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

June 9, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295, Bin - 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING MULTIPLE TECHNICAL SPECIFICATION CHANGES (TAC NOS. MF4560 AND MF4561)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 180 to Renewed Facility Operating License No. NPF-68 and Amendment No. 161 to Renewed Facility Operating License No. NPF-81 for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2, in response to your application dated July 18, 2014, as supplemented by letters dated February 27, 2015, and May 2, 2016.

The amendments revise the technical specifications (TSs) by adopting 22 previously NRC-approved Technical Specifications Task Force (TSTF) Travelers. One proposed change is not included in these license amendments and will be addressed by further correspondence. Southern Nuclear Operating Company, Inc. (SNC) stated that these TSTF Travelers are generic changes chosen to increase the consistency between the VEGP TSs, the Improved Standard Technical Specifications for Westinghouse plants (NUREG-1431), and the TSs of the other plants in the SNC fleet.

A copy of the related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Bob Martin

Bob Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 180 to NPF-68
2. Amendment No. 161 to NPF-81
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 180
Renewed License No. NPF-68

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility), Renewed Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated July 18, 2014, as supplemented by letters dated February 27, 2015, and May 2, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications (TSs) as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-68 is hereby amended to read as follows:

   C. <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

   (a) Implementation of the amendment related to TSTF-2-A, Revision 1, shall include relocation of the Surveillance Requirement 3.8.3.7 requirements for sediment cleaning of the fuel oil storage tanks every 10 years from the TSs to a document that is controlled by the licensee pursuant to 10 CFR 50.59.

   (b) Implementation of the amendment related to TSTF-110-A, Revision 2 (TSs 3.1.4, 3.1.6, 3.2.3, and 3.2.4), shall include relocation of the surveillance frequencies related to inoperable alarms to a document that is controlled by the licensee pursuant to 10 CFR 50.59.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed NPF-68
and Technical Specifications

Date of Issuance: June 9, 2016

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 161
Renewed License No. NPF-81

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility), Renewed Facility Operating License No. NPF-81 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated July 18, 2014, as supplemented by letters dated February 27, 2015, and May 2, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications (TSs) as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-81 is hereby amended to read as follows:

   C. Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 161, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days.

   (a) Implementation of the amendment related to TSTF-2-A, Revision 1, shall include relocation of the Surveillance Requirement 3.8.3.7 requirements for sediment cleaning of the fuel oil storage tanks every 10 years from the TSs to a document that is controlled by the licensee pursuant to 10 CFR 50.59.

   (b) Implementation of the amendment related to TSTF-110-A, Revision 2 (TSs 3.1.4, 3.1.6, 3.2.3, and 3.2.4), shall include relocation of the surveillance frequencies related to inoperable alarms to a document that is controlled by the licensee pursuant to 10 CFR 50.59.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed NPF-81
and Technical Specifications

Date of Issuance: June 9, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 180

RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

LICENSE AMENDMENT NO. 161

RENEWED FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. NPF-68, page 4	License No. NPF-68, page 4
License No. NPF-81, page 3	License No. NPF-81, page 3
TSs	TSs
1.1-6	1.1-6
1.4-1	1.4-1
1.4-2	1.4-2
1.4-3	1.4-3
1.4-4	1.4-4
-	1.4-5
-	1.4-6
-	1.4-7
-	1.4-8
3.1.2-1	3.1.2-1
3.1.4-2	3.1.4-2
3.1.4-3	3.1.4-3
3.1.6-3	3.1.6-3
3.1.7-1	3.1.7-1
3.1.7-2	3.1.7-2
3.1.7-3	3.1.7-3
3.2.1-1	3.2.1-1
3.2.2-1	3.2.2-1
3.2.3-1	3.2.3-1
3.2.4-1	3.2.4-1
3.2.4-3	3.2.4-3

3.2.4-4	3.2.4-4
3.3.4-1	3.3.4-1
3.3.4-3	-
3.4.2-1	3.4.2-1
3.4.5-2	3.4.5-2
3.4.9-1	3.4.9-1
3.4.9-2	3.4.9-2
3.4.11-1	3.4.11-1
3.4.11-2	3.4.11-2
3.4.11-3	3.4.11-3
3.4.12-4	3.4.12-4
3.4.16-1	3.4.16-1
3.6.3-4	3.6.3-4
3.6.3-5	3.6.3-5
3.7.5-1	3.7.5-1
3.7.5-3	3.7.5-3
3.7.5-4	3.7.5-4
-	3.7.5-5
3.8.3-3	3.8.3-3
3.8.3-4	-
3.9.1-1	3.9.1-1
3.9.4-2	3.9.4-2
3.9.6-1	3.9.6-1
3.9.6-2	3.9.6-2
5.5-3	5.5-3
5.5-15	5.5-15
5.5-17	5.5-17
5.5-18	5.5-18

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the environmental Protection Plan.

(3) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) Deleted

(5) Deleted

(6) Deleted

(7) Deleted

(8) Deleted

(9) Deleted

(10) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training and response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for Implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy

(2) Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, pursuant to the Act and 10 CFR Part 50, to possess but not operate the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as my be produced by the operation of the facility authorized herein.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter 1 and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 161 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance requirements (SRs) contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 74. The SRs listed below shall be

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the hot zero power temperatures.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e , its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers to only the requirement to specifically determine the ability to meet the acceptance criteria.

(continued)

1.4 Frequency

DESCRIPTION (continued)

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or,

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

(continued)

1.4 Frequency

EXAMPLES
(continued)

EXAMPLE 1.4-1 SINGLE FREQUENCY

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not other wise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, then SR 3.0.4 becomes applicable, The Surveillance must be performed within the Frequency requirements of SR 3.0.2, as modified by SR 3.0.3, prior to entry into the MODE or other specified condition or the LCO is considered not met (in accordance with SR 3.0.1) and LCO 3.0.4 becomes applicable.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2 MULTIPLE FREQUENCIES

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3 FREQUENCY BASED ON A SPECIFIED CONDITION

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Not required to be performed until 12 hours after ≥ 25% RTP. ----- Perform channel adjustment.	7 days

The interval continues, whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency and the provisions of SR 3.0.3 would apply.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----------------------NOTE----------------------- Only required to be met in MODE 1. -- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-----NOTE----- Only required to be performed in MODE 1. ----- Perform complete cycle of the valve.	7 days

The interval continues, whether or not the unit operation is in MODE 1,2, or 3 (the assigned Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

(continued)

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
-------------------------NOTE------------------------- Not required to be met in MODE 3. --- Verify parameter is within limits.	24 hours

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1, 2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Core Reactivity

LCO 3.1.2 The measured core reactivity shall be within ± 1% $\Delta k/k$ of predicted values.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Measured core reactivity not within limit.	A.1 Reevaluate core design and safety analysis, and determine that the reactor core is acceptable for continued operation. AND A.2 Establish appropriate operating restrictions and SRs.	7 days 7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	B.2	Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND		
	B.3	Verify SDM is ≥ the limit specified in the COLR	Once per 12 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours
	AND		
	B.5	Perform SR 3.2.2.1.	72 hours
	AND		
	B.6	Reevaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3	6 hours
D. More than one rod not within alignment limit.	D.1.1 Verify SDM is ≥ the limit specified in the COLR. OR D.1.2 Initiate boration to restore required SDM to within limit. AND D.2 Be in MODE 3.	1 hour 1 hour 6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions within alignment limit.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.6.2	Verify each control bank insertion is within the limits specified in the COLR.	In accordance with the Surveillance Frequency Control Program
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	In accordance with the Surveillance Frequency Control Program

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 The Digital Rod Position Indication (DRPI) System and the Demand Position Indication System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each inoperable rod position indicator and each inoperable demand position indicator.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DRPI per group inoperable for one or more groups.	A.1 Verify the position of the rods with inoperable position indicators indirectly by using movable incore detectors.	Once per 8 hours
	<u>OR</u>	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. More than one DRPI group inoperable.	B.1	Place the control rods under manual control.	Immediately
	<u>AND</u>		
	B.2	Monitor and Record RCS T_{avg}	Once per 1 hour
	<u>AND</u>		
	B.3	Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors.	Once per 8 hours
	<u>AND</u>		
	B.4	Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours
C. One or more rods with inoperable DRPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	C.1	Verify the position of the rods with inoperable DRPIs indirectly by using movable incore detectors.	8 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One demand position indicator per bank inoperable for one or more banks.	D.1.1	Verify by administrative means all DRPIs for the affected banks are OPERABLE.	Once per 8 hours
	<u>AND</u>		
	D.1.2	Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.	Once per 8 hours
	<u>OR</u>		
	D.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Verify each DRPI agrees within 12 steps of the group demand position for the full indicated range of rod travel.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)

LCO 3.2.1 $F_Q(Z)$ shall be within the steady state and transient limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. $F_Q(Z)$ not within steady state limit.	A.1	Reduce THERMAL POWER $\geq$ 1% RTP for each 1% $F_Q(Z)$ exceeds steady state limit.	15 minutes
	AND		
	A.2	Reduce Power Range Neutron Flux — High trip setpoints $\geq$ 1% for each 1% $F_Q(Z)$ exceeds steady state limit.	72 hours
	AND		
	A.3	Reduce Overpower ΔT trip setpoints $\geq$ 1% for each 1% $F_Q(Z)$ exceeds steady state limit.	72 hours
	AND		
	A.4	Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ------------NOTE------------ Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ----------------------------- $F^N_{\Delta H}$ not within limits.	A.1.1	Restore $F^N_{\Delta H}$ to within limits.	4 hours
	<u>OR</u>		
	A.1.2.1	Reduce THERMAL POWER to < 50% RTP. <u>AND</u>	4 hours
	A.1.2.2	Reduce Power Range Neutron Flux — High trip setpoints to ≤ 55% RTP.	72 hours
	<u>AND</u>		
	A.2	Perform SR 3.2.2.1.	24 hours
	<u>AND</u>		
			(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)

LCO 3.2.3 The AFD shall be maintained within the limits specified in the COLR.

-----NOTE-----
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER $\geq$ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	In accordance with the Surveillance Frequency Control Program

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ----------NOTE---------- Required Action A.6 must be completed whenever Required Action A.5 is implemented. -------------------------- QPTR not within limit.	A.1	Limit THERMAL POWER to ≥ 3% below RTP for each 1% of QPTR > 1.00.	2 hours
	AND		
	A.2.1	Perform SR 3.2.4.1.	Once per 12 hours
	AND		
	A.2.2	Limit THERMAL POWER to ≥ 3% below RTP for each 1% QPTR > 1.00.	----------NOTE---------- For performances of Required Action A.2.2 the Completion Time is measured from the completion of SR 3.2.4.1. -------------------------- 2 hours
	AND		
	A.3	Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	Within 24 hours after achieving equilibrium conditions with THERMAL POWER limited by Required Actions A.1 and A.2.2

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.6	------------NOTE------------ Perform Required Action A.6 only after Required Action A.5 is completed. ------------------------------------	
		Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1.	----------NOTE---------- Only one of the following Completion Times, whichever becomes applicable first, must be met. ------------------------------ Within 24 hours after reaching RTP <u>OR</u> Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 and A.2.2
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to $\leq$ 50% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	-----NOTE----- With one power range channel inoperable, the remaining three power range channels can be used for calculating QPTR. ----- Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program
SR 3.2.4.2	-----NOTE----- Only required to be performed if input to QPTR from one or more Power Range Neutron Flux channels is inoperable with THERMAL POWER ≥ 75% RTP. ----- Confirm that the normalized symmetric power distribution is consistent with QPTR.	Once within 12 hours <u>AND</u> In accordance with the Surveillance Frequency Control Program

3.3 INSTRUMENTATION

3.3.4 Remote Shutdown System

LCO 3.3.4 The Remote Shutdown System Functions shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required Functions inoperable	A.1	Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 551°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1	Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 551°F.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One required RCS loop not in operation with Rod Control System capable of rod withdrawal.	C.1	Restore required RCS loop to operation.	1 hour
	<u>OR</u>		
	C.2	Place the Rod Control System in a condition incapable of rod withdrawal.	1 hour
D. Two required RCS loops inoperable. <u>OR</u> No RCS loop in operation.	D.1	Place the Rod Control System in a condition incapable of rod withdrawal.	Immediately
	<u>AND</u>		
	D.2	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	<u>AND</u>		
	D.3	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	In accordance with the Surveillance Frequency Control Program

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 92%; and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Pressurizer water level not within limit.	A.1	Be in MODE 3	6 hours
	<u>AND</u>		
	A.2	Fully insert all rods.	6 hours
	<u>AND</u>		
	A.3	Place Rod Control System in a condition incapable of rod withdrawal.	6 hours
	<u>AND</u>		
	A.4	Be in MODE 4.	12 hours
B. One required group of pressurizer heaters inoperable.	B.1	Restore required group of pressurizer heaters to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 92%.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	Verify capacity of each required group of pressurizer heaters is ≥ 150 kW.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each PORV and each block valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more PORVs inoperable and capable of being manually cycled.	A.1 Close and maintain power to associated block valve.	1 hour
B. One PORV inoperable and not capable of being manually cycled.	B.1 Close associated block valve. AND B.2 Remove power from associated block valve. AND B.3 Restore PORV to OPERABLE status.	1 hour 1 hour 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One block valve inoperable.	C.1	Place associated PORV in manual control.	1 hour
	<u>AND</u>		
	C.2	Restore block valve to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	Be in MODE 4.	12 hours
E. Two PORVs inoperable and not capable of being manually cycled.	E.1	Close associated block valves.	1 hour
	<u>AND</u>		
	E.2	Remove power from associated block valves.	1 hour
	<u>AND</u>		
	E.3	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.4	Be in MODE 4.	12 hours
F. Two block valves inoperable.	F.1	Restore one block valve to OPERABLE status.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Required Action and associated Completion Time of Condition F not met.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.11.1	-------------------NOTES------------------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each block valve	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify both safety injection pumps are incapable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.2	Verify each accumulator is isolated.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.3	Verify RHR suction valves are open for each required RHR suction relief valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.12.4	-----------------------NOTE----------------------- Only required to be met when complying with LCO 3.4.12.b. --- Verify RCS vent size within specified limits	In accordance with the Surveillance Frequency Control Program

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

-----NOTE-----
LCO 3.0.4c is applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 1.0 μCi/gm.	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each 24 inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition C of this LCO.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.2	Verify each 14 inch purge valve is closed, except when the associated penetration(s) is (are) permitted to be open for purge or venting operations and purge system surveillance and maintenance testing under administrative control.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.3	-----NOTE----- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---------- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	-----NOTES----- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. The fuel transfer tube blind flange is only required to be verified closed once after refueling prior to entering MODE 4 from MODE 5. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.5	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.6	Perform leakage rate testing for containment purge valves with resilient seals.	In accordance with the Surveillance Frequency Control Program
SR 3.6.3.7	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

------------------------------------NOTE------------------------------------
LCO 3.0.4b is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable. <u>OR</u> ----------NOTE---------- Only applicable if MODE 2 has not been entered following refueling. --------------------------- One turbine driven AFW pump inoperable in MODE 3 following refueling.	A.1 Restore affected equipment to OPERABLE status.	7 days
B. One AFW train inoperable for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	-----------------------NOTE----------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---	
	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.2	-----------------------NOTE----------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. ---	
	Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.7.5.3	-----NOTE----- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.4	-----NOTES----- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator. 2. AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.5	Verify that each AFW pumphouse ESF supply fan starts and associated dampers actuate on a simulated or actual actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.5.6	Verify that the ESF outside air intake and exhaust dampers for the turbine-driven AFW pump actuate on a simulated or actual actuation signal.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.8.3.1	Verify each fuel oil storage tank contains ≥ 68,000 gal of fuel.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.2	Verify lube oil inventory is ≥ 336 gal.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.3	Verify fuel oil properties of new and stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.	In accordance with the Diesel Fuel Oil Testing Program
SR 3.8.3.4	Verify each DG has one air start receiver with a pressure ≥ 210 psig.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.5	Check for and remove accumulated water from each fuel oil storage tank.	In accordance with the Surveillance Frequency Control Program
SR 3.8.3.6	Verify each DG ventilation supply fan starts and the necessary dampers actuate on a simulated or actual actuation signal.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

------------------------------NOTE------------------------------
Only applicable to the refueling canal and refueling cavity when connected to the RCS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	In accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify each required containment penetration is in the required status.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.2	-----NOTE----- Not required to be met for containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.4.c.1. ----- Verify at least two containment ventilation valves in each open containment ventilation penetration providing direct access from the containment atmosphere to the outside atmosphere are capable of being closed from the control room.	In accordance with the Surveillance Frequency Control Program
SR 3.9.4.3	-----NOTE----- Only required for an open equipment hatch. ----- Verify the capability to install the equipment hatch.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level

LCO 3.9.6 Two RHR loops shall be OPERABLE, and one RHR loop shall be in operation.

-----NOTES-----

1. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

2. All RHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:

    a. The core outlet temperature is maintained > 10 degrees F below saturation temperature.

    b. No operations are permitted that would cause a reduction of the Reactor Coolant System (RCS) boron concentration; and

    c. No draining operations to further reduce RCS water volume are permitted.

APPLICABILITY: MODE 6 with the water level < 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Less than the required number of RHR loops OPERABLE.	A.1 Initiate action to restore required RHR loops to OPERABLE status.	Immediately
	OR	
	A.2 Initiate action to establish ≥ 23 ft of water above the top of reactor vessel flange.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR loop in operation.	B.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	<u>AND</u>		
	B.2	Initiate action to restore one RHR loop to operation.	Immediately
	<u>AND</u>		
	B.3	Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify one RHR loop is in operation and circulating reactor coolant at a flow rate of ≥ 3000 gpm.	In accordance with the Surveillance Frequency Control Program
SR 3.9.6.2	-------------------NOTE------------------- An operating RHR loop will meet this requirement for the RHR loop running unless the RHR loop is in a low flow system operation. --- Verify RHR loop locations susceptible to gas accumulation are sufficiently filled with water.	In accordance with the Surveillance Frequency Control Program

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentrations stated in 10 CFR 20, Appendix B (to paragraphs 20.1001-20.2401), Table 2, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that

(continued)

5.5 Programs and Manuals (continued)

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program (continued)

4. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

5. A one time exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

   Section 9.2.3: The next Type A test, after the March 2002 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 37 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.2% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria are $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are` $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals (continued)

5.5.18 Configuration Risk Management Program

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program shall include the following elements:

a. Provisions for the control and implementation of a Level 1 at power internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.

b. Provisions for performing an assessment prior to entering the LCO Condition for preplanned activities.

c. Provisions for performing an assessment after entering the LCO Condition for unplanned entry into the LCO Condition.

d. Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Condition.

e. Provisions for considering other applicable risk significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

5.5.19 Battery Monitoring and Maintenance Program

This program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Vented Lead-Acid Batteries for Stationary Applications," of the following:

a Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 180 TO RENEWED FACILITY OPERATING LICENSE NPF-68

AMENDMENT NO. 161 TO RENEWED FACILITY OPERATING LICENSE NPF-81

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

1.0 INTRODUCTION

By letter dated July 18, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14203A124), as supplemented by letters dated February 27, 2015, and May 2, 2016 (ADAMS Accession Nos. ML15058A891 and ML16123A134, respectively), Southern Nuclear Operating Company, Inc., et al (SNC, the licensee), submitted a license amendment request (LAR) for changes to the Technical Specifications (TSs) for the Vogtle Electric Generating Plant, Units 1 and 2 (VEGP). The LAR proposed to revise the VEGP TSs to incorporate 23 generic changes that have been made to NUREG-1431, "Standard Technical Specifications - Westinghouse Plants" (STS) (WOG), since VEGP adopted improved TSs based on WOG STS, Revision 1, issued in April 1995. The changes, which are identified by Technical Specification Task Force (TSTF) Traveler numbers, are:

1. TSTF-2-A, Revision 1, "Relocate the 10 year sediment cleaning of the fuel oil storage tank to licensee control"

2. TSTF-27-A, Revision 3, "Revise SR [Surveillance Requirement] Frequency for Minimum Temperature for Criticality"

3. TSTF-28-A, Revision 0, "Delete unnecessary Action to measure gross specific activity"

4. TSTF-45-A, Revision 2, "Exempt verification of CIVs [Containment Isolation Valves] that are not locked, sealed or otherwise secured"

5. TSTF-46-A, Revision 1, "Clarify the CIV surveillance to apply only to automatic isolation valves"

6. TSTF-87-A, Revision 2, "Revise 'RTBs [Reactor Trip Breakers] open' and 'CRDM [Control Rod Drive Mechanisms] de-energized' Actions to 'incapable of rod withdrawal"

7. TSTF-95-A, Revision 0, "Revise completion time for reducing Power Range High trip setpoint from 8 to 72 hours"

Enclosure 3

8. TSTF-110-A, Revision 2, "Delete SR frequencies based on inoperable alarms"

9. TSTF-142-A, Revision 0, "Increase the Completion Time When the Core Reactivity Balance is Not Within Limit"

10. TSTF-234-A, Revision 1, "Add Action for More Than One [D]RPI [Digital Rod Position Indicator] Inoperable"

11. TSTF-245-A, Revision 1, "AFW [auxiliary feedwater] train inoperable when in service"

12. TSTF-247-A, Revision 0, "Provide separate condition entry for each PORV [Power Operated Relief Valve] and block valve"

13. TSTF-248-A, Revision 0, "Revise Shutdown Margin definition for stuck rod exception"

14. TSTF-266-A, Revision 3, "Eliminate the Remote Shutdown System Table of Instrumentation and Controls"

15. TSTF-272-A, Revision 1, "Refueling Boron Concentration Clarification"

16. TSTF-273-A, Revision 2, "Safety Function Determination Program Clarifications"

17. TSTF-284-A, Revision 3, "Add 'Met vs. Perform' to Specification 1.4, Frequency"

18. TSTF-308-A, Revision 1, "Determination of Cumulative and Projected Dose Contributions in RECP [Radioactive Effluent Controls Program]"

19. TSTF-312-A, Revision 1, "Administratively Control Containment Penetrations"

20. TSTF-314-A, Revision 0, "Require Static and Transient F_Q Measurement"

21. TSTF-340-A, Revision 3, "Allow 7-Day Completion Time for a turbine-driven AFW pump inoperable"

22. TSTF-343-A, Revision 1, "Containment Structural Integrity"

23. TSTF-349-A, Revision 1, "Add Note to LCO 3.9.5 Allowing Shutdown Cooling Loops Removal from Operation"

The U.S. Nuclear Regulatory Commission (NRC, Commission) decoupled Item No. 19 above from this review and will complete its review of TSTF-312-A by a separate amendment. Therefore, these amendments only address the remaining 22 TSTFs.

The supplement dated February 27, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 3, 2015 (80 FR 11480).

The licensee also included in the LAR its TS Bases changes for the proposed changes to the TSs. These changes were presented for information only because these changes are controlled through TS 5.5.14, "Technical Specification (TS) Bases Change Program." The NRC staff reviewed the changes to ascertain whether they were consistent with the changes in the TSTFs and were technically correct.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 requires TSs for nuclear reactors to include items in the following categories: (1) Safety limits, limiting safety system settings, and limiting control settings; (2) Limiting conditions for operation [LCOs]; (3) Surveillance requirements [SRs]; (4) Design features; and (5) Administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

On July 22, 1993, the Commission published the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132). This Final Policy Statement discussed criteria for determining which items must be included in the TS as LCOs. These criteria were subsequently incorporated into 10 CFR 50.36 (60 FR 36953). Specifically, 10 CFR 50.36(c)(2)(ii) requires that an LCO be established for each item meeting one or more of the following criteria:

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.

In adopting the Standard Technical Specifications (STS) or a TSTF traveler, licensees propose revisions to their licensed nuclear reactor facility TSs to: (1) incorporate revised NRC policy and guidance regarding the content and format of TSs, (2) clarify a specification's intent by revising ambiguous language and correcting editorial errors, and (3) enhance specifications to correct inadequacies.

In determining the acceptability of TS changes, the NRC staff considers the requirements of 10 CFR 50.36 using the guidance in the WOG STS and the associated Bases for the SLs and LCOs, and the references cited in the Bases. As discussed in the Final Policy Statement, the NRC staff reviews, on a case-by-case basis, whether enforceable regulatory controls (e.g., 10 CFR 50.59) are needed for material moved to licensee-controlled documents, such as the updated safety analysis report, the Technical Requirements Manual, the TS Bases, the Quality Assurance Plan, etc. The NRC staff determines that plant-specific adoptions of STS format and content provide continued, adequate protection for the public health and safety when (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards and the facility's licensing basis.

The NRC staff used WOG STS, Revision 4, issued in April 2012, in its review of the TS changes proposed by VEGP. The NRC staff also referred to the TSTF travelers associated with the STS changes proposed for adoption by VEGP.

3.0 TECHNICAL EVALUATION

The NRC staff notes that per the application, the licensee's review of its plant-specific justifications and the approved traveler justifications for the proposed changes discussed below found no significant differences between the licensee's proposed changes and the approved TSTFs. The evaluation below discusses the differences that the licensee identified in its application.

3.1 TSTF-2-A, Revision 1, "Relocate the 10 Year Sediment Cleaning of the Fuel Oil Storage Tank to Licensee Control"

The NRC approved this change to STS Revision 1 on July 16, 1998. This traveler modified STS 3.8.3, "Diesel Fuel Oil, Lube Oil, Starting Air, and Ventilation," by relocating SR 3.8.3.6 from the TSs to plant-controlled documents (e.g., documents controlled by 10 CFR 50.59). The SR currently requires sediment cleaning of the fuel oil storage tanks every 10 years as specified in Regulatory Guide (RG) 1.137, "Fuel-Oil Systems for Standby Diesel Generators." RG 1.137 is referenced in the VEGP Updated Final Safety Analysis Report (UFSAR) Sections 1.9 and 9.5.4.2 and in the VEGP TS 3.8.3 Bases.

The equivalent SR in the VEGP TS is numbered SR 3.8.3.7.

TS SR 3.8.3.7 currently indicates that draining of the fuel oil stored in the supply tanks, removal of accumulated sediment, and tank cleaning will be done at a frequency specified in the licensee's Surveillance Frequency Control Program (SFCP).

The proposed change removes SR 3.8.3.7 from the TSs and relocates it to a document that is controlled by the licensee under 10 CFR 50.59.

The licensee states that administrative methods will be established to control performance of the 10-year diesel fuel oil storage tank cleaning activities that are currently described in SR 3.8.3.7. This has been included in the implementation section of this license amendment.

NRC Staff Evaluation:

The NRC staff has determined that the current SR 3.8.3.7 requirements are a maintenance activity and are not a necessary surveillance to demonstrate operability of the emergency diesel generators and the quality of the system, or that a safety limit will not be exceeded as required by 10 CFR 50.36 (See Section 2.0 above) for retention in the TSs. The proper quality of fuel oil related to sediment content is demonstrated by performance of current VEGP SR 3.8.3.3, which determines whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate, detrimental impact on diesel engine combustion. As a result, adequate controls exist in the TSs to allow relocation of SR 3.8.3.7 to licensee-controlled documents.

RG 1.137 states that, as a minimum, the fuel oil stored in the supply tanks should be removed, the accumulated sediment removed, and the tanks cleaned at 10-year intervals. As stated above, VEGP Final Safety Analysis Report (FSAR) Section 1.9 incorporates the guidance of RG 1.137, which would continue to be met upon the proposed relocation of subject surveillance to a document that is controlled by the licensee pursuant to the 10 CFR 50.59 regulations.

Since SR 3.8.3.7 does not meet the 10 CFR 50.36 criteria for retention in the TSs as discussed above, there is no change in the licensee's commitment to RG 1.137, and SR 3.8.3.7 is partially duplicative of SR 3.8.3.3, which is not proposed to be relocated, the staff concludes that removal of SR 3.8.3.7 from the TSs is consistent with the requirements of 10 CFR 50.36(c)(3) and is acceptable. In addition, the change is consistent with guidance in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants, Revision 4" because TSTF-2-A changes have been incorporated into the VEGP TSs.

3.2 TSTF-27-A, Revision 3, "Revise SR Frequency for Minimum Temperature for Criticality"

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS, issued in April 2001. This traveler revised WOG STS 3.4.2, "RCS [Reactor Coolant System] Minimum Temperature for Criticality," to modify the frequency of SR 3.4.2.1.

TS SR 3.4.2.1 currently states:

> Verify RCS T_{avg} in each loop ≥ 551°F." and specified a FREQUENCY of "**Once within 30 minutes and every 30 minutes thereafter when the $T_{avg} - T_{ref}$ deviation alarm is not reset and any RCS loop T_{avg} < 561°F**.

In a two-step process, the licensee first proposes to revise the frequency from, "once within 30 minutes and every 30 minutes thereafter when the $T_{avg} - T_{ref}$ deviation alarm is not reset and any RCS loop T_{avg} < 561°F." to "once every 12 hours" in accordance with TSTF-27-A, Revision 3, which the NRC staff accepts, as discussed below. Then, in a second step, the

licensee further proposes to revise the frequency to, "In accordance with the Surveillance Frequency Control Program," in accordance with TSTF-425.

The licensee explained the differences between its proposed changes and the approved TSTFs as follows:

> The frequency for ISTS SR 3.4.2.1, and its associated Note, are modified by TSTF-27-A. The changes in TSTF-27-A would modify the Frequency for SR 3.4.2.1 to a periodic frequency of 12 hours. As described in TS 5.5.21, Vogtle has adopted a Surveillance Frequency Control Program (SFCP) to control surveillances with periodic frequencies. The Frequency for SR 3.4.2.1, as modified by the changes identified in TSTF-27-A, will become a periodic frequency, and can be controlled under the SFCP. The Frequency for SR 3.4.2.1 is therefore modified to indicate that it is, In accordance with the Surveillance Frequency Control Program. The initial Frequency for this Surveillance will be 12 hours. The changes to SR 3.4.2.1 and the Bases for this SR are modified from that in TSTF-27-A to reflect this difference. NRC approval of the license change implementing the SFCP was provided in Amendment Numbers 158/140, dated January 19, 2011 (ACN ML102520083).

NRC Staff Evaluation:

(a) Assessment for the licensee's adoption of TSTF-27-A change:

In this LAR, the licensee stated that TS 3.4.2, "RCS Minimum Temperature for Criticality," is designed to prevent criticality outside of the normal operating regime. Verification that operation is within the pressure-temperature limits report limits is required when RCS pressure and temperature conditions are undergoing planned changes. The proposed frequency of once every 12 hours is considered reasonable in view of the control room indication available to monitor RCS status. During the approach to criticality, RCS temperature is closely watched. There are indications in the control room of deviations between actual and reference RCS temperature and on low RCS temperature to alert the operator if temperature is deviating from the program value. In addition, the operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached. Therefore, the NRC staff agrees with the licensee's justification, as stated above, and concludes that the fixed "12-hour" frequency is acceptable.

(b) Assessment for the licensee's adoption of TSTF-425 program for the change:

The licensee's proposed changes to the VEGP TS are different from those contained in TSTF-27-A in that the "12-hours" is replaced with, "In accordance with the Surveillance Frequency Control Program." VEGP has adopted TSTF-425-A, which allows relocation of selected SR frequencies from the TSs to a licensee-controlled document established in accordance with the SFCP described in VEGP TS 5.5.21. The NRC staff approval of the license amendment for adoption of TSTF-425-A was provided in Amendment Nos. 158/140, dated January 19, 2011 (ADAMS Accession No. ML102520083). According to

the approved TSTF-425, all surveillance frequencies can be relocated to the licensee controlled document <u>except</u>:

- Frequencies that reference other approved programs for the specific interval (such as the Inservice Testing Program or the Primary Containment Leakage Rate Testing Program);
- Frequencies that are purely event driven (e.g., "Each time the control rod is withdrawn to the 'full out' position");
- Frequencies that are event-driven but have a time component for performing the surveillance on a one-time basis once the event occurs (e.g., "within 24 hours after thermal power reaching ≥ 95% RTP"); and
- Frequencies that are related to specific conditions (e.g., battery degradation, age, and capacity) or conditions for the performance of an SR (e.g., "drywell to suppression chamber differential pressure decrease").

The NRC staff concludes that relocation of the subject surveillance frequency is not considered to be within the scope of the TSTF-425 four exceptions (discussed above) for which the TSTF-425 allowance is not allowed. Furthermore, the staff's approval letter, dated January 19, 2011 (ADAMS Accession No. ML102520083), for the licensee's TSTF-425 program (Amendment Nos. 158/140), states:

> The licensee's adoption of TSTF-425 requires application of Nuclear Energy Institute (NEI) 04-10 in the SFCP. NEI 04-10 requires performance monitoring of structures, systems, and components (SSCs) whose surveillance frequency has been revised as part of a feedback process to assure that the change in test frequency has not resulted in degradation of equipment performance and operational safety. The monitoring and feedback includes consideration of maintenance rule monitoring of equipment performance. In the event of degradation of SSC performance, the surveillance frequency will be reassessed in accordance with the methodology, in addition to any corrective actions which may apply as part of the maintenance rule requirements. The performance monitoring and feedback specified in NEI 04-10 is sufficient to reasonably assure acceptable SSC performance and is consistent with Regulatory Position 3.2 of RG 1.177 "An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications."

The NRC staff reviewed the licensee's submittals relative to the TSTF and STS and concludes that relocation of this frequency to the licensee's SFCP meets 10 CFR 50.36 and is, therefore, acceptable.

3.3 <u>TSTF-28-A, Revision 0, "Delete unnecessary Action to measure gross specific activity"</u>

The NRC approved this change to STS Revision 1 on September 27, 1996. This traveler revised WOG STS 3.4.16, "RCS Specific Activity," to delete Required Action B.1, which

requires performance of SR 3.4.16.2 to verify that reactor coolant dose equivalent 1-131 specific activity is within the specified limits.

TS LCO 3.4.16, Required Action B.1, currently states:

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Gross specific activity of the reactor coolant not within limit.	B.1 Perform SR 3.4.16.2 AND B.2 Be in MODE 3 with T_{avg} < 500°F	4 hours 6 hours

The licensee's proposed change removes Required Action B.1 and its associated CT. The change also renumbers Required Action B.2 as B.1 with no change to its CT.

NRC Staff Evaluation:

The staff's approval for TSTF-28-A recognized that there is little, if any, safety benefit to having the plant staff implementing Required Action B.1 (i.e., performance of SR 3.4.16.2), within 4 hours, while at the same time, per Required Action B.2, bringing the reactor to MODE 3 and < 500 degrees Fahrenheit (°F) within 6 hours.

The end result of the current Required Action B.1, which is to perform SR 3.4.16.2, is governed by a Note in SR 3.4.16.2, which says it only has to be performed in MODE 1. However, it is expected that if the reactor is in MODE 1 when the current Required Action B.2 must also be implemented, then the subsequent time in MODE 1 while implementing Required Action B.2 will be short. Thus, the NRC staff concludes that continued implementation of Required Action B.1 is not necessary and that deletion of this Action from the VEGP TS is acceptable. In addition, Required Action B.1 was deleted in NUREG-1431, Revision 2, as a result of the approved TSTF-28-A.

The NRC staff reviewed the licensee's submittals relative to the TSTF and STS and concludes that the proposed changes meet 10 CFR 50.36 and are, therefore, acceptable.

3.4 TSTF-45-A, Revision 2, "Exempt verification of CIVs that are not locked, sealed or otherwise secured"

The NRC approved this change to the STS Revision 1 on July 26, 1999. This traveler revised WOG STS 3.6.3, "Containment Isolation Valves (CIVs)," by changing valve position verification SRs.

TS SRs 3.6.3.3 and 3.6.3.4 currently state:

> Verify each containment isolation manual valve and blind flange that is located outside (SR 3.6.3.3) / inside (SR 3.6.3.4) containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.

The proposed change for the SRs would add a provision as shown in bold text below:

> Verify each containment isolation manual valve and blind flange that is located outside (SR 3.6.3.3)/inside (SR 3.6.3.4) containment **and not locked, sealed, or otherwise secured** and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.

The licensee provides additional information as follows:

> In addition, the proposed change is consistent with other Vogtle Surveillance Requirements to verify the position of valves, such as SR 3.5.2.2 (Emergency Core Cooling System valves), SR 3.7.5.1 (Auxiliary Feedwater System valves), SR 3.6.6.1 (Containment Spray and Cooling System valves), SR 3.7.7.1 (Component Cooling Water System valves), SR 3.7.8.1 (Nuclear Service Cooling Water System valves), and SR 3.7.14.1 (Engineered Safety Features ESF Room Cooler and Safety-Related Chiller System valves).

NRC Staff Evaluation:

TSTF-45-A stated that the proposed change was consistent with the valve position verification requirement for valves that have a function during an accident in other system TSs. The proposed change to the SRs listed above is consistent with current VEGP TS SR 3.5.2.5, which requires that each emergency core cooling system subsystem be demonstrated operable, in part, by performance of a valve alignment of each valve that is in the flow path that is **not locked, sealed, or otherwise secured** valves. The AFW system and the nuclear service cooling water system are also demonstrated to be operable, in part, by verifying that each valve in the flow path of the system that is **not locked, sealed, or otherwise secured** in position, is in its correct position (SRs 3.7.5.1, 3.7.8.1, and 3.7.8.2, respectively). Adding these words to the SRs excludes those valves that are locked, sealed, or otherwise secured in the closed position from the verification requirements of these SRs. This is acceptable since these valves were verified to be in the correct position upon locking, sealing, or securing.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(3) continue to be met because the revised SR provides the appropriate surveillance to ensure the necessary quality of components is maintained and the LCO will be met.

Based on the above, the NRC staff concludes that the proposed TS changes are acceptable and they are also consistent with TSTF-45-A.

3.5 TSTF-46-A, Revision 1, "Clarify the CIV surveillance to apply only to automatic isolation valves"

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC in Revision 2 of the STS issued in April 2001. The TSTF deletes the WOG STS SR 3.6.3.5 requirement to verify the isolation time of "each power operated"

containment isolation valve and only requires verification of each "automatic power operated isolation valve."

TS SR 3.6.3.5 currently states:

> Verify the isolation time of each **power operated and each** automatic containment isolation valve is within limits.

The proposed change would revise the SR as follows:

> Verify the isolation time of each ~~**power operated and each**~~ automatic power operated containment isolation valve is within limits.

NRC Staff Evaluation:

The original wording states that the surveillance applied to power operated and automatic isolation valves, which could result in an interpretation that power operated valves that do not receive an automatic closure signal for design basis events are also required to have a closure time associated with them. These changes removed the unintended requirement to verify isolation times of non-automatic power operated CIVs. Appropriate changes to the Bases for Surveillance and LCO 3.6.3 were also made.

The changes proposed for the corresponding CIV SRs in VEGP TS 3.6.3 are identical to those contained in the traveler. These changes only clarify the original intended scope of containment isolation valves covered by the existing Surveillance, and hence, are administrative. The current scope of these Surveillances has not changed and there is no impact on safety.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(3) continue to be met because the revised SR provides the appropriate surveillance to ensure the necessary quality of components is maintained and the LCO will be met.

Based on the above, the NRC staff concludes that these changes are acceptable.

3.6 TSTF-87-A, Revision 2, "Revise 'RTBs open' and 'CRDM de-energized' Actions to 'incapable of rod withdrawal' "

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS issued in April 2001. This traveler modifies certain Required Actions in WOG STSs 3.4.5, "RCS Loops - Mode 3," and STS 3.4.9, "Pressurizer," as explained in the licensee's proposed changes.

The following specifies changes to LCOs 3.4.5 and 3.4.9:

- TS LCO 3.4.5 currently specifies the following Conditions and Required Actions:

Condition C currently states:

One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.

Revised Condition C would state:

One required RCS loop not in operation, ~~and reactor trip breakers closed and~~ with Rod Control System capable of rod withdrawal.

Required Action C.2 currently states:

De-energize all control rod drive mechanisms (CRDMs).

Revised Required Action C.2 would state:

~~De-energize all control rod drive mechanisms (CRDMs).~~ Place the Rod Control System in a condition incapable of rod withdrawal.

Required Action D.1 currently states:

De-energize all CRDMs.

Revised Required Action D.1 would state:

~~De-energize all CRDMs.~~ Place the Rod Control System in a condition incapable of rod withdrawal.

- TS 3.4.9 currently specifies the following Required Actions:

Required Actions A.1 and A.2 currently state:

A.1 Be in Mode 3 with reactor trip breakers open,

AND

A.2 Be in MODE 4.

Revised Required Actions:

Required Action A.1 is revised, new Required Actions A.2 and A.3 and their CTs are added, and the current Required Action A.2 is renumbered to A.4 as follows:

A.1 Be in Mode 3.

A.2 Fully insert all rods," with a Completion Time of 6 hours.

AND

A.3 Place Rod Control System in a condition incapable of rod withdrawal with a Completion Time of 6 hours,

AND

A.4 Be in MODE 4.

NRC Staff Evaluation:

The licensee's proposed changes to VEGP TSs 3.4.5 and 3.4.9 are identical to those contained in the traveler. The licensee states that the intent of these Required Actions is to prevent introduction of positive reactivity by inadvertent rod withdrawal. While the proposed changes replace the specific methods of precluding rod withdrawal, rod withdrawal remains assured of being prohibited by plant/system configuration. The specific methods are still provided in the Bases as examples to guide plant operators, if needed. In addition, the TSTF states for Westinghouse plants, that these changes are necessary to eliminate undesirable secondary effects of opening the RTBs. In particular, by opening the RTBs, plant interlock P-4 is tripped, which results in a trip of the main turbine and will close the main and bypass feedwater lines if RCS T_{avg} is below the low setpoint in Mode 3. Forcing reliance on AFW in this condition is not the intent, nor is it desirable, over continued use of normal feedwater.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above, the NRC staff concludes that this change is acceptable.

In addition, the change is consistent with guidance in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants, Revision 4," because the TSTF-87-A, Revision 2, changes have been incorporated into the VEGP TSs.

3.7 TSTF-95-A, Revision 0, "Revise completion time for reducing Power Range High trip setpoint from 8 to 72 hours"

The NRC approved this change to STS Revision 1 on September 27, 1996. This traveler revised WOG STS 3.2.1 and STS 3.2.2. The proposed change revised the CTs for TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," Required Action A.2, and TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)," Required Action A.1.2.2, to provide a 72-hour CT instead of 8 hours to reset the Power Range Neutron Flux – High Trip setpoints to a lower value.

For TS 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," when Condition A, "$F_Q(Z)$ not within steady state limit," is not met, Required Action A.2 currently states "Reduce Power Range Neutron Flux - High trip setpoints ≥ 1% for each 1% $F_Q(Z)$ exceeds steady state limit." The CT for the Action is **8 hours**.

For TS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)," when Condition A, "$F_{\Delta H}^N$ not within limits" is not met, Required Action A.1.2.2 currently states, "Reduce Power Range Neutron Flux - High trip setpoints to 55% RTP." The CT for the Action is 8 hours.

The proposed change would revise the CT to 72 hours for both Required Actions.

<u>NRC Staff Evaluation</u>:

The licensee's proposed change to the VEGP TSs is identical to that contained in TSTF-95 in that the 8-hour CT is changed to 72 hours. A CT of 72 hours will allow time to perform a second flux map to confirm the results of the first one, or determine that the condition was temporary, without implementing an unnecessary trip setpoint change, during which there is increased potential for human error and a plant transient. Following a significant power reduction, at least 24 hours is required to re-establish steady state xenon prior to taking a flux map and approximately 8 to 12 hours to obtain a flux map and analyze the data. A significant potential for human error can be created by requiring the trip setpoints to be reduced within the same timeframe that a unit power reduction is taking place and within the current 8-hour period. Setpoint adjustment of the four channels is estimated to take approximately 12 hours. Further, setpoint changes should only be required for extended operation in this condition. The licensee states that the TSTF justification is also applicable to the VEGP TS requirements.

The VEGP also has a Power Range Neutron Flux, High Positive Rate Trip to provide protection against a large positive reactivity addition event during the proposed extended time to reduce the power range neutron flux-high trip setpoints. Therefore, the NRC staff has reasonable assurance that the extended time will not adversely affect safety margin.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above, the staff concludes that the Required Actions, as modified above, are acceptable. In addition, the changes are consistent with TSTF-95, Revision 0.

3.8 <u>TSTF-110-A, Revision 2, "Delete SR frequencies based on inoperable alarms"</u>

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS issued in April 2001. The proposed change would eliminate surveillance frequencies related to certain inoperable alarms by removal from the TS of the bolded text noted below.

TS 3.1.4, "Rod Group Alignment Limits," SR 3.1.4.1 currently states:

> Verify individual rod positions within alignment limit in accordance with the Surveillance Frequency Control Program **<u>AND</u> Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable**

TS 3.1.6, "Control Bank Insertion Limits," SR 3.1.6.2 currently states:

> Verify each control bank insertion is within the limits specified in the COLR, in accordance with the Surveillance Frequency Control Program **AND Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable**

TS 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (Relaxed Axial Offset Control (RAOC) Methodology)," SR 3.2.3.1 currently states:

> Verify AFD within limits for each OPERABLE excore channel in accordance with the Surveillance Frequency Control Program **AND Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable**

TS 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," SR 3.2.4.1 currently states:

> Verify QPTR is within limit by calculation in accordance with the Surveillance Frequency Control Program **AND Once within 12 hours and every 12 hours thereafter with the QPTR alarm inoperable**

Consistent with TSTF-110 and the licensee's application, the requirement to perform the surveillances more frequently when the associated alarms are inoperable is removed from the TSs and relocated to plant administrative documents that are controlled pursuant to 10 CFR 50.59.

The licensee amplifies its proposed change with the following information:

> The Vogtle Section 3.1 specification numbers are different from the ISTS Section 3.1 specification numbers. Vogtle Specification 3.1.4, "Rod Group Alignment Limits" is equivalent to Specification 3.1.5 in the ISTS, and Vogtle Specification 3.1.6, "Control Bank Insertion Limits" is equivalent to Specification 3.1.7 in the ISTS. This has no effect on the requested change.
>
> The ISTS contains two alternative specifications for Axial Flux Difference to reflect different methodologies. TSTF-110-A revised Specification 3.2.3A, "AFD (CAOC Methodology)," and Specification 3.2.3B, "AFD (RAOC Methodology)." Vogtle Specification 3.2.3 is equivalent to ISTS Specification 3.2.3B.

NRC Staff Evaluation:

- Assessment for the Deletion of TS Rod Position Deviation Monitor in SR 3.1.4.1:

The OPERABILITY, including position indication (i.e., trippable), of the shutdown and control rods is an initial assumption in the safety analyses that assumes rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available shutdown margin (SDM). Rod position indication is required to assess operability and misalignment.

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod inoperability or misalignment may cause increased power peaking due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, control rod alignment and OPERABILITY are related to core operation in design power peaking limits and the core design requirement of a minimum SDM. Limits on control rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The licensee's proposed change to the TS SR 3.1.4.1 (i.e., to remove requirements for surveillances when the rod position deviation monitor is inoperable), is identical to that contained in the traveler. The rod position monitor/alarm is provided as an aid to alert the operator if any rod deviates from the bank position by more than ± 12 steps. This SR verifies that each control bank insertion is within the limits specified in the Core Operating Limits Report (COLR).

One of the requirements of the COLR specified in the licensee's TS Section 5.6.5 requires that:

> The core operating limits shall be determined such that all applicable· limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

The NRC staff concludes that since performance of the SR per the licensee's COLR program assures that the plant is operating within the limits established in its accident analysis, as a result, actions based on the availability of the rod position monitor are not required to be retained in the TSs; therefore, the proposed change is acceptable.

- Assessment for Deletion of TS Rod Insertion Limit (RIL) Monitor in SR 3.1.6.2:

The insertion limits of the shutdown and control rods are initial assumptions in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect core power and fuel burnup distributions and assumptions of available SDM, and initial reactivity insertion rate. TS limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved. VEGP SRs 3.1.4.1 through 3.1.4.3 associated with these TSs rely solely upon the Bank Demand Position Indication system and Digital Rod Position Indication (DRPI) system. The RIL monitor is not used in satisfying these SRs.

The RIL monitor/alarm is provided as an aid to alert the operator if the control banks are outside of the insertion limits. It performs no protective functions assumed in a safety analysis. Considering that rod motion is limited and infrequent during steady-state power operations, verification that banks are within the limits once per 12 hours allows the operator to detect a bank that is approaching the insertion limits. This specified frequency takes into account other rod position information that is continuously available to the operator in the control room so that during actual rod motion, deviations can be immediately detected.

Therefore, since the rod position indication systems provide accurate information on rod positions, the operators routinely verify rod position are within limits and the RIL monitor serves no protective function, the NRC staff has concluded that removal of the modified SR when the RIL is inoperable is acceptable. In addition, this change is consistent with NUREG-1431, Revision 1, dated April 1995, as modified by the NRC staff-approved TSTF-110, Revision 2.

- Assessment for Deletion of TS Axial Flux Difference (AFD) Monitor Alarm for SR 3.2.3.1:

The AFD is a measure of the axial power distribution skewing to either the top or bottom half of the core. The AFD is sensitive to many core-related parameters such as control bank positions, core power level, axial burnup, axial xenon distribution, and to a lesser extent, reactor coolant temperature and boron concentration.

The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

TS limits on AFD were established in order to ensure core peaking factors are consistent with the assumptions used in the safety analysis. VEGP TSs address these requirements, which require that the indicated AFD be maintained within the target band specified in the COLR), and that the AFD is verified to be within limits on a routine basis using the operable excore channels. Additionally, there is an AFD monitor/alarm, which does not perform any protective functions, to aid the operator in maintaining AFD within limits. This monitor/alarm is not relied upon for these surveillances. However, there are TS requirements to perform the AFD verifications more frequently during periods when this monitor is not operable.

The licensee has proposed revision of SR 3.2.3.1 to remove SR frequency changes related to the operability of the AFD monitor alarm, and revision of the associated bases to remove discussion of the alarm operation.

The NRC staff agrees with the licensee's position that the AFD monitor alarm is provided as an operator aid for maintaining AFD within established limits and that it performs no protective functions assumed in a safety analysis. Although AFD indication ensures that core peaking factors do not exceed assumptions in the accident analysis, the NRC staff concludes that removal of the modified SR when the AFD monitor alarm is inoperable is acceptable, since the ex-core detectors (for which no change is proposed) are capable of verifying the AFD. Therefore, the NRC staff has concluded that removal of the AFD monitor alarm requirements in SR 3.2.3.1 is acceptable. In addition, this change is consistent with NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 1, dated April 1995, as modified by the NRC staff-approved TSTF-110, Revision 2, because the TSTF changes have been incorporated into the VEGP TS.

- NRC Staff Evaluation for Deletion of TS Quadrant Power Tilt Ratio (QPTR) Monitor requirements for SR 3.2.4.1.

The licensee states that the QPTR is a measure of the radial power distribution within the core. The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analysis. The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, VEGP LCO 3.2.3, "Axial Flux Difference"; LCO 3.2.4, "Quadrant Power Tilt Ratio"; and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that power distribution remains within the bounds used in the safety analysis.

Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation. The TS limit on QPTR is addressed by an LCO and SRs, which require that the QPTR be maintained within its limit, and that this is verified on a routine basis using the excore detectors. Additionally, there is a QPTR monitor/alarm, which does not perform any protective functions, to aid the operator in maintaining QPTR within the TS limit. Although this monitor/alarm is not relied upon for the TS-required QPTR surveillances, there are TS requirements to perform the QPTR verification more frequently during periods when this monitor is not operable.

The licensee has proposed revision of SR 3.2.4.1 to remove requirements associated with the QPTR monitor alarm, and revision of the associated bases to remove discussion of alarm operability. Specifically, the SR is modified to remove the requirement to calculate QPTR and verify that it is within limits in accordance with the licensee's SFCP when the QPTR alarm is inoperable. The TS requirements to maintain QPTR less than or equal to 1.02 when reactor power is greater than 50 percent rated thermal power (RTP) and the associated action statements are unchanged.

The NRC staff agrees with the licensee's position that the QPTR monitor alarm is provided as an operator aid for maintaining QPTR within established limits and that it performs no protective functions assumed in a safety analysis. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. This change removes requirements associated with the QPTR monitor alarm, but leaves the TS QPTR limit and QPTR SRs unchanged. Per 10 CFR 50.36, SRs assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met. SR 3.2.4.1 verifies that the QPTR, as indicated by the Nuclear Instrumentation System excore channels, is within its limits. The surveillance frequency for SR 3.2.4.1 is controlled under the SFCP. Valid inputs to the detector current comparator from the upper and lower sections from three or four power range channels are required for the QPTR alarm to be OPERABLE. This requirement is not being changed in the licensee's proposed change for the SR 3.2.4.1 frequency. For those causes of QPTR that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(3) continue to be met because the revised SR provides the appropriate surveillance to ensure the necessary quality of components is maintained and the LCO will be met.

Based on the above, the NRC staff concludes that the changes are acceptable.

The changes are also consistent with guidance in the STS and TSTF-110-A, Revision 2, because the TSTF changes have been incorporated into the VEGP TS.

3.9 TSTF-142-A, Revision 0, "Increase the Completion Time When the Core Reactivity Balance is Not Within Limit"

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS issued in April 2001. This traveler extended WOG STS LCO 3.1.3 (equivalent to VEGP TS 3.1.2), "Core Reactivity," the CT from 72 hours to 7 days for specific Required Actions.

Currently, when TS 3.1.2, "Core Reactivity, Condition A," "Measured core reactivity not within limit," is not met, Required Actions A.1 and A.2 must be completed within **72 hours.**

The revised CT would be extended from 72 hours to 7 days.

NRC Staff Evaluation:

The licensee states that upon determination that the reactivity balance is not within its limits, the proposed TS change would allow VEGP TSs 7 days to reevaluate core design and safety analyses to determine if the core is acceptable for continued operation, and to establish appropriate operating restrictions and perform appropriate SRs. Under the current TSs, had the reactivity balance been outside its limits, VEGP would have to be in MODE 3 within 6 hours. VEGP justifies this extension by stating that the actions to perform an evaluation would ensure appropriate corrective actions are taken to resolve all of the issues associated with the reactivity anomaly. In TSTF-142-A, the NRC approved the 7-day period for incorporation into the STS based on the conservatisms used in designing the reactor core and performing the safety analyses, and because of the low probability of a design basis accident approaching the core design limits occurring during the proposed 7-day period.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above, and since the 7-day period allows for appropriate actions to resolve the reactivity anomaly issues, the NRC staff concludes that the proposed change is acceptable.

3.10 TSTF-234-A, Revision 1, "Add Action for More than One [Digital] Rod Position Indicator (RPI) Inoperable"

The NRC approved this change to STS Revision 1 on January 13, 1999. This traveler revised WOG STS 3.1.8 by adding a new TS Action B to allow 24 hours to restore inoperability of more than one [D]RPI in a group. VEGP TS 3.1.7, "Rod Position Indication," is equivalent to Specification 3.1.8 in the STS.

The following changes are proposed for TS 3.1.7:

- The NOTE under ACTIONS in the LCO currently states:

  Separate Condition entry is allowed for each group with no more than one inoperable rod position indicator in the group and for each bank with no more than one inoperable demand position indicator in the bank.

  The NOTE is revised to read as follows:

  Separate Condition entry is allowed for each inoperable rod position indicator and each inoperable demand position indicator.

- Required Action A.1 currently states:

  Verify the position of the rods with inoperable position indicators by using movable incore detectors.

  Revised Required Action A.1 would state:

  Verify the position of the rods with inoperable position indicators **indirectly** by using movable incore detectors.

- Required Action B.1 currently states:

  Verify the position of the rods with inoperable DRPIs by using movable incore detectors.

  Revised Required Action B.1, which is renumbered C.1, would state:

  Verify the position of the rods with inoperable DRPIs **indirectly** by using movable incore detectors.

- A new Condition B, "More than one DRPI per group inoperable." with its associated Required Actions B.1, B.2, B.3, B.4 and associated Completion Times is added, as shown below.

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. More than one DRPI per Group inoperable.	B.1 Place the control rods under manual control. AND	Immediately
	B.2 Monitor and Record RCS T_{avg}. AND	Once per 1 hour
	B.3 Verify the position of the rods with inoperable position indicators indirectly by using the movable incore detectors. AND	Once per 8 hours
	B.4 Restore inoperable position indicators to OPERABLE status such that a maximum of one DRPI per group is inoperable.	24 hours

Due to the insertion of the new Condition B, the current Conditions B, C, and D, with associated Required Actions and CTs are renumbered as Conditions C, D, and E, with associated Required Actions and CTs. The CTs for renumbered Conditions C, D, and E are unchanged.

The licensee explains the differences between the proposed changes and the approved traveler as follows:

> The Vogtle Section 3.1 specification numbers are different from the ISTS Section 3.1 specification numbers. Vogtle Specification 3.1.7, "Rod Position Indication," is analogous to Specification 3.1.8 in the ISTS.
>
> The existing Vogtle Actions Note is worded differently than the ISTS Actions Note that is modified by TSTF-234-A due to a plant specific clarification. The proposed Actions Note is identical to the wording approved in TSTF-234-A. This has no effect on the proposed change or the justification.
>
> TSTF-234-A contains the bracketed text "[, or 8.1, as applicable]" in the Bases discussion for Condition C. This change was not adopted because it is not necessary to provide direction in the Bases that all applicable Conditions must be entered.

<u>NRC Staff Evaluation</u>:

TSTF-234 allows for more than one DRPI to be inoperable for a maximum of 24 hours, given that other indirect means of monitoring changes in rod position are available. This provides sufficient time to restore operability while minimizing shutdown transients during the time that the position indication is degraded.

The proposed change is consistent with TSTF-234, which allows verification of core peaking factors and shutdown margin to satisfy the action requirements, providing the non-indicating rods have not been moved. The additional time to restore an inoperable DRPI is appropriate because the proposed action would require that the control rods be under manual control, that reactor coolant system average temperature (RCS Tavg) be monitored and recorded hourly, and that rod position be verified indirectly every 8 hours thereafter, thereby assuring that the rod alignment and rod insertion LCOs are met. Therefore, the required shutdown margin will be maintained. Given the alternate position monitoring requirement, and other indirect means of monitoring changes in rod position (e.g., alarms on average versus reference temperature deviation), a 24-hour CT to restore all but one DRPI per group provides sufficient time to restore operability while minimizing shutdown transients during the time that the position indication system is degraded.

The licensee clarified that rod position verification will be performed every 8 hours, which is consistent with TSTF-234, which also allows verification of core peaking factors and shutdown margin to satisfy the action requirements, provided the non-indicating rods have not been moved. Additionally, consistent with TSTF-234, the TSs will require that the control rods be placed under manual control, the RCS T_{avg} be monitored and recorded hourly, and the rod position to be verified indirectly every 8 hours. The proposed change requires the use of the movable incore detectors as an indirect means of monitoring changes in rod position in order to assure that the rod alignment and rod insertion LCOs are met, consistent with TSTF-234.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above, the proposed changes are acceptable to the staff.

3.11 TSTF-245-A, Revision 1, "AFW train operable when in service"

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS issued in April 2001. This traveler revised WOG STS 3.7.5. The proposed change modifies TS SRs 3.7.5.1, 3.7.5.3, and 3.7.5.4 by adding a Note stating that, "AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation."

TS SR 3.7.5.1 currently states:

> Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.

TS SR 3.7.5.3 currently states:

> Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.

TS SR 3.7.5.4 currently specifies the following NOTE:

> ------------------------------------NOTE--
> Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 900 psig in the steam generator.
> --

The proposed change to SRs 3.7.5.1, 3.7.5.3, and 3.7.5.4 would add the following NOTE:

> AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation.

The licensee explains the differences between the proposed changes and the approved traveler as follows:

> ISTS SR 3.7.5.3 contains a note stating that the SR is "Not applicable in MODE 4 when steam generator is relied upon for heat removal." The approved Traveler deletes and replaces this note. Vogtle SR 3.7.5.3 does not currently include this note, and will add the note identified in the approved Traveler under this change. Additionally, the Bases for SR 3.7.5.1 includes clarifying text that is supplemental to that provided in the ISTS. As a result of this change, the supplemental text is no longer necessary and is deleted. These differences do not affect the applicability of the traveler justification.

<u>NRC Staff Evaluation</u>:

The licensee proposed changes to the SRs stated in TS SR 3.7.5.1, SR 3.7.5.3, and SR 3.7.5.4 in order to conform to the provisions in TSTF-245, Revision 1. The proposed changes would add a note to the SRs that would allow an AFW train to be considered operable at low-power operation when its components are being operated manually for steam generator (SG) level control in MODES 1, 2, and 3, and the components can be realigned for AFW mode of operation. SNC states that there are no differences between its plant-specific justification and the approved TSTF justification.

The NRC staff-approved TSTF-245, Revision 1, dated July 3, 2003 (ADAMS Accession No. ML040611028). The TSTF evaluation includes a letter from the NRC staff to the licensee for the Indian Point Nuclear Generating plant, dated May 23, 1997, which outlines the NRC staff's position on taking credit for manual actions. The letter states in part, "In general, it is not appropriate to take credit for manual action in place of automatic action for protection of safety limits to consider equipment operable." Thus, credit for any manual actions should be part of the plant's licensing basis. In order to credit manual actions, the licensee must evaluate

physical differences between automatic and manual actions and the ability to perform the manual actions. The May 23, 1997, letter states that the NRC staff has made a determination that for the AFW system on a typical pressurized-water reactor (PWR), manual actions versus automatic operation are permissible in certain circumstances. VEGP is a PWR type plant. Accordingly, the NRC staff concludes that an AFW train may be considered operable when an operator is controlling AFW manually to maintain SG levels in the normal control band during startup, normal shutdown, and hot standby conditions.

TS SR 3.7.5.1 verifies the correct alignment for manual, power operated, and automatic valves in the AFW system water and steam supply flow paths and provides assurance that the proper flow paths will exist for AFW operation. While the licensee is utilizing AFW in manual during low-power operations (i.e., startups and shutdowns), the plant operators must take manual control of the AFW pump(s) and control valves to maintain proper SG level. In doing so, the valves may no longer be in the position assumed in the accident analyses. Since the licensee's application states that the VEGP Operating Procedures and Emergency Operating Procedures contain steps to support realignment of the AFW system from manual steam generator level control mode to the emergency operation mode, when required, the NRC staff concludes that the licensee's proposed changes are consistent with TSTF-245, Revision 1, for VEGP.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(3) continue to be met because the revised SR provides the appropriate surveillance to ensure the necessary quality of components is maintained and the LCO will be met.

Based on the above, the NRC staff concludes that the changes are acceptable.

3.12 <u>TSTF-247-A, Revision 0, "Provide separate condition entry for each PORV and block valve"</u>

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS issued in April 2001. This traveler modified WOG STS 3.4.11, "Pressurizer PORVs," to extend the Actions Note, which allows separate Condition entry for each PORV to apply also for each block valve.

TS LCO, 3.4.11, ACTIONS NOTE currently states:

> Separate Condition entry is allowed for each PORV

Revised ACTIONS NOTE would state:

> Separate Condition entry is allowed for each PORV **and each block valve**

TS 3.4.11, Condition F currently states:

> More than one block valve inoperable.

TS 3.4.11, Revised Condition F would state:

> Two block valves inoperable.

TS LCO 3.4.11, Condition F, Required Actions, and associated CTs currently state:

F.1 Place associated PORVs in manual control within 1 hour, AND

F.2 Restore one block valve to OPERABLE status within 2 hours, AND

F.3 Restore remaining block valve to OPERABLE status within 72 hours

The proposed change deletes Required Actions F.1 and F.3 and associated CTs, and renumbers Required Action F.2 as Required Action F.1 as stated below:

F.1 Restore one block valve to OPERABLE status within 2 hours

The licensee states that there are no differences between the proposed changes and the approved TSTF; however, TSTF-247-A provides options depending on the number of PORV and block valves that are included in the plant design. The design of VEGP includes two PORVs and associated block valves. The options from TSTF-247-A for plants with three PORVs and associated block valves are not adopted.

The pressurizer of the RCS includes two PORVs that can be manually controlled from the control room. The block valves are used to isolate the PORVs. This is done to stop excessive leakage through a PORV seat or stuck-open PORV to stop RCS depressurization and coolant loss.

Per the licensee's TSs, when a PORV block valve is declared inoperable per LCO 3.4.11, Condition C, Required Action C.1 requires placing the associated PORV in manual control within 1 hour AND Required Action C.2 requires restoring the block valve to operable status within 72 hours. The licensee states that these Required Actions and CTs are not being changed in this amendment.

<u>NRC Staff Evaluation</u>:

Currently, the Note in LCO 3.4.11 allows a separate condition entry for each PORV where the LCO requires each PORV to be operable. This allows each PORV to be treated separately with a separate CT for each inoperable PORV.

The proposed change adds the PORV block valves to the Note, which, along with PORVs, would also allow the PORV block valves to be treated as separate entities with a separate CT for each inoperable PORV block valve. The change simply extends the separate condition entry for the PORVs in the current TSs to the PORV block valves. This is treating the PORV block valves in the same manner as the PORVs. Since the block valves back-up the PORVs in performing the same function in the event of PORV problems, the NRC staff concludes that this proposed change is acceptable.

The proposed Condition F is modified to apply when two block valves are inoperable. The licensee also proposes to delete Required Actions F.1 and F.3 in TS 3.4.11. These Required

Actions are for the condition of more than one (revised to two) block valves inoperable and require that the associated PORVs for the inoperable block valves are placed in manual control within 1 hour (required Action F.1) and restore the remaining block valve to operable status within 72 hours (Required Action F.3). The licensee's current Conditions and Required Actions provide compensatory actions for separate condition entry for each block valve, such as when a PORV block valve is declared inoperable per LCO 3.4.11, Condition C, Required Action C.1, requires placing the associated PORV in manual control within 1 hour AND Required Action C.2 requires restoring the block valve to operable status within 72 hours. These Required Actions and CTs are not affected by the proposed changes.

When a PORV block valve is declared inoperable, there is entry into Condition C, one block valve inoperable. The Required Actions are (1) to place the associated PORV in manual control within 1 hour and (2) restore the block valve to operable status within 72 hours. These Required Actions and CTs are not being changed in this amendment. Therefore, there would be a separate entry into Condition C for each inoperable block valve requiring the associated PORV to be in manual control within 1 hour. Accordingly, the NRC staff concludes that allowing separate condition entry for PORV block valves makes Required Action F.1 and F.3 redundant to Required Actions C.1 and C.2, and is, therefore, no longer necessary. Therefore, the NRC staff concludes that a separate condition entry for the PORV block valves is acceptable. The staff further concludes that the proposed deletion of Required Actions F.1 and F.3 is acceptable.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above, the NRC staff concludes that the changes are acceptable.

3.13 TSTF-248-A, Revision 0, "Revise Shutdown Margin definition for stuck rod exception"

The NRC approved this change to STS Revision 1 on October 31, 2000. This traveler revised WOG STS 1.1. This change revises the definition of shutdown margin to eliminate the requirement that shutdown margin calculations must assume the single rod cluster control assembly (RCCA) of highest worth is fully withdrawn if all RCCAs can be verified to be fully inserted by two independent means.

TS 1.1, "Shutdown Margin" (SDM) definition currently has the following paragraph:

> All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

Proposed changes (shown in bold text below) modify the paragraph as follows:

> All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully

withdrawn. **However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation.** With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

NRC Staff Evaluation:

The proposed change modifies the TSs by changing the definition of SDM to reflect the definition in the latest revision to Westinghouse STSs NUREG-1431. The revised definition includes a provision allowing an exception to the highest reactivity RCCAs penalty if there are two independent means of confirming that all RCCAs or control rods are fully inserted in the core.

SDM is the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition if all RCCAs are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. The VEGP COLR is the unit-specific document that provides cycle-specific parameter limits for the current fuel cycle. These cycle-specific parameter limits are determined for each fuel cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the documents listed in VEGP TS Section 5.6.5. The COLR, in conjunction with the TSs, ensures for each specific fuel cycle that all parameters, including SDM, meet the licensing basis requirements. While the control rods are withdrawn from the reactor core, the required amount of SDM includes the penalty for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. Once all control rods are fully inserted into the reactor and verified by two independent means, the SDM limit in the COLR assures that adequate SDM, as assumed in the UFSAR for accidents and transients that initiate from a shutdown condition are met. Once all control rods have been verified to be fully inserted into the core, requiring the SDM calculation to include the penalty for the single control rod of highest reactivity worth fully withdrawn would be overly conservative.

Per the VEGP FSAR, Revision 19, the axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Digital Rod Position Indication (DRPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± ⅝ inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The DRPI System provides a highly accurate indication of actual control rod position but at a lower precision than the step counters. This system is based on inductive analog signals from a series of coils spaced along a hollow tube with a center to center distance of 3.75 inches, which

is six steps. However, the magnetic drive rod concentrates the magnetic lines of flux developed in the coil resulting in a change in coil output voltage when the shaft is close to it. This provides a ± 4 step accuracy with all coils operable. To increase the reliability of the system, the inductive coils are connected alternately to data systems A or B. Thus, if one system fails, the DRPI will go on half accuracy (System A failure = +10, -4 steps and System B failure = - 10, +4 steps) with an effective coil spacing of 7.5 inches, which is 12 steps. The resolution of the rod position indicator channel is ± 5 percent of span (± 7.5 in. or ± 12 steps). Deviation of any RCCA from its group by 10 percent of span (15 inches or 24 steps) will not cause power distributions worse than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span (12 steps). Therefore, since indication from one system is sufficient to maintain alignment within 24 steps, operation with one system (in the event of failure of the other) is acceptable.

The NRC staff has reviewed the independence of the control rod indication and concludes that VEGP has two independent systems that verify all rods are fully inserted.

The change in the SDM definition does not change continued compliance with all applicable regulatory requirements and design criteria (e.g., train separation, redundancy, and single failure). The change simply allows the elimination of a calculational assumption when it can be demonstrated by the two sources to not be applicable. All plant systems will continue to function as designed and all plant parameters will remain within their design limits.

Revising the TS definition of SDM would not require core designers to revise any SDM boron calculations. Rather, it would afford the analytical flexibility for determining SDM for a particular circumstance. The proposed change does not involve any change in the design, configuration, or operation of the nuclear plant. The current plant safety analyses remain complete and accurate in addressing the design basis events and in analyzing plant response and consequences. The LCOs, limiting safety system settings, and safety limits specified in the TSs are not affected by the proposed change. As such, the plant conditions for which the design basis accident analyses were performed are not changed.

Furthermore, margin of safety is related to confidence in the ability of the fission product barriers to perform their accident mitigation functions. These barriers include the fuel and fuel cladding, the RCS, and the containment and containment-related systems. The proposed changes will not impact the reliability of these barriers to function. Radiological doses to plant operators or to the public will not be impacted as a result of the proposed change. The change in the TS definition of SDM will have no impact to these barriers.

The NRC staff reviewed the licensee's submittals relative to the TSTF and STS and concludes that the proposed change meets 10 CFR 50.36 and is, therefore, acceptable.

Based on the above evaluation, the NRC staff concludes the licensee's proposed change to the TSs definition of SDM is acceptable.

3.14 TSTF-266-A, Revision 3, "Eliminate the Remote Shutdown System Table of Instrumentation and Controls"

The NRC approved this change to STS Revision 1 on September 10, 1999. This traveler revised WOG STS 3.3.4 by removing a list of Remote Shutdown System instrumentation and controls from the TSs and placing them in the TS Bases.

TS LCO 3.3.4 currently states:

> The Remote Shutdown System Functions in Table 3.3.4-1 shall be OPERABLE.

Revised TS LCO 3.3.4 would state:

> The Remote Shutdown System Functions shall be OPERABLE.

The proposed change removes the LCO Table 3.3.4-1, "Remote Shutdown System Instrumentation and Controls," from the TSs, and places the table and its contents in the TS Bases.

NRC Staff Evaluation:

The VEGP Remote Shutdown System provides the operator with sufficient instrumentation and controls to place and maintain the unit in a safe shutdown condition from a location other than the control room. This capability is necessary to protect against the possibility that the control room becomes inaccessible.

If the control room becomes inaccessible, the operators can establish control at the remote shutdown panel and place and maintain the unit in MODE 3. Not all controls and necessary transfer switches are located at the remote shutdown panel. Some controls and transfer switches will have to be operated locally at the switchgear, motor control panels, or other local stations. Table 3.3.4-1 lists the readout location, transfer switch location, and controls location for remote shutdown instrumentation. The proposed change would relocate these requirements to the TS licensee-controlled documents.

GDC-19 requires that remote shutdown capability be provided. The VEGP FSAR, Revision 19, Chapter 3, discusses the licensee's position on GDC 19 and states:

> In the event that the operators are forced to abandon the control room, panel-mounted instrumentation and controls are provided on the train-related shutdown panels to achieve and maintain the plant in the safe shutdown condition.

Therefore, the NRC staff concludes that the relocation of instrumentation listed in TS Table 3.3.4-1 to the TS Bases will continue the licensee's compliance with its statement regarding operations of its remote shutdown functions, as the licensee has not proposed any change to its plans in this regard. The definition of "operable" in the VEGP specifications states that a system shall be operable or have operability when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, electrical power,

cooling and seal water, lubrication, and other auxiliary equipment that are required for the system to perform its specified safety function(s) are also capable of performing their related support function. This definition requires that all instrumentation and controls necessary for the remote shutdown function be operable in order for the Remote Shutdown System LCO to be met. The ability to transfer control of a function from the main control room to the remote shutdown panel is a required support function by the TS definition of operability. Therefore, LCO 3.3.4 is sufficient to ensure that the instruments and control circuits will be OPERABLE if unit conditions require that the Remote Shutdown System be placed in operation. SR 3.3.4.2 still requires the local panel transfer function to be tested, which is sufficient to assure that the system will be operable.

The relocation of the Remote Shutdown System table of instrumentation and controls from the TSs to the Bases is acceptable because it will be adequately controlled by NRC requirements in the TS 5.5.14 Bases control program. This approach provides an effective level of regulatory control and provides for a more appropriate change control process.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above, the NRC staff concludes that the changes are acceptable.

3.15 TSTF-272-A, Revision 1, "Refueling Boron Concentration Clarification"

The NRC approved this change to STS Revision 1 on December 21, 1999. This traveler revised WOG STS 3.9.1, "Boron Concentration," to add an Applicability Note to clarify that boron concentration limits do not apply to the refueling canal and the refueling cavity when those volumes are not connected to the RCS.

TS LCO 3.9.1 currently states:

> Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

The applicability of LCO 3.9.1 is revised by the addition of a note as follows:

> -- NOTE --
> Only applicable to the refueling canal and refueling cavity when connected to the RCS.
> --

NRC Staff Evaluation:

The licensee states that TS 3.9.1 limits the boron concentrations of the RCS, the refueling canal, and the refueling cavity during refueling to ensure that the reactor remains subcritical during MODE 6. The staff concludes that the proposed change is acceptable because boron concentration limits do not apply to the refueling canal and refueling cavity when these areas are not connected to the RCS, because any water in the refueling canal and refueling cavity

would not be in communication with the reactor fuel. For MODE 6, current VEGP TS SR 3.9.1.1 ensures that the coolant boron concentration in all filled portions of the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration of the coolant in each volume is determined periodically by chemical analysis. The licensee's revised SR 3.9.1.1 TS Bases state that if any dilution has occurred while the cavity or canal were disconnected from the RCS, this SR would ensure the correct boron concentration prior to communication with the RCS.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above evaluation, the NRC staff concludes the licensee's proposed change to the TSs is acceptable. In addition, the proposed change is consistent with guidance in the STS and approved TSTF-272.

3.16 <u>TSTF-273-A, Revision 2, "Safety Function Determination Program Clarifications"</u>

The NRC approved TSTF-273-A, Revision 2, to STS Revision 1, as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated August 16, 1999. This traveler modified STS 5.5.15, "Safety Function Determination Program (SFDP)," which implements the requirements of LCO 3.0.6.

TS 5.5.15, "Safety Function Determination Program (SFDP)," contains the following two separate paragraphs:

> A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable.
>
> The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

The proposed change adds the following statements (shown in bold) in those paragraphs:

> A loss of safety function exists when, assuming no concurrent single failure, **no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s),** a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable.
>
> The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and

> Required Actions of the LCO in which the loss of safety function exists are required to be entered. **When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.**

The licensee explains the proposed changes as follows:

> The proposed TS changes adds explanatory text to the LCO 3.0.6 Bases clarifying the appropriate LCO for loss of function, and that consideration does not have to be made for a loss of power in determining loss of function. Explanatory text is also added to the programmatic description of the Safety Function Determination Program (SFDP) in Specification 5.5.15 to provide clarification of these same issues.

<u>NRC Staff Evaluation</u>:

SNC proposed to revise TS 5.5.15 by incorporating TSTF-273-A, Revision 2, changes without deviations. The changes to TS 5.5.15, and the Bases for LCO 3.0.6, are intended to clarify the intent of LCO 3.0.6 in the event a single inoperable TS support system makes both redundant subsystems of a supported system inoperable (a loss of safety function condition). The STSs were developed such that the LCO Actions for a single support system inoperability would be addressed by that support system's Actions, without cascading to the supported system's LCO Actions. LCO 3.0.6 establishes this exception to LCO 3.0.2 for support systems that have an LCO specified in the TSs. However, LCO 3.0.6 also requires an evaluation under the SFDP to ensure that a loss of function does not exist.

The staff concludes that these changes do not affect the design, operation, or maintenance of VEGP, but only add clarification for determining when a loss of safety function condition exists and what LCO Actions are required to be taken when a safety function is lost. By clarifying the intent of the existing requirements of the SFDP and LCO 3.0.6, these changes remove an ambiguity that could lead to a misinterpretation of those requirements.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above, the NRC staff concludes the licensee's proposed change to the TSs is acceptable.

3.17 <u>TSTF-284-A, Revision 3, "Add 'Met vs. Perform' to Technical Specification 1.4, Frequency"</u>

TSTF-284-A, Revision 3, was approved by the NRC as documented in a letter from William Beckner (NRC) to James Davis (NEI), dated February 16, 2000 (ADAMS Accession No. ML003684596). The change inserts a discussion paragraph into Specification 1.4, and several new examples are added to facilitate the use and application of SR Notes that utilize the

terms "met" and "perform." The changes also modify certain SRs to appropriately use the "met" and "perform" exceptions.

(a) TS 1.4 "Frequency," third paragraph, currently states:

> Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

Revised TS 1.4 replaces the paragraph in its entirety with the following:

> Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.
>
> Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.
>
> The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." The word "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria. Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:
>
> a. The Surveillance is not required to be met in the MODE or other specified condition to be entered; or
> b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or
> c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

(b) TS 1.4, "Frequency," currently includes examples up through Example 1.4-3 regarding the Frequency Based on a Specified Condition.

The change would add Examples 1.4-4, 1.4-5, and 1.4-6 on SRs, as shown in revised TS pages 1.4-6, 1.4-7, and 1.4-8 in the preceding attachment to this license amendment.

(c) TS SR 3.4.11.1 currently specifies the following requirement:

SURVEILLANCE	FREQUENCY
SR 3.4.11.1--------------------NOTE------------------------ Not required to be performed with block valve closed in accordance with the Required Action of Conditions A, B, or E. -- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2 Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program

The change would add a NOTE (shown in bold) in SRs 3.4.11.1 and 3.4.11.2 as below:

SURVEILLANCE	FREQUENCY
SR 3.4.11.1--------------------NOTES------------------------ 1. Not required to be performed with block valve closed in accordance with the Required Action**s of this LCO.** **2. Only required to be performed in MODES 1 and 2.** --- Perform a complete cycle of each block valve.	In accordance with the Surveillance Frequency Control Program
SR 3.4.11.2--------------------NOTE------------------------ **Only required to be performed in MODES 1 and 2.** ------------------- ------------------------- Perform a complete cycle of each PORV.	In accordance with the Surveillance Frequency Control Program

(c) TS SR 3.4.12.4 NOTE currently states:

Only required to be **performed** when complying with LCO 3.4.12.b.

Revised TS SR 3.4.12.4 NOTE would state:

> Only required to be **met** when complying with LCO 3.4.12.b.

(d) TS SR 3.9.4.2 NOTE currently states:

> Only required for unisolated penetrations

Revised TS SR 3.9.4.2 NOTE would state:

> Not required to be met for containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.4.c.1.

The licensee explains the proposed changes as follows:

> The change inserts a discussion paragraph into Specification 1.4, and several new examples are added to facilitate the use and application of SR Notes that utilize the terms "met" and "perform." The changes also modify SR 3.4.11.1, SR 3.4.11.2, SR 3.4.12.4, and SR 3.4.9.2 to appropriately use "met" and "perform" exceptions.

The licensee explains the differences between the proposed changes and the approved traveler as follows:

> TSTF-284-A, Revision 3 includes changes to SR 3.1.11.1 and SR 3.1.11.2 of ISTS Specification 3.1.11, "SDM Test Exceptions." This LCO allows suspension of SDM requirements in MODE 2 provided specific conditions are met in order facilitate measurement control rod worth and SDM. The Vogtle Technical Specifications do not include a Specification that is analogous to ISTS TS 3.1.11, "SDM Test Exceptions," or SRs that are analogous to ISTS SRs 3.1.11.1 and 3.1.11.2. Therefore, the TS and Bases changes identified in TSTF-284-A for ISTS 3.1.11 are not adopted.
>
> Changes to the Actions Bases for Specification 3.4.11, "Pressurizer PORVs," are not adopted. The changes described in the TSTF are related to a Note in the ISTS that provides an exception to LCO 3.0.4 that allows entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves in order to demonstrate their operability. Consistent with NUREG-1431, Vogtle Technical Specification 3.4.11, and its associated Bases, do not include the Note providing this exception to LCO 3.0.4.
>
> The Bases changes identified in TSTF-284-A for SR 3.4.12.8 is not adopted. The Bases descriptions for corresponding Vogtle SR 3.4.12.4 is substantially different from the Bases text in TSTF-284-A, which is based on NUREG-1431, Revision 1. These differences result from the adoption of a Surveillance Frequency Control Program (SFCP), as described in TS 5.5.21, to control

periodic surveillance frequencies. Adoption of the SFCP included deletion of Bases text that provided the basis for surveillance frequency if control of the frequency had been moved to the SFCP. NRC approval of the license change implementing the SFCP was provided in Amendment Numbers 158/140, dated January 19, 2011 (ACN ML102520083).

NRC Staff Evaluation:

The change in TSTF-284, Revision 3, is to modify Improved Technical Specifications, Section 1.4, "Frequency," to clarify the usage of the terms "met" and "performed" to facilitate the application of SR Notes. Three new SR Examples, 1.4-4, 1.4-5 and 1.4-6, are added to illustrate the application of the terms.

STS Section 1.4, "Frequency," defines the proper use and application of SR frequency requirements in the STS format. It states that, "An understanding of the correct application of the specified frequency is necessary for compliance with the SR." It also establishes that, "The specified frequency consists of the requirements of the frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements." The purpose of TSTF-284, Revision 3, was to clarify, and make consistent, the use of Notes in the Surveillance column that modify frequency requirements as discussed below.

Specifically, this TSTF added the following language already contained in the two Boiling Water Reactor (BWR/4 and BWR/6) NUREGS, to STS Section 1.4 of the three PWR NUREGS (Babcock & Wilcox, Westinghouse and Combustion Engineering Pressurized Water Reactors, VEGP is a Westinghouse PWR facility):

> Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. [SR 3.0.1 states "SRs shall be met during Modes or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR."] They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.
>
> Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. [SR 3.0.4 states "Entry into a MODE or other specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency."] To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.
>
> The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met."

"Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

In addition, this change replaced existing language in all five NUREGS with the following clarification:

Some Surveillances contain Notes that modify the Frequency of performance or the conditions during which acceptance criteria must be satisfied. For these Surveillances, the Mode-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a Mode or other specified condition in the Applicability of the associated LCO if any of the following are satisfied:

- The Surveillance is not required to be met in the Mode or other specified condition to be entered; or
- The Surveillance is required to be met in the Mode or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or
- The Surveillance is required to be met, but not performed, in the Mode or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

VEGP's proposed changes stated under the "licensee's proposed changes," above, refer to the NUREG guidance as mentioned above.

In adopting TSTF-284, Revision 3, a licensee must verify that Notes such as illustrated by these four examples are used properly and only as necessary. This includes ensuring the associated Bases are also correct. Proper application of these Notes in the individual SRs in the NUREGs was verified. The staff concludes that the proposed changes VEGP's adoption of the TSTF changes conforms to the guidance provided in the NUREGs described above. These changes are administrative in nature because they only serve to clarify the meanings of the terms "met" and "performed" as used in SR Notes throughout the VEGP TSs. This change serves to improve TS usefulness by clarifying terminology usage and providing additional examples of the application of SR Notes. Therefore, these changes are acceptable.

In addition to the changes discussed above, the licensee also proposed a change to the NOTE in SR 3.4.11.1, which currently states that the SR is not required to be performed with block valve closed in accordance with the Required Action of Conditions A, B, or E. The proposed change concerns deletion of Conditions A, B, or E from the NOTE in the SR. These conditions provide actions for the inoperability of one or more PORVs.

<u>NRC Staff Assessment</u>:

The staff reviewed the licensee's proposed change, the approved TSTF-284, and the STS. The staff considers the testing of the PORV block valve to be unnecessary, if the block valve has

been closed due to an inoperable PORV. The additional assurance of block valve operability gained from the surveillance test is outweighed by the risk associated with the development of an unisolable leak in the RCS. Accordingly, the provision of TSTF-284, Revision 3, which allows the extension of the range of circumstances under which the surveillance testing of the PORV block valve is not required for any of the Actions of TS 3.4.11, instead of limiting to Conditions A, B or E as currently specified in the SR, is acceptable as a change to VEGP TSs. Therefore, the change to remove the surveillance testing under the specific conditions of the revised PORV block valve SR 3.4.11.1 of TS 3.4.11 is acceptable for VEGP.

Lastly, the staff review of the licensee's proposed addition of a new NOTE that SRs 3.4.11.1 and 3.4.11.2 are only required to be performed in MODES 1 and 2 is acceptable, since it allows the test to be performed in MODE 3 under operating temperature and pressure conditions prior to entering MODES 1 or 2.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO, and the appropriate remedial measures are specified if the LCO is not met.

Based on the above evaluation, the NRC staff concludes the licensee's proposed change to the TSs is acceptable. These changes are also consistent with the approved TSTF-284 changes as discussed in paragraph (c) above.

3.18 TSTF-308-A, Revision 1, "Determination of Cumulative and Projected Dose Contributions in RECP"

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS issued in April 2001. This traveler modified WOG STS 5.5.4, "Radioactive Effluent Controls Program," to describe the original intent of the dose projections.

TS 5.5.4, paragraph 'e' currently states:

> Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

Proposed change revises paragraph 'e' in its entirety as follows:

> Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days;

NRC Staff Evaluation:

Section 50.36(a) of 10 CFR, "Technical specifications on effluents from nuclear power reactors," requires each licensee to submit a report to the NRC that will allow an estimation of the maximum potential annual radiation doses to the public resulting from effluent releases.

Generic Letter (GL) 89-01, "Implementation of Programmatic and Procedural Controls for Radiological Effluent Technical Specifications," provides guidance in support of implementing programmatic controls in TSs for radioactive effluents and for radiological environmental monitoring that conforms to the applicable regulatory requirements. The regulation in 10 CFR 20.1302, "Compliance with dose limits for individual members of the public," paragraph (b), requires that a licensee show compliance with the annual dose limit in 10 CFR 20.1301, "Dose limits for individual members of the public," demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit.

GL 89-01 combined two SRs, the cumulative and projected dose determinations, into one program element. In combining these requirements, the new program element can be interpreted to require determining projected dose contributions for the calendar quarter and current calendar year every 31 days. This wording was misleading and resulted in misinterpretation of the intent of the original STS and was not consistent with the original surveillance. Therefore, TSTF-308-A was developed and subsequently approved by the NRC to not require dose projections for a calendar quarter and a calendar year every 31 days (i.e., to describe the actual intent of the dose projections).

VEGP TS 5.5.4, "Radioactive Effluent Controls Program," states that:

> The program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded.

TS 5.5.4.e is one of the elements in the program, which states:

> Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

The NRC staff reviewed the clarifications proposed by the licensee that were prepared using the guidance in TSTF-308 and concludes that the revised wording for the VEGP TSs clarifies the reporting requirements for projected doses, thus ensuring consistency with the requirements of 10 CFR 50.36a and 10 CFR 20.1302, and are, therefore, acceptable.

3.19 <u>TSTF-312-A, Revision 1, "Administratively Control Containment Penetrations"</u>

The NRC staff decoupled TSTF-312 from its review under the current request to be processed by a separate amendment.

3.20 <u>TSTF-314-A, Revision 0, "Require Static and Transient F_Q Measurement"</u>

The NRC approved this change to STS Revision 1 on January 13, 1999. This traveler revised WOG STS LCO 3.1.5, "Rod Group Alignment Limits," and LCO 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," to require measurement of both the steady state and transient portions of the Heat Flux Hot Channel Factor, $F_Q(Z)$. VEGP TS LCO 3.1.4, "Rod Group Alignment Limits," is equivalent to STS 3.1.5.

- Currently, when TS LCO 3.1.4, Condition B, for one rod not being within alignment limits, is not met, Required Action B.4 requires performing SR 3.2.1.1 to verify the **steady state** value of Fq(Z).

  Revised Required Action B.4 by adding a requirement to perform SR 3.2.1.2 to verify the **transient value** of Fq(Z)

- Similarly for TS LCO 3.2.4, when Condition A of QPTR is not within its limit, Required Actions A.3 and A.6 currently require only the performance of SR 3.2.1.1 to verify the **steady state** value of Fq(Z).

  Revised Required Actions A.3 and A.6 by adding a requirement to perform SR 3.2.1.2 to verify the **transient value** of Fq(Z).

<u>NRC Staff Evaluation</u>:

The licensee proposes to add Required Actions to perform SR 3.2.1.2, which requires confirming that the transient component of the heat flux hot channel factor, Fq(z), is within its limits, to Condition B associated with TS LCO 3.1.4, "Rod Group Alignment Limits," and to Condition A associated with LCO 3.2.4, "Quadrant Power Tilt Ratio." The NRC staff review considers the acceptability of the proposed additions.

As the licensee notes, Fq(z) is approximated by both a steady-state and a transient component of Fq. The proposed Required Actions will modify the TS to include, along with the present requirement to verify the steady-state Fq is within limits, a new requirement to verify that the transient Fq. In conditions in which other LCOs may not be met (i.e., the LCOs for Rod Group Alignment Limits and QPTR), this added Required Action, in addition to other Required Actions in the specified condition, would ensure that the power distribution remains appropriately limited within the bounds of the safety analyses.

Since Fq(z) is approximated by both a transient and a steady-state component, the additional SR to confirm that the transient component is within its limits is consistent with 10 CFR 50.36 requirements. Specifically, the requirement to perform the proposed SRs will provide an additional remedial action to follow until the LCO can be met, consistent with

10 CFR 50.36(c)(2)(i). Since the proposed change will add a new remedial action, it is more restrictive than the existing TS requirements.

Based on the considerations that (1) the proposed Required Actions will be more restrictive than the existing TS, and (2) the proposed Required Actions are consistent with 10 CFR 50.36 requirements, the NRC staff determined that the addition of SR 3.2.1.2 to the Required Actions associated with Condition B for TS LCO 3.1.4, and Condition A for TS LCO 3.2.4, is acceptable.

3.21 TSTF-340-A, Revision 3, "Allow 7 Day Completion Time for a turbine-driven AFW pump inoperable"

The NRC approved this change to STS NUREG-1431, Revision 1, on March 16, 2000. This traveler revised WOG STS 3.7.5, "Auxiliary Feedwater System," to extend Condition A CT to 7 days to restore an inoperable turbine-driven AFW steam supply, and to expand Condition A by adding an OR statement and a NOTE in the Condition. The added statement states, "One turbine driven AFW pump inoperable in MODE 3 following refueling," and the NOTE states, "Only applicable if MODE 2 has not been entered following refueling."

(As shown below, VEGP TS LCO 3.7.5 currently allows a CT of 7 days; therefore, this part of the TSTF change is not adopted by the licensee.)

VEGP TS 3.7.5, "Auxiliary Feedwater System," Condition A currently requires:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days

The proposed change would add a statement and a NOTE in Condition A as shown in bold below:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable. **OR** **--------------NOTE -------** **Only applicable if MODE 2 has not been entered following refueling.** **-----------------------------** **One turbine driven AFW pump inoperable in MODE 3 following refueling.**	A.1 Restore **affected equipment** to OPERABLE status.	7 days

NRC Staff Evaluation:

As stated above, the current VEGP TS already include the 7-day CT for one steam supply to a turbine driven AFW pump inoperable; therefore, that part of TSTF-340 is not the subject of this review.

However, the licensee proposed to modify TS 3.7.5, Condition A, so as to allow a 7-day CT for the turbine-driven AFW pump being inoperable in Mode 3, immediately following a refueling outage, if Mode 2 has not been entered. The purpose of the AFW system is to provide cooling water to the SGs for removal of decay heat when main feedwater flow is not available. This change was proposed on the basis that it could reduce the number of unnecessary mode changes by providing added flexibility in Mode 3 to repair and test the turbine-driven AFW pump, if the pump were to be declared inoperable following a refueling outage. In the proposed condition, there is minimal decay heat for the AFW system to have to remove through the SGs due to the time that irradiated fuel has decayed during the refueling outage and the replacement of irradiated fuel with un-irradiated fuel. The NRC staff agrees with that basis and concludes that the change is reasonable, given the redundant capabilities afforded by the AFW system, the time needed to perform repairs and testing of the turbine-driven pump, and the low probability of an accident occurring during this time period that would require the operation of the turbine-driven pump. In addition, there are alternate methods, such as feed and bleed, available to remove decay heat if necessary.

The NRC staff concludes that the requirements of 10 CFR 50.36(c)(2) continue to be met, because the minimum performance level of equipment needed for safe operation of the facility is contained in the LCO and the appropriate remedial measures are specified if the LCO is not met.

Based on the above evaluation, the staff concludes that this change is acceptable. It is also consistent with guidance in the STS and TSTF-340-A, Revision 3, because the applicable TSTF-340 changes have been incorporated into the VEGP TSs.

3.22 TSTF-343-A, Revision 1, "Containment Structural Integrity"

The NRC approved this change to STS, Revision 1, on December 6, 2005. This traveler revised WOG STS 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program," and STS 5.5.16, "Containment Leakage Rate Testing Program," for consistency with the requirements of 10 CFR 50.55a(g)(4). Per the application, (a) no change to VEGP TS 5.5.6 is proposed, and (b) VEGP TS 5.5.17 is equivalent to STS 5.5.16.

TS 5.5.17 currently specifies a program for the implementation of leakage rate testing of the containment with three exemptions.

The proposed change would add a fourth exception to the program as follows:

> The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME

> Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

NRC Staff Evaluation:

The licensee's proposed changes to the VEGP TS 5.5.17 are different from those contained in the traveler in that only a portion of these changes related to the visual inspection of the steel liner plate inside Containment will be adopted by this LAR. This was the subject of a request for additional information, and the licensee responded in its letter dated February 27, 2015. VEGP has previously adopted the other proposed changes by TSTF-343-A related to inspections of the containment tendons and the outside containment concrete surfaces. The NRC staff approval of the license amendment for adoption of the changes related to containment tendons was provided in Amendment Nos. 147/127, dated December 12, 2006 (ADAMS Accession No. ML062970484), and the license amendment for adoption of changes related to containment concrete surfaces was provided in Amendment Nos. 122/100, dated June 6, 2001 (ADAMS Accession No. ML011570674).

Specifically, the licensee proposed to change TS 5.5.17 by adding the following to the RG 1.163 exception list:

> The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

The proposed change would require the steel liner plate visual examination be performed pursuant to ASME Code, Section XI, Subsection IWE, rather than the visual inspection guidelines in RG 1.163. Subsection IWE requires the licensee to perform the general visual examinations of the containment liner three times in a 10-year interval, which is consistent with the current requirement specified in RG 1.163. The requirements for inspection in Subsection IWE of the ASME Code, Section XI, are more rigorous than those currently provided in RG 1.163. For the inspection of Class MC and metallic liners of Class CC components, the ASME Code requires that the examiner be knowledgeable in the requirements for design, inservice inspection, and testing of the components, and that examinations be performed by an examiner with visual acuity sufficient to detect evidence of degradation. In addition, Subsection IWE requires that the visual examinations be reviewed by an inspector employed by a State or municipality of the United States or an inspector regularly employed by an insurance company authorized to write boiler and pressure vessel insurance. The NRC staff concludes that the licensee is adopting more rigorous inspection requirements for the steel liner plate.

The NRC staff reviewed the licensee's submittals relative to the TSTF and STS and concludes that the proposed change meets 10 CFR 50.36.

Based on the above, the NRC staff concludes that the proposed TS change is acceptable.

3.23 TSTF-349-A, Revision 1, "Add Note to LCO 3.9.6 Allowing Shutdown Cooling (SDC) Loops Removal from Operation"

The NRC did not issue a letter approving this change to STS Revision 1; however, this change was incorporated by the NRC into Revision 2 of the STS issued in April 2001. This traveler revised WOG STS 3.9.6, "RHR and Coolant Circulation - Low Water Level," to add a Note to the LCO statement allowing securing the operating train of residual heat removal (RHR) for up to 15 minutes to support switching operating trains.

The proposed change adds a NOTE in TS LCO 3.9.6 as follows:

> All RHR pumps may be de-energized for ≤ 15 minutes when switching from one train to another provided:
>
> a. The core outlet temperature is maintained > 10 degrees F below saturation temperature;
> b. No operations are permitted that would cause a reduction of the Reactor Coolant System (RCS) boron concentration; and
> c. No draining operations to further reduce RCS water volume are permitted.

NRC Staff Evaluation:

The RHR system is used to remove core decay heat and reactor coolant sensible heat during unit cooldown and cold shutdown and to provide adequate mixing of borated coolant. Currently, VEGP TS 3.9.6 requires two RHR loops to be operable and one in operation when a unit is in Mode 6 with < 23 feet of water above the top of the reactor vessel flange. The existing LCO 3.9.6 also contains a Note that allows operational status changes in the RHR system to support surveillance testing.

With the adoption of TSTF-349-A, the licensee proposed to add a second Note to allow all RHR pumps to be de-energized for up to 15 minutes when switching from one train to another. This is a short period of time to be in without coolant flow through the reactor core. The new Note includes three restrictions, as stated above, when entering this condition. These restrictions will minimize the risk while switching trains and will improve the likelihood that RHR will be safely restored.

Based on the above evaluation, which notes the short duration and the three limitations, the NRC staff concludes that the proposed changes meet the requirements of 10 CFR 50.36 and are acceptable. The proposed changes are consistent with the guidance in the STS, and TSTF-349-A, Revision 1, since the two notes in the TSTF have been incorporated into the VEGP TSs.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (80 FR 11480). Accordingly, the amendments meet 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 amendments.

7.0 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: June 9, 2016

A copy of the related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Bob Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 180 to NPF-68
2. Amendment No. 161 to NPF-81
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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