the RCS temperature is ≥ 350°F unless pressurizer level is < 92%.

2. No RCP shall be started with any RCS cold leg temperature ≤ 350°F unless the secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures and the RCP is started at ≤ 25% of RCP speed.

In **RAI 16-5** the NRC staff requested that the licensee provide additional background information regarding the need for the notes and clarify why specifying the notes in LCO 3.4.3 is more effective than the existing presentation. In its response letter dated October 4, 2012, the licensee acknowledged that the current presentation of these two notes effectively serves their purpose of preventing RCS overpressurization and that the proposed changes are not needed. Consequently, the licensee withdrew all TS changes proposed under DOC M06, in lieu of providing the requested background information. Because the NRC staff previously approved the current presentation, maintaining the current presentation of these two notes is acceptable. Therefore, **RAI 16-5** is resolved.

Based on the above discussion, the NRC staff concludes that the licensee-proposed more restrictive changes comply with the VEGP licensing basis and the Commission's regulations and enhance plant safety by increasing restrictions on plant operation. Therefore, the more restrictive changes listed in Table M are acceptable.

C. Relocated Specifications

According to the final policy statement an LCO (i.e., a Specification subsection in a licensed reactor facility's current TS) that does not satisfy or fall within any of the four specified criteria for establishing an LCO may be relocated from a plant's current TS (an NRC-controlled document) to appropriate licensee-controlled documents. Such a Specification subsection generally would include the following elements:

- an LCO statement (i.e., structure, system, or component operability and operating requirements; and limits on process, core power, and operational parameters);
- applicability statements (i.e., operational modes or other specified conditions);
- remedial action statements (i.e., conditions with associated required actions and completion times);
- associated SRs; and
- any associated figures and tables.

In its LAR letter and supplements to it, the licensee proposed relocating two such Specification subsections from the CTS to the technical requirements manual (TRM). The licensee states in its LAR letter that the TRM is a licensee-controlled document incorporated by reference in the VEGP FSAR. Therefore, changes to the TRM are subject to the FSAR change requirements of 10 CFR 52.98. Control of changes to information relocated to the TRM from the plant-specific TS of a combined license is described in detail in Part 3.0 of this SE.

Table R attached to this SE lists the two Specification subsections that are proposed for relocation from the VEGP CTS to the licensee-controlled TRM. Table R includes the following information:

- DOC identifier;
- Reference to the relocated current Specification subsection;
- Summary description of the relocated current Specification subsection;
- Title of the document where the relocated current Specification subsection will reside (i.e., the new location); and
- Regulatory process for controlling future changes to the provisions in the relocated current Specification subsection (i.e., the regulatory change control process).

The NRC staff's evaluation of the relocation of the two Specification subsections listed in Table R is provided after the following background discussion regarding the relevant accidents that are postulated to occur during refueling operations—the fuel handling accidents. This discussion clarifies that although the two Specifications being evaluated require containment closure and air filtration functions for fuel handling accident (FHA) consequence mitigation, these functions are not credited by the FHA radiological consequence analyses. The following summary is drawn from Section 15.7.4 of the VEGP FSAR, which describes key aspects of the design basis FHA radiological consequence assessment:

- "The fuel handling accident is defined as the dropping of a spent fuel assembly such that every rod in the dropped assembly has its cladding breached so that the activity in the fuel/cladding gap is released."
- "The design basis FHA initial airborne release "source term" is the "inventory of fission products available for release at the time of the accident." It is derived from the reactor core source term (detailed in FSAR Appendix 15A, "Evaluation Models and Parameters for Analysis of Radiological Consequences of Accidents") by taking into account the following factors, which are conservatively chosen to be consistent with Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000:
    - Fission Product Gap Fraction — the fraction of the fuel assembly's fission product inventory (radionuclides of krypton, xenon, and iodine) that is located in the gap between the fuel and the cladding.
    - Iodine Chemical Form — 99.85-percent elemental iodine (based on the assumption of an instantaneous conversion of cesium iodide to elemental iodine "when released from the fuel into the low pH water pool") and 0.15-percent organic iodine (which is "not subject to scrubbing removal in the water pool"); also, "no credit is taken for plateout" of elemental iodine on the fuel cladding.
    - Fuel Assembly Power Level — the "accident involves a fuel assembly that operated at the maximum rated fuel rod peaking factor"; and "all of the fuel rods in the damaged fuel assembly have been operating at the maximum [rated] fuel rod radial peaking factor."
    - Radiological Decay — the "fission product decay time experienced prior to the fuel handling accident is at least 48 hours." (Note that fuel movement in the reactor pressure vessel prior to this period is prohibited by current LCO 3.9.7, "Refueling Operations – Decay Time.") Section 15A.3.1.3 states that "no additional decay is assumed after the activity is released to the environment."
    - Release Pathways — "the overall [water] pool scrubbing decontamination factor for [elemental] iodine is assumed to be 200." [4]
        - *For the incontainment FHA:*
            - "After the activity escapes from the [refueling cavity] water pool, it is assumed that it is released directly to the environment within a 2-hour period without credit for any additional iodine removal process."
            - No credit is taken for the ESFAS Function for automatic closure of containment purge lines on detection of high radioactivity in containment.
            - No credit is taken for removal of airborne iodine by the filters in the containment purge lines.

[4] FSAR Section 15.7.4.2 is inconsistent with the Bases for CTS 3.7.5, "Plant Systems – Spent Fuel Pool Water Level," which refers to an "assumed pool scrubbing factor of 500 for elemental iodine." The TS bases for CTS 3.9.4, "Refueling Operations – Refueling Cavity Water Level," are also inconsistent with the TS bases for CTS 3.7.5 because they do not cite an explicit value of the refueling cavity water pool scrubbing factor for elemental iodine. These inconsistencies have no bearing on the evaluation of the relocation of CTS 3.9.5 and CTS 3.9.6 because the specified functions of containment penetration isolation and containment air filtration are not credited in the FHA radiological consequence analyses.

– *For the spent fuel pool FHA:*

  - “Activity released from the [spent fuel water] pool is assumed to pass directly to the environment with no credit for holdup or delay of release in the [fuel] building.”
  - “There is assumed to be no filtration [after that assumed for water pool scrubbing] in the release pathway.”

The NRC staff has previously reviewed the results of the radiological consequence assessments of the FHA for the AP1000 design, as documented in Section 15.3 of NUREG-1793, Supplement 2, Final Safety Evaluation Report of the AP1000 Design, which states that the applicant [Westinghouse Electric Corporation] had concluded in DCD Tier 2 that:

> “the AP1000 design will provide reasonable assurance that the radiological consequences resulting from any of the above DBAs [a list that includes fuel-handing accidents] will fall within the offsite dose criterion of 0.25 Sv (25 rem) total effective dose equivalent (TEDE), as specified in 10 CFR 50.34(a)(1)(ii)(D), and the control room operator dose criterion of 0.05 Sv (5 rem), as specified in GDC 19, “Control Room,” of Appendix A to 10 CFR Part 50.”

(1) DOC R1 – Relocation of CTS Subsection 3.9.5, “Refueling Operations – Containment Penetrations” to the TRM

Current TS 3.9.5 provides requirements for containment penetrations “during movement of irradiated fuel assemblies within containment.” The containment penetrations addressed by LCO 3.9.5 are those that provide direct access from the containment atmosphere to the outside atmosphere, and include equipment hatches, air lock doors, containment spare penetrations, and all other direct access containment penetrations, such as the containment purge lines. In particular, LCO 3.9.5 states:

“The containment penetrations shall be in the following status:

a. The equipment hatches closed and held in place by four bolts or, if open, the containment air filtration system (VFS) shall be OPERABLE and operating;

b. One door in each air lock closed or, if open, the VFS shall be OPERABLE and operating;

c. The containment spare penetrations closed or, if open, the VFS shall be OPERABLE and operating;

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

   1. Closed by a manual or automatic isolation valve, blind flange, or equivalent, or

   2. Capable of being closed by an OPERABLE Containment Isolation signal.

- NOTE -

Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

---"

Note: paragraph "d" of current LCO 3.6.8, "Containment Systems – Containment Penetrations," and paragraph "d" of current LCO 3.9.5 are identical; paragraph "d" of current LCO 3.6.8 is revised in new LCO 3.6.7 by DOC L18, as follows:

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere**, if open, can be** ~~either:~~

~~1.~~ closed by a manual or automatic isolation valve, blind flange, or equivalent~~, or~~ **prior to steaming into the containment.**

~~2. capable of being closed by an OPERABLE Containment Isolation signal.~~

This change is reasonable because no automatic containment isolation instrumentation function is currently required in MODE 5 or 6. In addition, the purpose of CTS LCO 3.6.8 and ITS LCO 3.6.7 is to ensure the capability to mitigate a loss of normal reactor coolant system heat removal by supporting the passive residual heat removal system (PRHR) and the passive containment cooling system (PCS). The purpose is not to mitigate an incontainment FHA.

By requiring that containment equipment hatches, personnel airlock doors, and spare penetrations be closed or that the nonsafety-related containment air filtration system (VFS) be OPERABLE and in operation, LCO 3.9.5 ensures a capability for FHA radiological consequence mitigation beyond that assumed to be available by the incontainment FHA radiological consequence analyses. Those analyses also take no credit for automatic closure of the containment purge lines, which can occur on detection of high radioactivity in the containment; neither is operator action to manually close the containment purge lines assumed.

In its assessment of generic TS 3.9.5 in Section 16.4 of NUREG-1793, Supplement 2, the NRC staff acknowledged that LCO 3.9.5 meets none of the LCO selection criteria of 10 CFR 50.36(c)(2)(ii), as follows:

"AP1000 TS 3.9.5 for containment penetrations during movement of irradiated fuel assemblies within containment differs from the STS [NUREG-1431, Revision 2] in two respects. Unlike the STS, maintaining closure of the containment penetrations during fuel movement does not satisfy the criteria of 10 CFR 50.36(c)(2)(ii). Rather, TS 3.9.5 is provided as an additional level of defense for the incontainment fuel handling accident (FHA). The design-basis, incontainment FHA safety analysis shows acceptable dose consequences without crediting containment closure or filtration of containment ventilation exhaust. In addition, LCO 3.9.5 specifies the option of placing the non-safety-related containment air filtration system (VFS) in operation in lieu of satisfying the closure provision for the equipment hatch, the personnel airlock, and the containment spare penetrations. This option for meeting the LCO will ensure filtration of containment ventilation exhaust in the event of a FHA involving fuel damage. The staff finds that these two differences are appropriate for the AP1000 design because containment closure and VFS operation are not credited in the inside-containment FHA analysis. Also, specifying two options for meeting the LCO provides operational flexibility during fuel movement inside containment. Therefore, TS 3.9.5 is acceptable."

Current TS 3.6.8, "Containment Systems – Containment Penetrations" (renumbered and modified as new TS 3.6.7 as proposed by this LAR; see DOC D07 and DOC L18) requires the containment closure design feature to mitigate a loss of normal decay heat removal in MODES 5 and 6, an event which is not related to the incontainment FHA. Specifically, new LCO 3.6.7, which is applicable in MODES 5 and 6, states:

"The containment penetrations shall be in the following status:

a. The equipment hatches closed and held in place by four bolts or, if open, can be closed prior to steaming into the containment.

b. One door in each air lock closed or, if open, can be closed prior to steaming into the containment.

c. The containment spare penetrations, if open, can be closed prior to steaming into the containment.

d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere, if open, can be closed by a manual or automatic isolation valve, blind flange, or equivalent prior to steaming into the containment."

Note: paragraph "d" of current LCO 3.6.8 and paragraph "d" of current LCO 3.9.5 are identical; paragraph "d" is revised in new LCO 3.6.7 by DOC L18, as stated above.

For mitigating a loss of normal decay heat removal in MODE 6, the above—AP1000 design-unique—containment penetration status and manual closure-capability requirements were included in the VEGP plant-specific TS in current LCO 3.6.8 (by way of the AP1000 generic TS). However, to provide reasonable assurance that the containment closure mitigation strategy would also be available to respond to an incontainment FHA, the VEGP plant-specific TS also included (by way of the AP1000 generic TS) very similar containment penetration status and manual closure-capability requirements in current LCO 3.9.5. Both of these current LCOs require a closed containment, but they also allow containment to be open provided a specified condition is satisfied. The notable difference between the current LCOs is the different conditions that must be met to permit operation in MODE 6 with open containment penetrations (equipment hatches, personnel airlock doors, and spare penetrations) without reliance on required actions. To allow operation with a specified penetration open, current LCO 3.6.8 requires the capability to achieve containment closure, "prior to steaming into the containment"; whereas, LCO 3.9.5 requires that the VFS be OPERABLE and in operation. Because of this difference, and because these current LCOs are intended to mitigate different events with the unit initially in MODE 6, the NRC staff could not grant the requested relocation of current TS 3.9.5 without additional justification.

The NRC staff sought additional information from the licensee in **RAI 16-32, Issue 1**, which contained two parts:

"(a) The licensee is requested to withdraw the proposed relocations of current TS 3.9.5and 3.9.6 from the LAR since their inclusion resolved staff concerns during the review of the AP600 and AP1000 designs. The licensee may propose to include these requirements as short-term availability controls, along with an appropriate RTNSS [regulatory treatment of nonsafety systems] [also referred to as investment protection short term availability controls] evaluation;

(b) Alternatively, the licensee may provide further justification to support the relocation of current TS 3.9.5 and 3.9.6. As a part of this revised justification, the licensee is requested to (1) state where the requirements of current TS 3.9.5 and 3.9.6 will be relocated, (2) state how the information in the bases will be maintained or placed in the FSAR, (3) describe how each requirement will be revised before being implemented, and (4) commit to control changes to those requirements in accordance with 10 CFR 52.98, which is the correct reference for changes to FSAR information."

In its response, the licensee considered both (a) and (b) and provided the following information:

> "The NRC Staff concerns related to the inclusion of TS 3.9.5 and TS 3.9.6 during the review of the AP600 are discussed in NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," Section 16.3, and referenced FSER Open Item 16.3-1. In NUREG-1512, the Staff summarized the issue as follows:
>
> > 'The staff took the position that, although the analyzed dose from a fuel handling accident may not exceed the dose acceptance criteria, the principle of defense-in-depth makes it prudent to establish some type of containment barrier to a postulated release from a fuel handling accident.'"

The licensee also pointed to the discussion, previously submitted under DOC R1 and DOC R2 in LAR Enclosure 1, regarding the PRA insights about containment penetration closure and containment ventilation exhaust for FHA dose mitigation:

> "The Investment Protection Short Term Availability Controls found in FSAR 16.3 also did not meet Technical Specifications 10 CFR 50.36 selection criteria; however, they were selected for control in FSAR 16.3 based on PRA insights that identified systems, structures and components that are important in protecting the utility's investment and for preventing and mitigating severe accidents. This provided a defense-in-depth control associated with these insights. As provided in LAR DOC R1 and R2, the PRA shows that the importance in this regard for the current TS 3.9.5 and TS 3.9.6 controls is not significant (i.e., is less than 1.3% of the AP1000 Shutdown Large Release Frequency). Therefore, not only do the controls in current TS 3.9.5 and TS 3.9.6 not meet 10 CFR 50.36 for inclusion in TS, they also would not meet the type of evaluation criteria that was applied to identifying controls applicable for FSAR Section 16.3."

Although having the capability for establishing some type of containment barrier to a postulated release from an incontainment FHA is a prudent defense-in-depth measure, including this capability in the TS is not warranted by 10 CFR 50.36. The NRC staff notes that requirements for containment closure with the unit in MODE 6, which remain in new TS LCO 3.6.7, provide assurance that the relocated defense-in-depth measure of maintaining the capability to close containment will be available to mitigate a FHA as well as a loss of normal shutdown cooling. Specifically:

- New LCO 3.6.7 provides containment closure controls that are similar to those provided by CTS LCO 3.9.5, because its applicability of MODES 5 and 6 encompasses the entire period when CTS 3.9.5 would be applicable, except for the brief period following removal of the last fuel assembly from the reactor pressure vessel (which exits MODE 6) until the assembly is removed from containment.

- New LCO 3.6.7.a, b, and c (current LCO 3.6.8.a, b, and c) provide TS closure requirements for the equipment hatches, air lock doors, and containment spare penetrations. All three of these requirements allow the associated penetration to be open, provided that each is capable of being closed "prior to steaming into the containment." This compares to the alternate provision of current LCO 3.9.5.a, b, and c that allows these penetrations to be open provided that the VFS is OPERABLE and operating. However, since no VFS filtration of radioactive material exiting the refueling cavity water pool is credited by the FHA dose assessment, this alternate provision is relocated from the plant-specific TS. (See discussion of DOC R2 below.) So the current allowance—which is retained by new LCO 3.6.7.a, b, and c—to open these penetrations in MODE 6 is less restrictive than the requirement of current LCO 3.9.5 that these penetrations must be closed.

  According to CTS 3.9.4, "Refueling Operations – Refueling Cavity Water Level," during movement of irradiated fuel assemblies within containment, which is when an incontainment FHA could occur, the refueling cavity water level must be maintained at least 23 ft above the top of the reactor vessel flange, which is an assumed initial condition in the FHA analyses. According to CTS 3.9.7, "Refueling Operations – Decay Time," movement of irradiated fuel assemblies within the reactor pressure vessel cannot begin unless the reactor has been subcritical for at least 48 hours, which is also an assumed initial condition in the FHA analyses.

  If any of the equipment hatches, air lock doors, or containment spare penetrations are open, the open penetrations must be capable of being closed prior to steaming into the containment. With the above initial conditions on water level and elapsed decay time, assumed by the FHA analyses, the "time permitted for containment closure" appears to be no less than 6 hours as indicated by the curve labeled, "Mode 6, Cavity Flooded, Temperature = 120 deg," on CTS 3.6.8 bases Figure B 3.6.8-1, "Time Prior to Coolant Inventory Boiling," which is unchanged by this LAR. In the event of an incontainment FHA, the controls in place to close such open penetrations may not be capable of closing them before expiration of the conservatively brief 2-hour release period assumed by the FHA dose assessment. Nevertheless, barring unusual circumstances, the NRC staff reasonably anticipates that the specified capability for closing these penetrations can reduce the release of radioactive material from a damaged fuel assembly to the outside atmosphere. Therefore, a measure of "defense-in-depth" to protect against an incontainment FHA is also afforded by the containment penetration closure requirements of new LCO 3.6.7.a, b, and c.

- Current LCO 3.6.8.d provides the requirements for each penetration providing direct access from the containment atmosphere to the outside atmosphere. These penetrations include the containment air filter supply and exhaust penetrations, and the vent and purge valves, and the vacuum relief valves. For postulated shutdown events in MODES 5 and 6, reactor coolant system (RCS) heat removal is provided by either passive residual heat removal (PRHR) or in-containment refueling water storage tank (IRWST) injection and containment sump recirculation to the IRWST. To support RCS heat removal, containment closure is required to limit the loss of the cooling water inventory from containment.

CTS LCO 3.6.8.d.1 states:

> "The containment penetrations shall be in the following status: d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either: 1. [is] closed by a manual or automatic isolation valve, blind flange, or equivalent, or"

This is revised by DOC L18 so that ITS LCO 3.6.7.d states,

> "The containment penetrations shall be in the following status: d. Each penetration providing direct access from the containment atmosphere to the outside atmosphere~~, either:~~ **if open, can be** closed by a manual or automatic isolation valve, blind flange, or equivalent~~, or~~ **prior to steaming into containment**."

The direct access containment penetration alternate closure provision of CTS LCO 3.6.8.d.2 ("capable of being closed by an OPERABLE Containment Isolation signal") is omitted from ITS LCO 3.6.7.d by DOC L18, as described above. The only containment isolation signals that are required to be OPERABLE by CTS 3.3.2, "Engineered Safety Features Actuation System (ESFAS) Instrumentation," in MODES 5 and 6 are manual initiation functions. As discussed by DOC L18, the current applicability of MODES 5 and 6 is omitted by the corresponding ITS instrumentation function applicability, except as noted:

– CTS 3.3.2 Function 3.a, "Containment Isolation" on "Manual Initiation," which is applicable in MODES 1, 2, 3, and 4; and MODES 5 and 6 unless the associated containment penetration flow path is isolated.

  CTS 3.3.2 Function 3.a corresponds to:
  ITS 3.3.9, "ESFAS Manual Initiation," Function 3, "Containment Isolation - Manual Initiation," which is applicable in MODES 1, 2, 3, and 4.

  In the new presentation, ITS 3.3.9 Function 3 supports the design actuation logic that initiates containment isolation when an actuation signal for manual initiation of containment isolation is present, as shown on FSAR Figure 7.2-1 (Sheet 13).

  CTS 3.3.2 Function 3.b, "Containment Isolation" on "Manual Initiation of Passive Containment Cooling (PCS)," which is applicable in MODES 1, 2, 3, and 4; and MODES 5 and 6 unless the associated containment penetration flow path is isolated or decay heat is ≤ 6.0 MWt. CTS Function 3.b is initiated by:

  CTS 3.3.2 Function 12.a, "PCS Actuation" on "Manual Initiation," which is applicable in MODES 1, 2, 3, and 4; and MODES 5 and 6 unless decay heat is ≤ 6.0 MWt.

  CTS 3.3.2 Functions 3.b and 12.a correspond to:
  ITS 3.3.9, "ESFAS Manual Initiation," Function 8, "PCS Actuation - Manual Initiation," which is applicable in MODES 1, 2, 3, and 4; and MODES 5 and 6 unless decay heat is ≤ 6.0 MWt.

  In the new presentation, ITS 3.3.9 Function 8 supports the design actuation logic that initiates containment isolation when an actuation signal for manual initiation of PCS is present, as shown on FSAR Figure 7.2-1 (Sheet 13).

– CTS 3.3.2 Function 3.c, “Containment Isolation” on “Safeguards Actuation,” which is applicable in MODES 1, 2, 3, and 4; and MODE 5 unless the associated containment penetration flow path is isolated, is initiated by:

  CTS 3.3.2 Function 1.a, “Safeguards Actuation” on “Manual Initiation,” which is applicable in MODES 1, 2, 3, and 4; and MODE 5.

  CTS 3.3.2 Functions 3.c and 1.a correspond to:
  ITS 3.3.9, “ESFAS Manual Initiation,” Function 1, “Safeguards Actuation - Manual Initiation,” which is applicable in MODES 1, 2, 3, and 4; and MODE 5.

  In the new presentation, ITS 3.3.9 Function 1 supports the design actuation logic that initiates containment isolation when an actuation signal for manual initiation of safeguards is present, as shown on FSAR Figure 7.2-1 (Sheets 11 and 13).

The above listing demonstrates that no automatic containment isolation functions are currently required in MODES 5 and 6. Thus, CTS LCO 3.6.8.d.2 actually requires that the direct-access containment penetrations be capable of closing from a manual initiation signal only. Even so, the above ESFAS manual functions for containment isolation are not needed to support any accident analysis assumptions in MODES 5 and 6. Therefore, deleting CTS LCO 3.6.8.d.2 is acceptable.

In MODE 6, ITS LCO 3.6.7.d only requires the capability to manually close all open direct access containment isolation valves prior to reaching a sufficiently high temperature in the refueling cavity water pool to initiate steaming into the containment. If decay heat exceeds 6.0 MWt, LCO 3.3.9 Function 8 is required to be OPERABLE in MODE 6. In the event that passive containment cooling actuation is manually initiated, containment isolation will also occur to minimize the loss of cooling water inventory from containment.

The incontainment FHA analyses take no credit for automatic closure of the containment purge lines on detection of high radioactivity, which is reflected by the VEGP design having no automatic containment isolation in MODES 5 and 6, including no automatic containment isolation function on a containment radioactivity high signal. CTS 3.3.2 and ITS 3.3.8 (both entitled “ESFAS Instrumentation”) specify the containment radioactivity high 1 and high 2 ESFAS functions as follows, with no requirement that they be operable in MODE 5 or 6, or during movement of irradiated fuel assemblies within containment:

– CTS 3.3.2, “ESFAS Instrumentation,”

  Function 16.d, “Chemical Volume and Control System (CVS) Makeup Isolation” on “**Containment Radioactivity – High 2**,” which is applicable in MODES 1 and 2, and in MODE 3 unless the associated CVS makeup flow path is isolated (Table 3.3.2-1 Footnote (e)); and

  Function 17.a, “Normal Residual Heat Removal System (RNS) Isolation” on “**Containment Radioactivity – High 2**,” which is applicable in MODES 1 and 2, and in MODE 3 unless the associated RNS flow path is isolated (Table 3.3.2-1 Footnote (e)); and which become

– ITS 3.3.8, “ESFAS Instrumentation,”

Function 4, “**Containment Radioactivity – High 2**,” which is applicable in MODES 1, 2, and 3.

– CTS 3.3.2, “ESFAS Instrumentation,”

Function 19.a, “Containment Air Filtration System Isolation” on “**Containment Radioactivity – High 1**,” which is applicable in MODES 1, 2, and 3, and in MODE 4 with the RCS not being cooled by the RNS (Table 3.3.2-1 Footnote (b)), and which becomes

– ITS 3.3.8, “ESFAS Instrumentation,”

Function 3, “**Containment Radioactivity – High 1**,” with the same applicability (Table 3.3.8-1 Footnote (b)).

While a containment with closed penetrations (or penetrations with direct access from the containment atmosphere to the outside atmosphere that are capable of automatic closure), or a containment with open penetrations but with the VFS in operation would both mitigate the radiological consequences of an FHA, neither would prevent an FHA. They would only reduce the radiation dose were fuel failure to occur. The radioactivity release rates conservatively postulated by the FHA analyses are not large enough to result in a large early radioactivity release because a decay time of 48 hours since the reactor was last critical is required to have expired by CTS LCO 3.9.7 before irradiated fuel assemblies can be removed from the reactor vessel and placed in the spent fuel pool. Requiring the reactor to be subcritical for at least 48 hours allows short-lived fission products in the fuel assemblies of the reactor core to significantly decay before commencing movement of these fuel assemblies within containment and within the spent fuel pool in the fuel handling area of the auxiliary building.

<u>Comparison to 10 CFR 50.36(c)(2)(ii) Criteria</u>

Criterion 1. The status of containment penetrations and the VFS during movement of irradiated fuel assemblies within containment is neither an instrument used for nor an instrument capable of detecting a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2. The status of containment penetrations and the VFS during movement of irradiated fuel assemblies within containment is not a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis.

Criterion 3. The status of containment penetrations and the VFS during movement of irradiated fuel assemblies within containment is not a structure, system, or component which functions or actuates as a part of the primary success path in the mitigation of a design basis fuel handling accident.

Criterion 4. As documented in the VEGP PRA, the status of containment penetrations and the VFS during movement of irradiated fuel assemblies within containment is not credited for mitigating the consequences of shutdown events with reduced RCS inventory in MODES 5 and 6. This is because LCO 3.9.4 precludes moving fuel in containment with reduced RCS inventory. However, CTS 3.6.8 (ITS 3.6.7), “Containment Penetrations,” supports these shutdown events, and requires that containment penetrations be either closed, or be capable of being closed prior to steaming into containment. For events with reduced RCS inventory that occur soon after the reactor was last critical, all the containment penetrations would likely be closed because the time prior to coolant inventory boiling is a small fraction of an hour, as depicted on CTS bases Figure B 3.6.8-1.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the NRC staff concludes that requirements regarding the status of containment penetrations and the VFS during movement of irradiated fuel assemblies within containment may be relocated from the VEGP CTS. Current Specification 3.9.5 will be relocated to the TRM, and changes to the TRM will be controlled pursuant to 10 CFR 52.98.

(2) DOC R2 – Relocation of CTS Subsection 3.9.6, "Refueling Operations - Containment Air Filtration System (VFS)" to the TRM

CTS 3.9.6, "Containment Air Filtration System (VFS)," provides requirements for the VFS exhaust subsystem during movement of irradiated fuel assemblies in the spent fuel pool in the fuel handling area of the auxiliary building. This Specification provides an additional level of defense-in-depth against the possibility of a fission product release from an FHA in the spent fuel pool.

The radiologically controlled area ventilation system (VAS) and the containment air filtration system (VFS) serve the fuel handling area of the auxiliary building and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas which are provided with a separate ventilation system. The VFS also serves the containment by providing, among other functions, filtration to limit the release of airborne radioactivity at the site boundary to within acceptable levels. During conditions of abnormal airborne radioactivity in the fuel handling area of the auxiliary building, auxiliary and/or annex buildings, the VFS filtration units provide filtered exhaust to minimize unfiltered offsite releases.

The VFS also provides the safety-related functions of

- containment isolation
  - CTS/ITS 3.6.3, "Containment Isolation Valves"; and
- containment vacuum relief
  - ITS 3.3.8, "ESFAS Instrumentation," Function 1, "Containment Pressure – Low 2";
  - ITS 3.3.9, "ESFAS Manual Initiation," Function 15, "Containment Vacuum Relief Valve Actuation – Manual initiation"; and
  - ITS 3.6.9, "Vacuum Relief Valves."

If high airborne radioactivity is detected in the exhaust air from the fuel handling area, the auxiliary building, or the annex buildings, the VAS supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment ; the associated VAS subsystem then aligns to the VFS filtered exhaust subsystem, which starts. The VFS filtered exhaust subsystem prevents exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at ≥ 0.125 inches water gauge negative pressure relative to the outside atmospheric pressure.

Current LCO 3.9.6 states, “One VFS exhaust subsystem shall be OPERABLE.” A VFS exhaust subsystem is considered OPERABLE when its associated (a) exhaust fan is capable of operating; (b) high-efficiency particulate air (HEPA) filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and (c) the associated heater and ductwork are capable of operating. To meet these LCO OPERABILITY conditions, the following automatic actuations must also be OPERABLE (as verified by SR 3.9.6.2): On an actual or simulated actuation signal (high airborne activity detected in the exhaust air from the fuel handling area or the auxiliary/annex building zones; or detection of a high pressure differential with respect to the outside environment caused by a disruption in the normal ventilation airflow rate);

- the VAS supply and exhaust duct isolation dampers must automatically close to isolate the outside environment from the affected area—either the fuel handling area, or one or both zones in the auxiliary/annex building;
- the VAS subsystem for the isolated area (e.g., fuel handling area) must automatically align to the VFS filtered exhaust subsystem; and
- the VFS filtered exhaust subsystem must automatically start and operate.

The VFS exhaust subsystem maintains a slight negative pressure differential in the isolated zone or area (as verified by CTS SR 3.9.6.3).

For an FHA in the spent fuel pool in the fuel handling area of the auxiliary building, the dose analysis does not rely on the availability of the VAS to isolate and realign to VFS, or the OPERABILITY of the VFS exhaust subsystem to meet the offsite radiation exposure limits. Specifications that support spent fuel pool FHA analysis assumptions are current TS 3.7.5, "Plant Systems - Spent Fuel Pool Water Level," and current TS 3.9.7, “Refueling Operations - Decay Time.” These Specification subsections support ensuring the design basis radiological consequences resulting from an FHA in the spent fuel pool are within the dose values provided in FSAR Section 15.7.4.

While the VFS exhaust subsystem is designed to mitigate the consequences of an FHA in the spent fuel pool, it would only reduce the radiation dose were fuel failure to occur. The radioactivity release rates conservatively postulated by the FHA analyses are not large enough to result in a large early radioactivity release because a decay time of 48 hours since the reactor was last critical is required to have expired by current LCO 3.9.7 before irradiated fuel assemblies can be removed from the reactor vessel and placed in the spent fuel pool. Requiring the reactor to be subcritical for at least 48 hours allows short-lived fission products in the fuel assemblies of the reactor core to significantly decay before commencing movement of these fuel assemblies within containment and within the spent fuel pool in the fuel handling area of the auxiliary building.

In the LAR 12-002 application, the licensee states that while the VFS exhaust subsystem is modeled in the PRA, its importance in limiting the likelihood of a severe accident sequence that has shown to be significant to public health and safety is not significant (i.e., is less than 1.3% of the AP1000 Shutdown Large Release Frequency).

<u>Comparison to 10 CFR 50.36(c)(2)(ii) Criteria</u>

Criterion 1. The VFS exhaust subsystem is neither an instrument used for nor an instrument capable of detecting a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2. The OPERABILITY of the VFS exhaust subsystem is not a design feature or operating restriction that is an initial condition of a design basis accident or transient analysis.

Criterion 3. The VFS exhaust subsystem is not used as part of a primary success path in the mitigation of a FHA in the spent fuel pool in the fuel building (fuel handling area of the auxiliary building).

Criterion 4. The VFS exhaust subsystem is not credited with mitigating the consequences of any shutdown events as documented in the VEGP PRA.

Since the 10 CFR 50.36(c)(2)(ii) criteria are not met, the NRC staff concludes that requirements for the VFS exhaust subsystem during movement of irradiated fuel assemblies in the fuel building (fuel handling area of the auxiliary building) may be relocated from the VEGP TS. Current Specification 3.9.6 will be relocated to the TRM, and changes to the TRM will be controlled pursuant to 10 CFR 52.98.

<u>Administrative Changes to the Provisions of Specifications 3.9.5 and 3.9.6 Following Relocation</u>

In **RAI 16-32**, **Issue 1**, the NRC staff requested that the licensee provide further justification to support the relocation of Specifications 3.9.5 and 3.9.6, and describe in the justification how the relocated requirements will be revised before being implemented in the licensee-controlled document. In its response, the licensee provided the following information concerning the relocation of Specifications 3.9.5 and 3.9.6.

> "The commitment to relocate the TS [3.9.5 and 3.9.6] and associated Bases involves only format changes to reflect TRM content versus Technical Specification content, as well as appropriate terminology changes such as requiring "functionality" versus "operability." For example, the numbering scheme would provide for a unique numbering different than "LCO 3.9.5" numbering for the Technical Specification. Similar formatting, numbering, and editorial changes would be made in capturing pertinent Definitions, Use and Applications, as well as LCO Applicability and SR Applicability Technical Specifications for inclusion in the TRM. All such changes (as well as any future changes) would be made in accordance with 10 CFR 52.98. All changes, including initial adoption into FSAR Chapter 16, would be reported to the NRC in accordance with applicable Regulations."

The staff finds the described administrative changes associated with moving Specifications 3.9.5 and 3.9.6 and the associated bases to the TRM to be appropriate.

D. Detail Removed Changes

When requirements have been shown to give little or no safety benefit, their removal from the TS may be appropriate. In some cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of the owners groups' comments on STS. These changes proposed by the licensee were grouped in the following types:

Type 1 — Removing Details of System Design and System Description, Including Design Limits

The design of the facility is required to be described in the FSAR by 10 CFR 52.79. The regulations at 10 CFR 52.98 specify controls for changing the facility as described in the FSAR. The TS bases also contain descriptions of system design. Specification 5.5.6, "Technical Specifications (TS) Bases Control Program," provides a means for processing changes to the bases. Removing details of system design can be acceptable when the associated CTS requirements which are being retained without these details are adequate to ensure safe operation of the facility. In addition, retaining such details in TS is unnecessary to ensure proper control of changes. Therefore, it is acceptable to remove Type 1 details from the VEGP TS and place them in licensee-controlled documents.

Type 2 — Removing Procedural Details for Meeting TS Requirements

Details for performing TS SRs are more appropriately specified in the plant procedures. Prescriptive procedural information in a TS requirement is unlikely to contain all procedural considerations necessary for the plant operators to comply with TS, and referral to plant procedures is therefore necessary in any event. Changes to procedural details include those associated with limits retained in the ITS. For example, in Specification Section 5.4, "Procedures," Specification 5.4.1 requires that written procedures shall be established, implemented, and maintained covering activities that include all programs specified in Specification Section 5.5, "Programs and Manuals." Specification 5.5.3, "Inservice Testing Program," requires a program to provide controls for inservice testing (IST) of American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Class 1, 2, and 3 components. The IST program includes defining testing frequencies specified in the ASME Operation and Maintenance Standards and Codes (OM Codes), and applicable addenda. Specifically, procedural details concerning the status of open containment access hatches and doors are unnecessary to clarify the requirement in CTS LCO 3.6.8 that such penetrations can be closed prior to steaming in containment. Since the CTS requirements retained in the ITS are adequate to ensure safe operation of the facility, the NRC staff concludes that it is acceptable to remove Type 2 details from the VEGP CTS and place them in licensee-controlled documents.

Table D attached to this SE lists the proposed detail removed changes to the VEGP CTS. Table D is organized in ITS section order and includes the following:

- DOC identifier;
- Reference to affected TS requirements;
- Summary description of the removed details;
- Name of the licensee-controlled document to contain the removed details (location);
- Regulation (or TS Specification) for controlling future changes to relocated requirements (change control process); and
- Type of change.

During the review of the detail removed changes, the NRC staff issued RAIs to seek further clarification from the licensee to ensure completeness and accuracy of the proposed changes. The NRC staff evaluated the licensee's responses to these RAIs, found the enhanced justifications consistent with the original proposal and acceptable.

Based on the above discussion, the NRC staff concludes that the removed detailed information and requirements described in Table D are not needed to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. The NRC staff also concludes that the removed detailed information and requirements do not fall within any of the four criteria set forth in 10 CFR 50.36(c)(2)(ii). Accordingly, these requirements may reside in one or more of the following licensee-controlled documents for which changes are adequately governed by a regulatory or TS requirement:

- TS bases controlled in accordance with TS 5.5.6, “Technical Specifications (TS) Bases Control Program”
- FSAR controlled by 10 CFR 52.98

E. Less Restrictive Changes

Less restrictive changes include deletions of CTS requirements or relaxations to portions of CTS requirements that are being retained in the ITS. When requirements have been shown to give little or no safety benefit, their relaxation or removal from the TS may be appropriate. In many cases, relaxations previously granted to individual plants on a plant-specific basis were the result of (1) generic NRC actions, (2) new NRC staff positions that have evolved from technological advancements and operating experience, or (3) resolution of Owners Groups’ comments on STS.

The NRC staff grouped the licensee-proposed less restrictive changes to CTS requirements in the following ten categories or types:

Category 1 — Relaxation of LCO Requirement (L-1)

Category 2 — Relaxation of Applicability Requirement (L-2)

Category 3 — Relaxation of Completion Time (L-3)

Category 4 — Relaxation of Required Action (L-4)

Category 5 — Relaxation of SR Testing Limitations (L-5)

Category 6 — Relaxation of SR Frequency (L-6)

Category 7 — Replacement of SR with an Equivalent Requirement (L-7)

Category 8 — Deletion of Reporting Requirement (L-8)

Category 9 — Deletion of LCO 3.0.8 (L-9)

Category 10 — Deletion of the Definition of CORE ALTERATION (L-10)

A discussion of each of the “Less Restrictive Changes” is included in Table L, which is an attachment to this SE. The NRC staff’s evaluation of each category below summarizes the overall findings for each category and provides the overall conclusion.

Category 1 — Relaxation of LCO Requirement

The current TS contain operating limits that provide little or no safety benefit to the operation of the plant. Relaxation of current LCO requirements include deleting LCO requirements for equipment or systems that establish system capability beyond that assumed to function by the applicable accident analyses, or that are implicitly necessary for LCO-required systems, components, and devices to be OPERABLE. Current TS changes of this type allow operators to focus more on operational matters that are important to safety.

Licensee-proposed relaxations of current TS LCOs also include:

1) Removing explicit OPERABILITY requirements for instrumentation functions that are proposed to be implicitly required by the LCOs for their supported instrumentation functions. For example, explicit LCO, applicability, action, and surveillance requirements related to RTS and ESFAS interlocks or permissive functions are proposed to be implicitly covered by the requirements for the supported RTS and ESFAS instrumentation functions, and so are deleted.
2) Changing the number of required divisions or channels to the minimum number recommended by the applicable regulatory guidance. For example, in Regulatory Guide (RG) 1.97 the post-accident monitoring (PAM) instrumentation redundancy guidance stipulates requiring two OPERABLE channels for each specified PAM function.

The resulting revised LCOs maintain an adequate degree of protection because they are consistent with the accident analyses. They also provide reasonable operational flexibility without adversely affecting the safe operation of the plant. The revised LCOs also maintain consistency with STS guidance and the VEGP current licensing basis. Therefore, the NRC staff finds that the licensee's proposed changes involving relaxation of LCO requirements are acceptable.

Category 2 — Relaxation of Applicability Requirement

The current TS require compliance with each LCO during the applicable MODES or other conditions specified in each Specification subsection's applicability statement. When current TS applicability requirements are inconsistent with the applicable accident analysis assumptions for an LCO-required system, subsystem, or component, the licensee proposed to change the applicability requirements by removing the inconsistent provisions and establishing a consistent set of applicability requirements in the improved TS.

The licensee also proposed replacing prescriptive applicability conditions with less prescriptive conditions that provide greater operational flexibility. For example:

1) Instead of requiring the LCO to be met when reactor trip breakers are closed, require the LCO to be met when either one or more control rods are not fully inserted, or the capability to withdraw control rods is OPERABLE.
2) Instead of requiring the LCO to be met in MODE 5 with the 'calculated' reactor decay heat > 6.0 MWt, require meeting the LCO in MODE 5 with the reactor decay heat > 6.0 MWt.

Such changes are designated as less restrictive because one specific method for meeting the intended TS requirements is being removed from the TS, allowing alternate methods to establish the equivalent conditions.

These changes do not adversely affect safety because each resulting Specification subsection's applicability statement will ensure the associated LCO, action, and surveillance requirements are met during the operational conditions that are assumed by the applicable accident analyses, thereby maintaining an adequate degree of protection. The revised applicability requirements also maintain consistency with STS guidance and the VEGP current licensing basis. Therefore, the NRC staff finds that the licensee's proposed changes involving relaxation of applicability requirements are acceptable.

Category 3 — Relaxation of Completion Time

Upon discovery of a failure to meet an LCO the plant is in a degraded condition. An LCO's associated ACTIONS specify time limits for completing required actions of the associated applicable conditions, which correspond to the degraded condition. Required actions establish remedial measures that must be taken within specified completion times. Completion times specify limits on the duration of plant operation in a degraded condition and allow reasonable time to accomplish the required action. The current TS include deterministic completion times for various degrees of degraded conditions. For example, a loss of redundancy in an LCO-required system is typically allowed a 72-hour completion time to restore the inoperable subsystem to OPERABLE status. In addition, current TS specify repair times for an LCO-required support system that are consistent with the repair times of the supported LCO-required system. The deterministic completion times in the current TS are consistent with the guidance for completion times in the latest revision of the STS, except for those proposed for change by this LAR.

Some current TS LCOs have ACTIONS with conditions and required actions that have a second completion time based not on when the associated condition was entered, but on when the LCO was initially not met. Such completion times were meant to limit the duration of plant operation with an LCO not met when entry into one condition is independent of another condition because the conditions are for diverse equipment. For example, in current TS 3.8.5, "Distribution Systems – Operating," LCO 3.8.5 requires both "ac instrument and control bus" subsystems, and "dc electrical power distribution bus" subsystems to be OPERABLE. The ACTIONS address inoperability of these two subsystems in separate conditions. Operating experience has shown that this additional completion time is unnecessary to prevent abuse of such LCOs by repeatedly entering and exiting ACTIONS conditions to perform maintenance or testing during power operation without ever restoring compliance with the LCO. Such completion times were removed in the latest version of the STS in accordance with TSTF-439-A, "Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO," Revision 2.

The ACTIONS table for some current Specifications specifies a Note that allows for separate condition entry for inoperable equipment, or sets of equipment, such as the four channels of an instrumentation function, or each containment penetration flow path, or for each isolation valve. In some cases, such a Note is provided for a specific ACTIONS condition. Each set of equipment or each component so identified tracks the required action completion time clock independently in the event it is inoperable. For example, current TS 3.7.2, "Main Steam Isolation Valves (MSIVs)," Condition D contains a Note that states, "Separate Condition entry is allowed for each MSIV." Condition D states,

> "One or two MSIVs inoperable in MODE 2, 3, or 4.
>
> <u>OR</u>
>
> One or more of the turbine stop valves and its associated turbine control valve, all turbine bypass valves, or moisture separator reheater 2nd stage steam isolation valves inoperable in MODE 2, 3, or 4."

Basing the separate condition entry on just the MSIVs seems inconsistent with the Specification's OPERABILITY requirement; current LCO 3.7.2 states, "The minimum combination of valves required for steam flow isolation shall be OPERABLE." This means that of the various isolation valves in each main steam line flow path, at least one isolation valve must be OPERABLE. Included in each main steam line flow path are not only the MSIVs, but also the turbine stop valves, turbine control valves, turbine bypass valves, and moisture separator reheater 2nd stage steam isolation valves. As described by DOC A094, this list should also include MSIV bypass valves and main steam line drain valves. Therefore, the Note is revised to say, "Separate Condition entry is allowed for each main steam line flow path." This change is a relaxation of completion time because two inoperable valves in separate flow paths will be allowed to track their own completion time clocks. Since this is was the intent of the current Note, there is no adverse safety impact.

Completions times in STS are based on the OPERABILITY status of redundant TS-required features, the capacity and capability of remaining TS-required features, providing a reasonable time for repair or replacement of required features, vendor-developed standard repair times, and the low probability of a design-basis accident occurring during the repair period. The licensee's proposed changes involving relaxation of completion times do not adversely affect plant safety, since the revised completion times are based on these same considerations. The revised completion times are, therefore, consistent with the guidance established by the STS, taking into consideration the VEGP current licensing basis. Therefore, the NRC staff finds that the licensee's proposed changes involving relaxation of completion time requirements are acceptable.

Category 4 — Relaxation of Required Action

As stated in 10 CFR 50.36(c)(2)(i), "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." The current TS specify required actions to restore inoperable equipment to the required functional capability or performance level (that is, OPERABILITY), to implement remedial measures providing an equivalent level of protection, or to place the plant in an operational MODE or condition in which the LCO is not applicable.

Licensee-proposed relaxations of current TS action requirements include:

1) Adding alternatives to restoring compliance with the LCO—alternate actions that provide equivalent levels of protection. Such optional remedial measures include placing equipment in the safety state assumed by the safety analysis of design basis accidents and transients. Such changes are designated as less restrictive because they provide additional operational flexibility.
2) Replacing a current action requirement with equivalent action requirements that provide additional operational flexibility. For example, instead of requiring reactor trip breakers to be open, require insertion of all control rods and disabling the capability to withdraw control rods. Another example is the change to Required Action D.1 of current TS 3.5.2, "Core Makeup Tanks (CMTs) – Operating." This required action is changed from "Vent noncondensible gases" to "Restore CMT inlet line noncondensible gas volume to within limit." Such changes are designated as less restrictive because one specific method for meeting the intended TS requirements is being removed from the TS, allowing alternate methods to establish the equivalent conditions.

3) Changing condition statements to more closely describe the intended situation involving a failure to meet the LCO. For example, the current TS 3.5.2 ACTIONS condition that states, "One core makeup tank (CMT) inoperable due to presence of noncondensible gases in one high point vent," appears to apply upon discovery of any amount of noncondensible gases. This condition is changed to state "One CMT inlet line with noncondensible gas volume not within limit." The presence of some noncondensible gases does not mean that the CMT injection capability is immediately inoperable, but that gases are collecting and should be vented.

   This change makes the condition language consistent with the CMT OPERABILITY verification by current SR 3.5.2.4, "Verify the volume of noncondensible gases in each CMT inlet line has not caused the high-point water level to drop below the sensor." Since this surveillance specifies the limit on noncondensible gas volume (in terms of a water level sensor indication value), the condition need only refer to a CMT inlet line with noncondensible gas volume not within limit. Likewise, the associated required action is changed from "Vent noncondensible gases" to "Restore CMT inlet line noncondensible gas volume to within limit." These changes are also consistent with the Writer's Guide, paragraph 4.1.6.f.

   The revised condition and required action statements are considered less restrictive because they clarify when the condition would apply and when the condition would be considered corrected.

4) Removing action requirements that are redundant to other action requirements or that are not an effective or practical response to the specified plant condition, and therefore not appropriate. Examples of such action requirements proposed for removal are:

   a) Required Actions D.1.1, D.2.1, and D.2.2 of current TS 3.3.1, "Reactor Trip Instrumentation," for Function 2.a, "Power Range Neutron Flux High Setpoint." These action requirements are removed because current TS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," already specifies adequate compensatory measures that address the potential impact on core radial power distribution monitoring when one or more power range neutron flux channels are inoperable.

   b) Required Action B.1 of current TS 3.4.10, "RCS Specific Activity," requires performing SR 3.4.10.2 within 4 hours upon discovery that DOSE EQUIVALENT XE-133 > 280 μCi/gm. SR 3.4.10.2 states, "Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm. Since current Required Action B.2 requires placing the unit in MODE 3 below 500 degrees F within 6 hours, which exits the Specification's Applicability, performing this surveillance has no safety benefit and is an unnecessary burden on the unit operators. Therefore, removing the required action to perform the surveillance does not adversely affect plant safety.

5) Removing actions associated with LCO or applicability requirements that have been removed or changed in such a way that the required action is no longer appropriate or needed. For example, explicit LCO, applicability, action, and surveillance requirements related to RTS and ESFAS interlocks or permissive functions are proposed to be implicitly included in the TS requirements for the supported RTS and ESFAS instrumentation functions, and so are deleted. Although removing these explicit requirements is considered less restrictive, in a number of cases, the proposed supported function's requirements exceed the current requirements for the supporting permissive function. Therefore, taken together, these changes have no adverse effect on plant safety.

6) Replacing existing ESFAS instrumentation Specification action requirements for supported actuated ESF system valves (end devices that receive an actuation signal from the actuation function) with a required action to declare the supported valves inoperable. The new action leads to immediately entering the applicable conditions and required actions of the ESF system Specification's ACTIONS table. The ESF system Specification's current action requirements are appropriate for the system's affected components and the level of degradation of the supporting instrumentation actuation function. Therefore, changes of this kind do not adversely affect plant safety.

The NRC staff finds that the resulting TS action requirements provide measures that adequately compensate for the inoperable equipment, and are commensurate with the safety importance of the equipment's function. Accordingly, the action requirements in the improved TS will ensure safe operation of the plant during periods in which an LCO is not met. The revised action requirements are, therefore, consistent with the guidance established by the STS, taking into consideration the VEGP current licensing basis. Therefore, the NRC staff finds that the licensee's proposed changes involving relaxation of required actions are acceptable.

Category 5 — Relaxation of SR Testing Limitations

Prior to placing the plant in a specified operational MODE or other condition stated in the applicability of an LCO, and in accordance with the specified surveillance frequency thereafter, current TS require establishing the OPERABILITY of each LCO-required component by meeting the SRs associated with the LCO. This usually entails performance of specified tests that demonstrate OPERABILITY of LCO-required components, or verification that specified parameters are within LCO limits or surveillance acceptance criteria. A successful demonstration of OPERABILITY requires meeting the specified test's acceptance criteria and any specified conditions for the conduct of the test.

Licensee-proposed changes involving relaxations of current SR testing limitations affect both the specified acceptance criteria and the conditions of performance, and include:

1) Removing a surveillance performance condition, such as minimum reactor coolant pump speed, and including an acceptance criterion on minimum core flow, to be consistent with the limit on minimum core flow stated in the LCO. This change has no adverse impact on plant safety because the LCO limit on minimum core flow is consistent with the core flow assumption in the applicable accident analysis.

2) Removing a surveillance performance condition that enables allowing the performance of the surveillance with the unit in MODE 1, 2, 3, or 4. For example, performing current SR 3.8.1.3 ("Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.") is only allowed if the conditions of surveillance column Note 2 are satisfied. Note 2 states,

   > "This Surveillance shall not normally be performed in MODE 1, 2, 3, or 4 unless the spare battery is connected to replace the battery being tested. However, portions of the Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the plant is maintained or enhanced."

   Stating this condition explicitly is unnecessary because the VEGP Class 1E dc electrical system design dictates that a battery service test would be performed only after the battery has been replaced by a spare battery. In the VEGP design, a spare battery bank and spare battery charger enable testing, maintenance, and equalization of battery banks offline. This configuration provides the capability for each battery bank or battery charger to be separately tested and maintained (including battery discharge tests, battery cell replacement, battery charger replacement) without limiting continuous plant operation at 100-percent power. Removing an unneeded condition restricting the performance of a SR has no adverse effect on plant safety.

The licensee's proposed changes involving relaxations of SR testing limitations do not impact safe operation of the unit because they optimize test requirements for the affected safety systems while increasing operational flexibility. Such changes are also consistent with STS guidance and the VEGP current licensing basis. The NRC staff therefore finds that licensee-proposed changes involving relaxations of SR testing limitations are acceptable.

Category 6 — Relaxation of SR Frequency

Prior to placing the plant in a specified operational MODE or other condition stated in the applicability of an LCO, and in accordance with the specified surveillance frequency thereafter, current TS require establishing the OPERABILITY of each LCO-required component by meeting the SRs associated with the LCO. This usually entails performance of specified tests that demonstrate the OPERABILITY of LCO-required components, or the verification that specified parameters are within LCO limits. A successful demonstration of OPERABILITY requires meeting the specified test's acceptance criteria and any specified conditions for the conduct of the test, at the surveillance's specified frequency (or test interval), which is appropriate for the LCO-required system's expected reliability and availability. A relaxation of a SR frequency includes decreasing its specified frequency of performance (increase the test interval). Such relaxations will optimize test requirements for the affected safety systems and also increase operational flexibility. Such changes do not adversely affect plant safety because the expected reliability and availability of the affected components are not diminished. The licensee's proposed changes involving SR frequency relaxations are also consistent with STS guidance and the VEGP current licensing basis. Therefore, the NRC staff finds that licensee-proposed changes involving SR frequency relaxations are acceptable.

Category 7 — Replacement of SR with an Equivalent Requirement

Failure to meet certain SRs in current TS for instrumentation systems (i.e., CTS 3.3.2, "Engineered Safety Features Actuation System [ESFAS] Instrumentation") may require entering an ACTIONS condition with required actions that are more restrictive than necessary to compensate for the resulting degraded capability of the affected system or component, and therefore inappropriate. Deleting such an SR and adding an equivalent requirement to the TS is a change in presentation that avoids the unintended overly restrictive required actions by more clearly conveying the existing Specification's intended limitations on plant operation in the event the surveillance is not met.

Changes involving licensee-proposed replacement of current SRs with equivalent requirements include:

1) Removing SRs associated with LCO or applicability requirements that have been removed or changed in such a way that the SR is no longer appropriate or needed. For example, explicit LCO, applicability, action, and surveillance requirements related to

RTS and ESFAS interlocks or permissive functions are proposed to be implicitly covered by the equivalent requirements for the supported RTS and ESFAS instrumentation functions, and so are deleted. Although removing these explicit requirements is considered less restrictive, in a number of cases, the proposed supported function's requirements exceed the current requirements for the supporting permissive function. Therefore, taken together, these changes have no adverse effect on plant safety.

2) Removing SR 3.3.2.7 ("Perform ACTUATION DEVICE TEST") and SR 3.3.2.8 ("Perform ACTUATION DEVICE TEST for squib valves") from current TS 3.3.2, "ESFAS Instrumentation," and Table 3.3.2-1, Function 26.a, "ESF Actuation Subsystem," and adding in their place an equivalent surveillance, with the same 24-month frequency, in the Specification for the actuated component. For example, new SR 3.4.11.4 states, "Verify each stage 1, 2, and 3 ADS valve actuates to the open position on an actual or simulated actuation signal." The current action requirements in Specification 3.4.11 are adequate to compensate for failing the new surveillance for the ADS valves. Therefore, such changes have no adverse effect on plant safety.

The currently intended operational restrictions that result from failure to meet the affected SRs are more clearly presented by new, but equivalent surveillance requirements, for which the associated currently specified action requirements are adequate to compensate for a failed test of the affected system. Accordingly, such changes will have no adverse effect on plant operational safety. The licensee's proposed changes involving replacement of current SRs with equivalent requirements are also consistent with STS guidance and the VEGP current licensing basis. Therefore, the NRC staff finds that licensee-proposed changes involving replacement of current SRs with equivalent requirements are acceptable.

<u>Category 8 — Deletion of Reporting Requirement</u>

The current TS Section 5.0, "Administrative Controls," contains various administrative control requirements to ensure safe operation of the plant, including reporting requirements in accordance with 10 CFR 50.4, "Written Communications." However, consistent with the STS, as modified by NRC approved STS change traveler, TSTF-369, "Removal of Monthly Operating Report and Occupational Radiation Exposure Report," Revision 1, the improved TS would omit the currently-specified monthly operating report and the currently-specified occupational radiation exposure report. Deletion of these two reporting requirements is beneficial because it reduces the administrative burden on the licensee and in turn allows increased attention by the licensee to facility operations important to safety. The NRC staff finds that the licensee's proposed changes involving deletion of reporting requirements will have no impact on the safe operation of the plant because they are consistent with the approved traveler, the STS, and the VEGP current licensing basis. Therefore, these changes are acceptable.

<u>Category 9 — Deletion of LCO 3.0.8</u>

Current TS LCO 3.0.8 is an AP1000 generic TS-unique provision that applies in MODES 5 and 6. It is analogous to LCO 3.0.3, which applies in MODES 1, 2, 3, and 4. Current LCO 3.0.8 applies in MODES 5 and 6 when the associated ACTIONS are not met or an associated ACTION is not provided. In some Specification subsections, LCO 3.0.8 is explicitly excluded from being applied by way of an ACTIONS Note or a Required Actions Note. In conjunction with the change to eliminate LCO 3.0.8, these Notes are no longer necessary and are eliminated. In current TS Section 3.0, LCO 3.0.8 states:

"When an LCO is not met and the associated ACTIONS are not met or an associated ACTION is not provided, action shall be initiated within 1 hour to:

a. Restore inoperable equipment and

b. Monitor Safety System Shutdown Monitoring Trees parameters

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.8 is not required.

LCO 3.0.8 is only applicable in MODES 5 and 6."

Each required action provided in individual Specification subsections that are applicable in MODE 5, MODES 5 and 6, or MODE 6 specify one of the following:

(1) Restore compliance with the LCO;

(2) Exit the Applicability; or

(3) Impose compensatory measures.

LCO 3.0.8.a imposes a "restore" action, but without a stated completion time. This action duplicates the "restore" action already imposed in various Specifications that are applicable in MODES 5 and 6, and for these Specifications provides no additional safety benefit.

The LCO 3.0.8.b action to "Monitor Safety System Shutdown Monitoring Trees parameters," is adequately addressed by the current TS 5.4.1.b requirement to implement the emergency operating procedures that implement NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33. The Shutdown Emergency Response Guidelines, outlined in the FSAR Section 19E.3.3, are captured within the development of these emergency operating procedures. The monitoring of shutdown safety status trees provides a systematic method of determining the safety status of the plant. However, this monitoring is an integral part of the operating procedures during shutdown operations. As such, the LCO 3.0.8.b monitoring requirement is redundant to monitoring required to comply with TS 5.4.1.b. Therefore, removing LCO 3.0.8 (and references to it) will not adversely impact plant safety.

The NRC staff finds that the removal of LCO 3.0.8 will have no impact on plant safety, and is consistent with the STS and the VEGP current licensing basis. Therefore, licensee-proposed changes related to removing LCO 3.0.8 from current TS are acceptable.

<u>Category 10 — Deletion of Definition of CORE ALTERATION</u>

Current TS are revised to eliminate the use of the defined term "CORE ALTERATION" and incorporate changes reflected in STS change traveler, TSTF-471-A, "Eliminate use of term CORE ALTERATIONS in ACTIONS and Notes," Revision 1. The NRC staff review of TSTF-471 (as documented in a letter dated December 7, 2006; ADAMS Accession No. ML062860320) concluded that removing from STS the MODE-6 action requirements to immediately suspend CORE ALTERATIONS is acceptable.

CORE ALTERATION is defined in current TS Section 1.1 as follows:

> "CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position."

Movement of any fuel, sources, or reactivity control components, within the reactor vessel can only occur when the reactor vessel head is removed. Therefore, action requirements related to suspending CORE ALTERATIONS only apply in MODE 6.

Two events considered in the FSAR Chapter 15 accident analyses during MODE 6 are the incontainment FHA and the boron dilution incident. (Section C of part 4.0 of this SE discusses assumptions used in the analysis of the incontainment FHA.) The suspension of CORE ALTERATIONS, except for suspension of movement of irradiated fuel, will not prevent or impair the mitigation of an incontainment FHA because the accident analyses show acceptable radiological consequences without crediting mitigation systems other than the control room ventilation filtration system.

A boron dilution incident is initiated by a dilution source which results in the boron concentration dropping below that required to maintain the required shutdown margin. In current TS 3.9.1, "Refueling Operations – Boron Concentration," which applies in MODE 6, LCO 3.9.1 states:

> "LCO 3.9.1 Boron concentration of the Reactor Coolant System (RCS), the fuel transfer canal, and the refueling cavity shall be maintained within the limit specified in COLR."

As described in the bases for current Specification 3.9.1, "plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of keff ≤ 0.95 during fuel handling with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by procedures." The boron dilution incident is mitigated by stopping the dilution. In the event that boron concentration is not within limit, current Required Action A.2 ("Suspend positive reactivity additions.") will immediately suspend irradiated fuel movement within the reactor vessel and dilution of the RCS. Therefore, the suspension of CORE ALTERATIONS (Required Action A.1) has no effect on the mitigation of a boron dilution incident.

Current TS 3.9.2, "Refueling Operations – Unborated Water Source Flow Paths," applies in MODE 6, and is provided to preclude an RCS boron dilution incident. Current LCO 3.9.2 states:

> "LCO 3.9.2 Each unborated water source flow path shall be isolated."

In the event one or more unborated water source flow paths are not isolated, current Required Action A.1 ("Suspend CORE ALTERATIONS") will immediately suspend irradiated fuel movement within the reactor vessel, but will not address the potential for dilution of the RCS represented by the unisolated flow path(s). That is addressed by Required Action A.2 ("Initiate actions to isolate flow paths.") Suspending CORE ALTERATIONS when an unborated water source flow path valve is not secured in the closed position does not provide compensation or reduce the probability of an RCS dilution event. Since the action to suspend CORE ALTERATIONS provides no safety benefit, it is not needed. Therefore, current Required Action A.1 is deleted. Also, current required Action A.2 is modified as

ITS 3.9.2 Required Action A.1 to state, “Initiate actions to ~~isolate flow paths~~ **secure one valve in the flow path in the closed position**.” This change is consistent with changes to current LCO 3.9.2 and current Condition A, made in accordance with DOC A115:

“LCO 3.9.2 ~~Each~~ **One valve in each** unborated water source flow path shall be ~~isolated~~ **secured in the closed position**.”

“Condition A. One or more **unborated water source** flow paths ~~not isolated~~ **with no valve secured in the closed position**.”

Current TS 3.9.3, “Refueling Operations - Nuclear Instrumentation,” applies in MODE 6, and is provided to alert the operator to unexpected changes in core reactivity. Current LCO 3.9.3 states:

“LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.”

In the event one required source range neutron flux monitor is inoperable, current Required Action A.1 (“Suspend CORE ALTERATIONS.”) will immediately suspend irradiated fuel movement within the reactor vessel, but will also preclude reductions in core reactivity (e.g., insertion of control rods or removal of fuel assemblies). This is overly restrictive. Therefore current Required Action A.1 is revised to require suspending positive reactivity additions only; negative reactivity additions would be allowed when one of two required nuclear instrumentation monitoring channels is inoperable. The revised action requirement is sufficient to preclude addition of fuel assemblies to the core or withdrawal of control rods from assemblies in the core, because these actions could add reactivity to the core.

The NRC staff concludes that removing CORE ALTERATION and related action requirements to suspend CORE ALTERATIONS has no adverse effect on plant safety, because the actions that remain in or are being added to the affected ACTIONS tables will require immediately suspending positive reactivity additions and initiating action to secure one valve in each unborated water source flow path in the closed position.

Suspending CORE ALTERATIONS has no effect on the initial conditions or mitigation of any design basis accident or transient. In addition, requirements to suspend CORE ALTERATIONS impose an operational burden with no corresponding safety benefit. The licensee’s proposed deletion of the definition of CORE ALTERATION is, therefore, appropriate and consistent with the STS, taking into consideration the VEGP current licensing basis. The NRC staff concludes, therefore, that the licensee’s proposed changes involving deletion of the definition of CORE ALTERATION are acceptable.

Table L attached to this SE lists the less restrictive changes to the VEGP CTS. Table L is organized in ITS section order and includes the following:

- DOC identifier;
- Summary description of each less restrictive change adopted;
- Change Type; and
- Reference to ITS and CTS requirements.

During the review of the less restrictive changes, the NRC staff issued RAIs to seek further clarification from the licensee to ensure completeness and accuracy of the proposed changes.

The NRC staff evaluated the licensee's responses to these RAIs, found the enhanced justifications consistent with the original proposal and acceptable.

Based on the above discussion, the NRC staff concludes that the less restrictive changes described in Table L are acceptable because they are consistent with current licensing practices and in compliance with the Commission's regulations.

F. Control of Specifications, Requirements, and Information Relocated from CTS

In this TS upgrade, the licensee proposes to relocate specifications, requirements, and detailed information from the CTS to licensee-controlled documents. This is discussed in Sections 4.C and 4.D of this SE. The facility and procedures described in the FSAR, TS Bases, and the TRM can be revised only in accordance with the provisions of 10 CFR 52.98, which ensure that records are maintained, and that appropriate controls are established for requirements removed from CTS.

G. Technical Evaluation Summary

The NRC staff finds that the VEGP ITS provide clearer, and more readily understandable requirements to ensure safer operation of the units. Further, based on the considerations discussed above, the NRC staff finds that the VEGP ITS satisfy the Commission's Final Policy Statement and 10 CFR 50.36.

Based on these findings, the NRC staff concludes that the proposed ITS for VEGP, as documented in the licensee's application and supplemental letters, are acceptable.

5.0 LIMITATIONS AND CONDITIONS

In reviewing the proposed ITS for VEGP, the NRC staff has relied upon the licensee's commitment to relocate certain requirements from the CTS to licensee-controlled documents as described in Table R, "Relocated Specifications" (Attachment 4 to this SE) and Table D, "Removed Details" (Attachment 5 to this SE). These tables, and Sections 4.C and 4.D of this SE, reflect the relocations described in the licensee's submittals on the TS upgrade. Such commitments from the licensee are important to the TS upgrade because the acceptability of removing certain requirements from the CTS is based on those requirements being relocated to licensee-controlled documents where further changes to the requirements will be controlled by applicable regulations or other requirements (e.g., 10 CFR 52.98). No new license condition is included in this license amendment.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 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. The NRC staff has determined that the amendment involves no significant change in the types, or no significant increase in the amounts 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 (77 FR 31662; published on May 29, 2012). 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.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted 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.

Attachments:
1. List of Acronyms, Initializations, and Abbreviations
2. Table A - Administrative Changes
3. Table M - More Restrictive Changes
4. Table R - Relocated Specifications
5. Table D – Detail Removed Changes
6. Table L - Less Restrictive Changes

LIST OF ACRONYMS, INITIALIZATIONS, AND ABBREVIATIONS

ADS	Automatic Depressurization System
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CMT	Core Makeup Tank
COL	Combined License
COLR	Core Operating Limits Report
COT	Channel Operational Test
CT	Completion Time
CTS	Current TS
CVS	Chemical and Volume Control System
DC	Design Certification
DBA	Design Basis Accident
DOC	Description of Change (from the CTS)
DVI	Direct Vessel Injection
EFPD	Effective Full-Power Days
ESF	Engineered Safety Features
ESFAS	Engineered Safety Feature Actuation System
FHA	Fuel Handling Accident
Fn	Function
FR	Federal Register
FSAR	Final Safety Analysis Report
GDC	General Design Criterion (of Appendix A to 10 CFR Part 50)
GTS	Generic TS
HX	Heat Exchanger
IRWST	In-containment Refueling Water Storage Tank
IST	Inservice Testing
ITS	Improved TS
LAR	License Amendment Request
LCO	Limiting Condition for Operation
MFIV	Main Feedwater Isolation Valve
MFCV	Main Feedwater Control Valve
MSIV	Main Steam Isolation Valve
MSSV	Main Steam Safety Valve
NRC	Nuclear Regulatory Commission
NRO	Office of New Reactors
OPDMS	On-Line Power Distribution Monitoring System
PAM	Post Accident Monitoring
PCS	Passive Containment Cooling System
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PRHR	Passive Residual Heat Removal

Attachment 1

<u>LIST OF ACRONYMS, INITIALIZATIONS, AND ABBREVIATIONS</u> (continued)

PTLR	Pressure Temperature Limits Report
PTS	Plant-specific TS
PXS	Passive Core Cooling System
RAI	Request for Additional Information
RCS	Reactor Coolant System
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RG	Regulatory Guide
RNS	Normal Residual Heat Removal System
RPV	Reactor Pressure Vessel
RTP	Rated Thermal Power
RTS	Reactor Trip System
SDM	Shutdown Margin
SE	Safety Evaluation
SG	Steam Generator
SR	Surveillance Requirement
SSCs	Structures, Systems, and Components
STS	Standard Technical Specifications (NUREG-1430; NUREG 1431)
TADOT	Trip Actuating Device Operational Test
TRM	Technical Requirements Manual
TS	Technical Specifications
TSTF	Technical Specification Task Force
VAS	Radiological Controlled Area Ventilation System
VES	Main Control Room Emergency Habitability System
VFS	Containment Air Filtration System
VFTP	Ventilation Filter Test Program