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

April 29, 2015

Mr. Dennis L. Koehl
President and CEO/CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: APPROVAL OF CHANGE TO TECHNICAL SPECIFICATION 3.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION (TAC NOS. MF3319 AND MF3320)

Dear Mr. Koehl:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 205 to Facility Operating License No. NPF-76 and Amendment No. 193 to Facility Operating License No. NPF-80 for the South Texas Project, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 6, 2014, as supplemented by letters dated June 9, December 4, and December 17, 2014.

The amendments revise the TSs with respect to the required actions and allowed outage times for inoperable reactor trip breakers. Specifically, TS 3.3.1, "Reactor Trip System Instrumentation," operator actions are revised to reduce unnecessary shutdowns and increase operational flexibility by allowing more time to make required repairs for inoperable reactor trip breakers consistent with allowed outage times for associated logic trains.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 205 to NPF-76
2. Amendment No. 193 to NPF-80
3. Safety Evaluation

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UNITED STATES
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STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-498

SOUTH TEXAS PROJECT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 205
License No. NPF-76

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)* acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated January 6, 2014, as supplemented by letters dated June 9, December 4, and December 17, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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;

   D. The issuance of this license 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.

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*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-76 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 205, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Facility Operating License No. NPF-76 and the Technical Specifications

Date of Issuance: April 29, 2015

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

STP NUCLEAR OPERATING COMPANY

DOCKET NO. 50-499

SOUTH TEXAS PROJECT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
License No. NPF-80

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

   A. The application for amendment by STP Nuclear Operating Company (STPNOC)* acting on behalf of itself and for NRG South Texas LP, the City Public Service Board of San Antonio (CPS), and the City of Austin, Texas (COA) (the licensees), dated January 6, 2014, as supplemented by letters dated June 9, December 4, and December 17, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, 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;

   D. The issuance of this license 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.

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*STPNOC is authorized to act for NRG South Texas LP, the City Public Service Board of San Antonio, and the City of Austin, Texas, and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-80 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 193, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Facility Operating
License No. NPF-80 and the
Technical Specifications

Date of Issuance: April 29, 2015

ATTACHMENT TO LICENSE AMENDMENT NOS. 205 AND 193

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

DOCKET NOS. 50-498 AND 50-499

Replace the following pages of the Facility Operating Licenses, Nos. NPF-76 and NPF-80, and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Operating License NPF-76

REMOVE	INSERT
-4-	-4-

Facility Operating License No. NPF-80

REMOVE	INSERT
-4-	-4-

Technical Specifications

REMOVE	INSERT
3/4 3-4	3/4 3-4
3/4 3-8	3/4 3-8
--	3/4 3-8a

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 205, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SER)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Safety Parameter Display System (Section 18, SSER No. 4)*

Before startup after the first refueling outage, HL&P[**] shall perform the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to issues as described in Section 18.2 of SER Supplement 4.

(6) Supplementary Containment Purge Isolation (Section 11.5, SSER No. 4)

HL&P shall provide, prior to startup from the first refueling outage, control room indication of the normal and supplemental containment purge sample line isolation valve position.

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

** The original licensee authorized to possess, use and operate the facility was HL&P. Consequently, historical references to certain obligations of HL&P remain in the license conditions.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 193 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. STPNOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Not Used

(4) Initial Startup Test Program (Section 14, SR)*

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) License Transfer

Texas Genco, LP shall provide decommissioning funding assurance, to be held in decommissioning trusts for South Texas Project, Unit 2 (Unit 2) upon the direct transfer of the Unit 2 license to Texas Genco, LP, in an amount equal to or greater than the balance in the Unit 2 decommissioning trust immediately prior to the transfer. In addition, Texas Genco, LP shall ensure that all contractual arrangements referred to in the application for approval of the transfer of the Unit 2 license to Texas Genco, LP to obtain necessary decommissioning funds for Unit 2 through a non-bypassable charge are executed and will be maintained until the decommissioning trusts are fully funded, or shall ensure that other mechanisms that provide equivalent assurance of decommissioning funding in accordance with the Commission's regulations are maintained.

(6) License Transfer

The master decommissioning trust agreement for Unit 2, at the time the direct transfer of Unit 2 to Texas Genco, LP is effected and thereafter, is subject to the following:

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Safety Injection Input from ESFAS	2	1	2	1, 2	9A
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
f. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
20. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10, 12A

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. For Functional Units with installed bypass test capability, the inoperable channel may be placed in bypass, and must be placed in the tripped condition within 72 hours.

   Note: A channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.1.1, provided no more than one channel is in bypass at any time.

b. For Functional Units with no installed bypass test capability,

   1. The inoperable channel is placed in the tripped condition within 72 hours, and

   2. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - (Not Used)

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 24 hours restore the inoperable channel to OPERABLE status, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9A - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 24 hours restore the inoperable channel to OPERABLE status, or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

b. With the number of OPERABLE channels more than one less then the Minimum Channels OPERABLE requirement, within 1 hour restore at least one inoperable channel to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status or initiate action to fully insert all rods within 48 hours, and place the rod control system in a condition incapable of rod withdrawal within the next hour.

ACTION 11 - (Not Used)

ACTION 12 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours, or be in at least HOT STANDBY within the next 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 12A - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, within 48 hours restore it to OPERABLE status or initiate action to fully insert all rods, and within the next hour place the rod control system in a condition incapable of rod withdrawal.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 205 AND 193 TO

FACILITY OPERATING LICENSE NOS. NPF-76 AND NPF-80

STP NUCLEAR OPERATING COMPANY, ET AL.

SOUTH TEXAS PROJECT, UNITS 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By application dated January 6, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14035A075), as supplemented by letters dated June 9, December 4, and December 17, 2014 (ADAMS Accession No. ML14184B363, ML14365A040, and ML15008A026, respectively), STP Nuclear Operating Company (the licensee), requested changes to the Technical Specifications (TSs) for South Texas Project (STP), Units 1 and 2. The supplements dated December 4 and 17, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on August 5, 2014 (79 FR 45481).

The proposed changes would revise TS 3.3.1, "Reactor Trip System Instrumentation," with respect to the required actions and allowed outage times (AOTs) for inoperable reactor trip breakers (RTBs). The operator actions are revised to reduce unnecessary shutdowns and increase operational flexibility by allowing more time to make required repairs for inoperable RTBs consistent with AOTs for associated logic trains.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The NRC's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

As stated in 10 CFR 50.36, the licensee's TSs must have surveillance requirements relating to test, calibration, or inspection to ensure that the facility maintains the necessary quality of

systems and components, facility operations are within safety limits, and facility equipment will meet the limiting conditions for operation. The surveillance requirements may include mode restrictions based on the safety aspects of conducting the surveillance in excluded modes.

In determining whether an amendment to a license will be issued to the applicant, the Commission is guided by the considerations which govern the issuance of initial licenses to the extent applicable and appropriate. In this instance, the NRC staff must be able to conclude that that the Reactor Trip System (RTS)/Engineered Safety Feature Actuation System (ESFAS) functions affected by these proposed TS changes will perform their required safety functions in accordance with the responses to design basis accidents described in Chapter 15 of the licensee's final safety analysis report.

The NRC staff also based its review upon the following General Design Criteria (GDC) of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50:

- GDC 20, "Protection system functions," requires the protection system be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- GDC 21, "Protection system reliability and testability," requires the protection system be designed for high functional reliability, testability, redundancy, and independence commensurate with the safety functions to be performed.

- GDC 22, "Protection system independence," requires the protection system be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function.

- GDC 23, "Protection system failure modes," requires the protection system be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced.

- GDC 29, "Protection against anticipated operational occurrences," requires the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The NRC staff reviewed the license amendment request (LAR) for consistency with NUREG-1431, Revision 4.0, "Standard Technical Specifications, Westinghouse Plants," issued April 2012 (ADAMS Accession No. ML12100A222) (NUREG-1431), and the Westinghouse Electric Company LLC topical report WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment

of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," issued March 2003 (proprietary)[1].

Additionally, the NRC staff used the acceptance guidelines specified in Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued May 2011 (ADAMS Accession No. ML100910006), and RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," issued May 2011 (ADAMS Accession No. ML100910008).

Also, the NRC staff used NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (ADAMS Accession No. ML071700658), Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load" (ADAMS Accession No. ML12193A107), and Section 16.1, "Risk-Informed Decision Making: Technical Specifications" (ADAMS Accession No. ML070380228).

3.0 TECHNICAL EVALUATION

Description of Reactor Trip Breakers

The licensee provided the following description of the RTBs in its LAR dated January 6, 2014:

> Two reactor trip breakers (RTB), arranged in series, connect three-phase ac [alternating current] power from the control rod drive motor generator sets to the rod drive power cabinets supplying power to the control rod drive mechanisms (CRDM). Opening either of the RTBs interrupts power to the CRDMs and allows the shutdown rods and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power.
>
> During normal operation the output from the solid state protection system (SSPS) provides a direct voltage signal to the undervoltage coil on each reactor trip breaker and bypass breakers, if in use. Direct current holds a trip plunger out against its spring, allowing ac power to be available at the rod drive power cabinets. SSPS consists of two logic trains, each capable of opening a separate and independent reactor trip breaker. SSPS takes binary inputs (voltage, or no-voltage) from the process and nuclear instrumentation channels

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[1] By letter dated March 19, 2003, from R. H. Bryan, Chairman, Westinghouse Owners Group (WOG), to the NRC, the WOG transmitted copies of WCAP-15376-P-A, Revision 1 (Proprietary), WCAP-15377-NP-A, Revision 1 (Non-Proprietary), each entitled "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times" and dated March 2003. The "-A" designates that WCAP is "accepted" by the NRC staff. The "accepted" version incorporates the staff's acceptance letter, safety evaluation, and all requests for additional information and responses concerning the Revision 0 versions of the WCAPs. The transmittal letter, WCAP-15376-P-A, WCAP-15377-NP-A, the associated NRC Safety Evaluation Report, and other related documents are available in ADAMS package No. ML030870767.

corresponding to the conditions of plant parameters. When a required logic combination is completed, a reactor trip signal (i.e. no voltage) is generated to the undervoltage trip coil. The reactor trip signal energizes the shunt trip auxiliary relay coils of the RTBs to trip the breakers open. The shunt trip auxiliary relay coils provide a diverse means to trip the RTBs.

The reactor trip system is designed to permit periodic testing during power operation without initiating a protective action unless a trip condition actually exists. Where only parts of the system are tested at any one time, the testing sequence provides the necessary overlap between the parts to assure complete system operation.

The licensee stated in Section 1.0, "Summary Description" of its application that there are no modifications to the setpoint actuations, trip setpoints, surveillance requirements or channel response that would affect the safety analyses associated with the proposed changes. The NRC staff has reviewed the amendment application and, because the only aspects of the RTS/ESFAS instrumentation being changed are the test bypass times and CTs, the NRC staff concludes that the RTS/ESFAS instrumentation involved with the proposed changes in the application continue to meet GDCs 20-23 and 29.

3.1 NRC Staff Review of TS 3.3.1, Table 3.3-1, Actions

3.1.1 Minimum Channels Operable, Modes 1 and 2

TS 3.3.1, Table 3.3-1, "Reactor Trip System Instrumentation," directs entry into Action 9 in the event that the Minimum Channels Operable column requirements are not met for Functional Unit 20, "Reactor Trip Breakers," when in Modes 1 and 2.

Current TS Table 3.3-1, Action 9 states:

> With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

Revised TS Table 3.3-1, Action 9 would state:

> With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 24 hours restore the inoperable channel to OPERABLE status, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

The proposed amendment modifies Action 9 to increase the AOT in the event that the Minimum Channels Operable column requirement is not met and increases the amount of time that one channel may be bypassed for surveillance testing provided the other channel is operable.

The NRC staff reviewed the proposed modification to Action 9 for consistency with NUREG-1431. The staff reviewed the NUREG-1431 that describes extending the RTB AOT for one inoperable breaker for up to 24 hours, and the bypass test time for an RTB for up to 4 hours for surveillance testing.

The NRC staff reviewed NUREG-1431, Section 3.3, INSTRUMENTATION, that describes extending the RTB AOT for one RTB inoperable for up to 24 hours, and extending the bypass test time for up to 4 hours. The LAR proposed AOT and bypass test times for one inoperable RTB are appropriate since they align with the NUREG-1431 timeframes. Additionally, the NRC staff reviewed NUREG-1431, Table 3.3.1-1, Function 21, "Automatic Trip Logic," and compared this to the LAR proposed modifications to Action 9, and determined that they were consistent.

Therefore, the NRC staff has determined that since the proposed STP AOT and bypass test times were consistent with NUREG-1431, and the NUREG-1431 surveillance intervals and out-of-service times were determined based on the SRP criteria of maintaining an appropriate level of reliability of the Reactor Protection System instrumentation, then the NRC staff concludes that the changes are acceptable.

3.1.2 Minimum Channels Operable, Modes 3, 4, and 5

When STP is in Modes 3, 4, and 5, the RTS breakers are in the closed position; and the control rod drive system is capable of rod withdrawal, STP's TS 3.3.1, Table 3.3-1, directs entry into Action 10 in the event that the Minimum Channels Operable column requirements are not met for Functional Unit 20, "Reactor Trip Breakers." Entry into Action 10 is also required for Functional Unit 1, "Manual Reactor Trip," and Functional Unit 21, "Automatic Trip and Interlock Logic," in the event that the Minimum Channels Operable column requirements are not met for these Functional Units (also for Modes 3, 4, and 5; the RTS breakers are in the closed position; and the control rod drive system is capable of rod withdrawal).

Current TS Table 3.3-1, Action 10 states:

> With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

Revised TS Table 3.3-1, Action 10 would state:

> With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status or initiate action to fully insert all rods within 48 hours, and place the rod control system in a condition incapable of rod withdrawal within the next hour.

The licensee indicated that the proposed amendment modifies Action 10 to be consistent with the NUREG-1431.

The NRC staff reviewed the proposed modification to Action 10 for consistency with the NUREG-1431. The staff reviewed the NUREG-1431, Table 3.3.1-1, Function 19, "Reactor Trip Breakers (RTBs)," which describes the action statement to restore the channel to operable

status within 48 hours or initiating action to fully insert all rods and place the control rod system in a condition incapable of rod withdrawal within 49 hours. The staff then compared the NUREG-1431 statement to the licensee's proposed modification. In comparing the statement, the staff noted that that licensee's proposed CT of 'within the next hour' is consistent with the NUREG-1431's 49-hour completion. The staff concludes that this is acceptable.

The NRC staff reviewed the licensee's current TS requirements located in TS 3.3.1, Table 3.3-1, and the corresponding table notation for Table 3.3-1, pertaining to entry into Action 10 in the event that the Minimum Channels Operable column requirements are not met for Functional Unit 20 when in Modes 3, 4, and 5, and the RTS breakers are in the closed position and the control rod drive system is capable of rod withdrawal. The staff also reviewed the licensee's proposed modification impact on Functional Unit 1, "Manual Reactor Trip," and Functional Unit 21, "Automatic Trip and Interlock Logic."

The NRC staff compared the TS requirements and determined that the wording was consistent with that found in the NUREG-1431. For Functions 1, 20, and 21, the proposed Action 10 continues to require the full insertion of the rods, and removing the capability to withdraw rods. Additionally, the staff's review of the LAR determined that taking action to fully insert the rods and rendering the rods incapable of rod movement is an equivalent change to the current STP TS 3.3.1 of opening the reactor trip system breakers. The staff's review noted that no changes are proposed to the time from the current TS Action 10 of opening the reactor trip system breakers to the proposed TS Action 10 of fully inserting all rods and rendering rods incapable of withdrawal.

Therefore, the NRC staff concludes that the proposed STP AOT and bypass test times and the Action 10 modification continues to prevent rod movement from the fully inserted position were consistent with NUREG-1431, and the NUREG-1431 surveillance intervals and out-of-service times were determined based on the SRP criteria of maintaining an appropriate level of reliability of the Reactor Protection System instrumentation. Therefore, the NRC staff concludes that the changes are acceptable.

3.1.3 Inoperable Diverse Trip Feature, Modes 1 and 2

STP's TS 3.3.1, Table 3.3-1, directs entry into Action 12 in the event one of the diverse trip features of an RTB is inoperable for Functional Unit 20, "Reactor Trip Breakers," when in Modes 1 and 2.

Current TS Table 3.3-1, Action 12 states:

> With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

Revised TS Table 3.3-1, Action 12 would state:

> With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours, or be in at least HOT STANDBY within the next 6 hours. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

The licensee indicated that the proposed amendment modifies Action 12 in order to be consistent with the requirement for an inoperable diverse trip feature.

The NRC staff reviewed the proposed modification to Action 12 for consistency with the NUREG-1431. The staff reviewed NUREG-1431, Table 3.3.1-1, Function 21, "Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms," which describes restoring the inoperable trip mechanism to operable status within 48 hours or be in Mode 3 (which is hot standby) within 54 hours.

The licensee's current Action 12 indicates that with one of the diverse trip features inoperable, restore it to operable status within 48 hours or declare the breaker inoperable, and apply Action 9. The current Action 9 statement is to '...be in at least hot standby within 6 hours.' Also, the breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to operable.

The replacement of '...apply Action 9' with '...be in at least hot standby within 6 hours' for the proposed Action 12 is equivalent and consistent with the NUREG-1431's 54-hour CT. Also, the NRC staff's review and comparison of the licensee's proposed Action 12 to the licensee's current Action 12 found that the requirements remain consistent with the TS requirement to be in hot standby within 6 hours.

Therefore, the NRC staff concludes that the proposed STP CT and bypass test times were consistent with NUREG-1431, and the NUREG-1431 surveillance intervals and out-of-service times were determined based on the SRP criteria of maintaining an appropriate level of reliability of the Reactor Protection System instrumentation. Therefore, the NRC staff concludes that the changes are acceptable.

3.1.4 Inoperable Diverse Trip Feature, Modes 3, 4, and 5

The licensee proposes to add new Action 12A to Functional Unit 20, "Reactor Trip Breakers," in TS 3.3.1, Table 3.3-1, which would address the condition where one diverse trip feature for an RTB is inoperable when the RTS breakers are in the closed position and the control rod drive system is capable of rod withdrawal in Modes 3, 4, and 5.

The licensee stated in its application that the STP TS do not contain a function specifying an action for when a diverse trip feature is inoperable.

New TS Table 3.3-1, Action 12A would state:

> With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, within 48 hours restore it to OPERABLE status or initiate action to fully insert all rods, and within the next hour place the rod control system in a condition incapable of rod withdrawal.

The licensee also stated that this change is consistent with NUREG-1431. Table 3.3.1-1, Function 20, "Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms," specifies that for Modes 3, 4, and 5 and with the rod control system capable of rod withdrawal or one or more rods not fully inserted, an inoperable diverse trip feature (undervoltage or shunt trip mechanism) would require entry into Condition C. Condition C requires restoration of the channel or train (in this case, the diverse trip feature) to operable status within 48 hours. If the diverse trip feature cannot be restored to operable status within the allowed 48-hour CT, the limiting condition for operation must be met. To achieve this status, actions must be initiated within the same 48 hours to insert all rods, and the rod control system must be placed in a condition incapable of rod withdrawal within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With all rods inserted and the rod control system incapable of rod withdrawal, Condition C is satisfied.

The NRC staff also reviewed proposed Action 12A for addressing the condition where one diverse trip feature for an RTB is inoperable when in Modes 3, 4, and 5, and the RTS breakers are in the closed position and the control rod drive system is capable of rod withdrawal. The staff also reviewed NUREG-1431, Table 3.3-1, Functional Unit 20, "Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms," and compared it to the LAR proposed Action 12A.

In a request for additional information (RAI) dated November 5, 2014 (ADAMS Accession No. ML14309A788), the NRC staff requested the following additional information from the licensee:

> Since New Action 12A is not in the licensee's current licensing basis (CLB), please explain why New Action 12A is needed and provide a technical evaluation for this change. In addition, please explain, given the CLB TS, how would the inoperability of a diverse trip feature impact the operability of a reactor trip breaker during the modes of applicability. Please include discussion of any TS actions entered, since this New Action 12A is currently not in STP's TSs.

In its RAI response dated December 4, 2014, the licensee stated that:

> New Action 12A is being proposed to clarify the actions to take to address an inoperable diverse trip feature (reactor trip breaker undervoltage mechanism or shunt trip mechanism) while operating in Modes 3, 4, and 5. New Action 12A would provide the same option for Modes 3, 4, and 5 that Action 12 provides for Modes 1 and 2.
>
> The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the Control Rod Drive Mechanisms (CRDMs). Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown

rods and control rods to fall into the core by gravity. During normal operation the output from the SSPS is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the SSPS output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is de-energized to trip the breaker open upon receipt of a reactor trip signal from the SSPS. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

With one of the diverse trip features inoperable, the reactor trip breakers would still open in the event that the other diverse trip feature is de-energized.

For Modes 1 and 2 in the current STP TSs, the inoperability of a diverse trip feature would require entering Action 12 to restore the feature to operable status within 48 hours or declare the breaker inoperable and enter Action 9 to be in Hot Standby within 6 hours.

For Modes 3, 4, and 5 in the current STP TSs when the Reactor Trip System breakers are in the closed position, the inoperability of a diverse trip feature would require entering Action 10 to restore the inoperable channel to operable status within 48 hours or open the RTBs within the next hour.

New Action 12A has the same result as the current Action 10. In new Action 12A, if the diverse trip feature is not returned to operable status within 48 hours, the rod control system is placed in a condition incapable of rod withdrawal within the next hour.

The NRC staff has reviewed the licensee's RAI response and the LAR. The NRC staff has determined that the proposed changes to TS 3.3.1, Table 3.3-1, new Action 12A are acceptable because if the diverse trip feature cannot be restored to operable status within the allowed 48-hour CT, actions must be initiated within the same 48 hours to fully insert all rods, and the rod control system must be placed in a condition incapable of rod withdrawal within the next hour. With rods fully inserted and the rod control system incapable of rod withdrawal, the Condition C is satisfied.

The NRC staff's review of the licensee's proposed Action 12A to the NUREG-1431 determines that the requirements remain consistent with the TS requirement to restore the channel to operable status within 48 hours or initiate action to fully insert all rods. Furthermore, there is a low probability of an event occurring during this interval, as discussed in Section 3.2 of this safety evaluation (SE), and the redundant diverse trip mechanism is operable and adequate to perform the safety function.

Therefore, the NRC staff concludes that the proposed STP CT was consistent with NUREG-1431, and the NUREG-1431 surveillance intervals and out-of-service times were

determined based on the SRP criteria of maintaining an appropriate level of reliability of the Reactor Protection System instrumentation. Therefore, the NRC staff concludes that the proposed changes are acceptable.

3.1.5 Conclusion

The NRC staff reviewed the proposed LAR and determined that the modification to Action 9, Action 10, Action 12, and the addition of Action 12A in TS 3.3.1 to be acceptable. The staff found the changes are consistent with NUREG-1431, in which the specific changes were based on NRC-approved WCAP-15376-P-A and the SRP Chapter 7.2 acceptance criteria. Further, the NRC staff has determined that in accordance with 10 CFR 50.36, there is reasonable assurance that the RTS/ESFAS functions affected by the proposed changes to TS 3.3.1 will continue to perform their required safety functions thereby protecting public health and safety.

3.2 Risk Evaluation

The proposed amendment adopts changes as described in WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," issued March 2003.

The following table (non-proprietary) summarizes the proposed WCAP-15376-P-A changes, as applicable to STP:

Reactor Protection System Component	Completion Times (hours)		Bypass Test Time (hours)	
	Current	Proposed	Current	Proposed
Logic Cabinets	No Change		No Change	
Master Relays				
Analog Channels				
Reactor Trip Breakers	N/A	24	2	4

The changes proposed by the licensee employ a risk-informed approach to justify changes to CTs and bypass test times. The risk metrics, change in Core Damage Frequency (ΔCDF), change in Large Early Release Frequency (ΔLERF), Incremental Conditional Core Damage Probability (ICCDP), and Incremental Conditional Large Early Release Probability (ICLERP), developed in WCAP-15376-P-A and that the licensee used to evaluate the impact of the proposed changes are consistent with those presented in RGs 1.174 and 1.177.

3.2.1 NRC Staff Review of Applicability of WCAP-15376-P-A to STP, Units 1 and 2

To show that WCAP-15736 is applicable to STP, Units 1 and 2, the licensee addressed the conditions and limitations of the NRC staff's SE dated December 20, 2002, and the implementation guidance developed by the Westinghouse Owner's Group that compares plant-specific data to the generic analysis assumptions. The evaluation compared the general baseline assumptions, including surveillance, maintenance, calibration, actuation signals, procedures, and operator actions, to confirm that the generic evaluation assumptions used in WCAP-15736 are also applicable to STP, Units 1 and 2.

The following paragraphs discuss the licensee's evaluation of the SE conditions and limitations of WCAP-15376-P-A to be applicable to STP, Units 1 and 2.

Condition 1: Confirm the applicability of the WCAP-15376-P-A analysis for its plant.

The licensee's submittal provided the evaluation for applicability of WCAP-15376-P-A. The evaluation included a comparison of parameters and assumptions with STP plant-specific data. The data comparison included logic cabinet types; component bypass test time; component test intervals; typical at-power maintenance intervals; plant procedures in place for selected operator actions; Anticipated Transient Without Scram (ATWS) Mitigating System Actuation Circuitry; ATWS contribution to core damage frequency (CDF); total internal events CDF; total transient event frequency; and total internal events Large Early Release Frequency (LERF) to plant-specific values. The licensee also evaluated containment failure modes, as requested by WCAP-15376-P-A, and determined that the LERF analysis is applicable to STP.

Based on the evaluation presented in Section 3.2.2.1 of this SE, the NRC staff considers that Condition 1 is satisfied for STP, Units 1 and 2.

Condition 2: Address the Tier 2 and Tier 3 analysis, including risk-significant configuration insights, and confirm that these insights are incorporated into the plant-specific Configuration Risk Management Program (CRMP).

Based on the evaluation presented in Sections 3.2.2.2 and 3.2.2.3 of this SE, the NRC staff considers that Condition 2 is satisfied for STP, Units 1 and 2.

Condition 3: Evaluate the risk impact of concurrent testing of one logic cabinet and associated RTB on a plant-specific basis to ensure conformance with the WCAP-15376-P-A evaluation, including the guidance of RGs 1.174 and 1.177.

The licensee showed that the generic analysis presented in WCAP-15376-P-A is applicable to STP, Units 1 and 2. WCAP-15376-P-A previously did not specifically evaluate or preclude concurrent testing of one logic cabinet and associated RTB. In response to an NRC staff question on WCAP-15376-P-A, the Pressurized-Water Reactor Owner's Group (PWROG) provided risk estimates for the logic train and associated RTB train out-of-service configuration. The resulting generic ICCDP estimate was within the acceptance guidelines of RG 1.177. Based on the applicability of WCAP-15376-P-A to STP, Units 1 and 2, and an ICCDP estimate within the acceptance guidelines of RG 1.177, the NRC staff considers that Condition 3 is satisfied.

Condition 4: Confirm that the model assumptions for human reliability in WCAP-15376-P-A are applicable to the plant-specific configuration.

The licensee confirmed that the assumptions regarding human reliability used in WCAP-15376-P-A is applicable to STP, Units 1 and 2. This review concluded that for the operator actions identified in WCAP-15376-P-A, plant procedures are available consistent with the assumptions in WCAP-15376-P-A. Based on the above, the NRC staff considers that Condition 4 is satisfied.

Condition 5: For future digital upgrades with increased scope, integration, and architectural differences beyond those of Eagle 21, the generic applicability to WCAP-15376-P-A should be considered on a plant-specific basis.

Because the licensee's proposed changes do not involve a digital upgrade, Condition 5 is not applicable to the implementation of WCAP-15376-P-A at STP, Units 1 and 2.

Condition 6: Review the plant-specific set point calculation methodology to ensure that the extended surveillance test intervals do not adversely impact the plant-specific set point calculations and assumptions for instrumentation associated with the extended surveillance test intervals.

The licensee stated in its LAR that the proposed change does not involve a change to surveillance frequencies. Based on the NRC staff's review of the LAR, the NRC staff determined that there are no changes to surveillance frequencies and considers that Condition 6 does not apply to this LAR.

3.2.2 WCAP-15376-P-A Tier 1, 2, and 3 Analysis to STP, Units 1 and 2

3.2.2.1 Tier 1: Probabilistic Risk Assessment Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk based on the STP, Units 1 and 2, implementation of WCAP-15376-P-A. The Tier 1 NRC staff review involves (1) evaluation of the validity of the probabilistic risk assessment (PRA) and its application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Technical Adequacy

The objective of the PRA technical adequacy review is to determine whether WCAP-15376-P-A, which is used in evaluating the proposed RTB CT and test bypass time, is of sufficient scope and detail for this application. WCAP-15376-P-A provided a generic PRA model for the evaluation of the CT and test bypass time. The NRC staff found this generic model and the WCAP-15376-P-A evaluation to be acceptable on a generic basis in the SE dated December 20, 2002. Although the SE accepted the use of a representative model as generally reasonable, the application of the representative model and the associated results to a specific plant introduces a degree of uncertainty because of modeling, design, and operational differences. Therefore, each licensee adopting WCAP-15376-P-A will need to confirm that the analysis and results are applicable to its plant.

The NRC staff reviewed the information provided in the proposed license amendment and the findings and conditions of the staff's adoption of WCAP-15376-P-A via its SE. The WCAP-15376-P-A conditions and limitations identified by the staff were considered limiting for STP, in that the WCAP-15376-P-A does not specify the use of the STP PRA or plant-specific estimates of ΔCDF, ΔLERF, ICCDP, or ICLERP in the implementation of the WCAP. However, in its SE for WCAP-15376-P-A, the staff found that the applicability of the generic PRA analysis for the proposed CT, bypass test time, and surveillance test intervals changes to other Westinghouse plants may not be representative based on design variations in actuated systems

and the contribution to plant risk from accident classes impacted by the proposed change. The staff, therefore, concluded that each licensee would need to address any differences between its plant and the representative plant that could increase the CT, bypass test time, or surveillance test interval risk significance.

The licensee reviewed the scope and detail of the STP PRA using the WCAP-15376-P-A PRA parameters to demonstrate the plant-specific applicability of the proposed CT and bypass test times. The licensee compared logic cabinet types; component bypass test time; component test interval; typical at-power maintenance intervals; plant procedures in place for selected operator actions; ATWS Mitigating System Actuation Circuitry; ATWS contribution to CDF; total internal events CDF; total transient event frequency; and total internal events LERF to plant-specific values.

In an RAI dated November 5, 2014, the NRC staff requested that the licensee explain how it considered uncertainty bounds in the data consistent with RG 1.174 and RG 1.177 to ensure the conclusions remained valid for the plant-specific case. In its RAI response dated December 4, 2014, the licensee stated that the ATWS contribution to CDF of 4.3E-08 per year is small compared to the acceptance guideline of 1E-06 per year in RG 1.174. This is consistent with RG 1.174 and RG 1.177 as a detailed uncertainty analysis would not change the acceptance criteria.

Based on the comparison in accordance with the implementation guidelines for WCAP-15376-P-A, the NRC staff concludes that WCAP-15376-P-A is applicable to STP, Units 1 and 2.

Peer Review

In April 2002, the STP PRA underwent an industry peer review performed in accordance with the Nuclear Energy Institute (NEI) 00-02, "Industry PRA Peer Review Process." As a part of the NRC staff review of Risk-Managed Technical Specifications (RMTS) for STP, Units 1 and 2,[2] the licensee submitted its assessment of the STP PRA against each of the supporting requirements of American Society of Mechanical Engineers (ASME) Standard RA-S-2002 for its internal events PRA model. Where the standard provides separate requirements for capability categories, the licensee based its assessment on category II, consistent with the guidance of NEI 06-09, "Risk-Managed Technical Specifications (RMTS) Guidelines." The licensee did not identify any exceptions to the standard. Based on the licensee's assessment and the NRC staff reviews, the staff determined that the STP PRA internal events model satisfied the guidance of Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," February 2004 (ADAMS Accession No. ML040630078), and conformed to capability category II of the ASME standard for the supporting requirements.

In RG 1.200, Revision 2, issued March 2009 (ADAMS Accession No. ML090410014), the NRC staff endorsed ASME/American Nuclear Society (ANS)-RA-Sa-2009, "Standard for Level 1 /

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[2] Thadani, M. C., U.S. Nuclear Regulatory Commission, letter to James J. Sheppard, STP Nuclear Operating Company, "South Texas Project, Units 1 and 2 - Issuance of Amendments Re: Broad-Scope Risk-Informed Technical Specifications Amendments (TAC Nos. MD2341 and MD2342," dated July 13, 2007 (ADAMS Accession No. ML071780186).

Large Early Release Frequency [LERF] Probabilistic Risk Assessment [PRA] for Nuclear Power Plant Applications." The NRC staff concluded that "if the results of this self-assessment are used to demonstrate the technical adequacy of a PRA for an application, differences between the current version of the standard as endorsed in Appendix A and the earlier version be identified and addressed." This analysis would result in the identification of "gaps" between the 2005 ASME standard and the 2009 standard for internal events.

The NRC staff recognizes that there were only minor clarifications between the supporting requirements for the two standards, as they relate to the topic of this application. Therefore, the staff sent an RAI dated November 5, 2014, in which it requested clarification from the licensee on the changes between Revision 5 of the PRA model, referenced above and in the approved RMTS program, and Revision 7.2 of the PRA model that was referenced in the submittal. The NRC staff also requested that the licensee include any applicable Facts and Observations to the RTB TS changes. The licensee's response dated December 4, 2014, confirmed that the upgrades and updates to the PRA model have received the applicable review and none of those changes affected the RTB TS change and there are no open Facts and Observations related to the TS change. Therefore, the STP internal events PRA was determined, by NRC staff, to be of sufficient technical adequacy to support the specific aspects of this application based on the prior peer review in accordance with RG 1.200, Revision 1.

PRA Results and Insights

Cumulative Risk

WCAP-15376-P-A presented comparisons to a base-case risk analysis, which represents the changes previously approved under WCAP-14333. For STP's case, the cumulative impact on the CDF for 2-out-of-4 logic was within the RG 1.174 acceptance guidelines of less than 1E-6 per year, representing a very small change. The cumulative impact on CDF for 2-out-of-3 logic was within the RG 1.174 acceptance guideline for a very small change. For STP, the cumulative risk is limited from the Technical Specifications Optimization Program (TOP) condition to WCAP-15376-P-A implementation. Since the proposed change for STP is from TOP to WCAP-15376-P-A, the expected change in cumulative risk is expected to be less than the WCAP-15376-P-A estimates and is acceptable.

External Events

In an RAI dated November 5, 2014, the NRC staff requested that the licensee explain the contribution of external event risk to the reactor trip breaker TS change. In its response dated December 4, 2014, the licensee evaluated the proposed reactor protection system CT and test bypass time for their potential impact on external events, including fires; seismic events; high winds, and flooding events. The following table (non-proprietary) presents the external event risk contribution to the proposed TS change.

External Events Risk Contributions			
INITIATOR	IE Frequency	CDF	%CDF
Tornado Induced Failure of Switchyard and Essential Cooling Pond	1.22E-06	1.11E-06	18.31
Fire Zone 047 Scenario X	1.46E-05	3.65E-07	6.02
Switchyard and Essential Cooling Pond Failure Due to Breach of Main Cooling Reservoir	3.20E-07	2.91E-07	4.80
Fire Zone 071 Scenario X	2.34E-07	2.13E-07	3.51
Fire Zone 047 Scenario B	2.72E-03	2.09E-07	3.45
Control Room Fire Scenario 18	2.12E-06	9.12E-08	1.50
Fire Zone 047 Scenario BC	3.18E-06	5.91E-08	0.98
Seismic Event, 0.4g Acceleration	7.74E-07	4.04E-08	0.67
Control Room Fire Scenario 23	1.61E-06	2.62E-08	0.43
Seismic Event, 0.6g Acceleration	6.14E-08	2.08E-08	0.34
Fire Zone 147 Scenario	1.08E-03	1.19E-08	0.20
External Flooding Scenarios 2 Through 6	1.05E-08	9.49E-09	0.16
Seismic Event, 0.2g Acceleration	2.89E-06	9.35E-09	0.15
Seismic Event, 0.1g Acceleration	3.02E-05	1.73E-09	0.03
Control Room Fire Scenario 10	3.43E-06	1.04E-09	0.02
Flood Induced LOOP - Scenario 1	3.20E-06	5.41E-10	0.01
Group Subtotal	**3.87E-03**	**2.46E-06**	**40.58**

IE = initiating event g = gravity LOOP = loss of offsite power

The licensee considers both SSPS channels failed for the control room fire scenario 18 initiator and all seismic initiators. The licensee also stated in the RAI response that "operator action to manually trip the reactor is credited in all external events."

Total Risk Contribution

The NRC staff considered whether the estimated fire and seismic risk, in conjunction with the internal event risk, could exceed the RG 1.174 total baseline CDF of 1E-4 per year with the implementation of WCAP-15376-P-A. The licensee estimated combined total CDF is estimated to be about 6.06E-6 per year (3.6E-6 per year + 2.46E-6 = [Total CDF for Internal Events + External Event CDF Contribution Group Subtotal]) for STP. RG 1.174 states that while there is no requirement to calculate the total CDF, if there is an indication that the CDF may be considerably higher than 1E-4 per year, the focus should be on finding ways to decrease rather than increase risk. Given the conservative nature of the licensee's estimate of the fire, seismic, high winds, and flooding risk, the NRC staff concludes that the total CDF is not expected to be higher than 1E-4 per year and is, therefore, acceptable for this application.

3.2.2.2 Tier 2—Avoidance of Risk-Significant Plant Configurations

A licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change.

Based on WCAP-15376-P-A and licensee evaluations, the licensee identified the following Tier 2 conditions as a regulatory commitment (see also Section 3.3 of this document):

- Procedures will be revised or created to ensure activities that degrade the availability of reactor coolant system pressure relief, auxiliary feedwater flow, ATWS Mitigating System Actuation Circuitry (AMSAC), and turbine trip will not be scheduled when a reactor trip breaker is out-of-service.

- Procedures will be revised or created to ensure activities that could degrade other components of the reactor protection system including master relays, slave relays, and analog channels will not be scheduled concurrently with a logic cabinet out of service.

- Procedures will be revised or created to ensure activities on electrical systems (e.g., alternating current and direct current power) that support the functionality of the following systems or components will not be scheduled when a reactor trip breaker is unavailable:

    - Reactor coolant pressure relief,
    - Auxiliary feedwater,
    - AMSAC,
    - Turbine trip protective circuitry, and
    - Reactor protection system including master relays, slave relays, and analog channels.

The NRC staff concludes that reasonable controls for the implementation and subsequent evaluation of the proposed changes pertaining to the above commitments are best provided by the licensee's administrative processes, including its commitment management program. The above commitments do not warrant the creation of regulatory requirements (i.e., items requiring prior NRC approval of subsequent changes) and are not the basis for NRC approval of this license amendment.

The licensee evaluated concurrent component outage configurations and common cause failures to confirm the applicability of the Tier 2 restrictions. Based on the above and the discussion in Condition 3 of this SE, the NRC staff concludes that the licensee's Tier 2 analysis supports the implementation of WCAP-15376-P-A and satisfies the conditions of the staff's SE acceptance of WCAP-15376-P-A regarding Tier 2.

3.2.2.3 Tier 3—Risk-Informed Configuration Risk Management Program

Risk assessment of online configurations for both STP units uses the latest updated version of the STP PRA model. The licensee has implemented an on-line maintenance tracking and control process that requires an integrated review to identify risk-significant plant configurations prior to both planned maintenance activities and emergent conditions during plant operations. This SE also considers the licensee's implementation and monitoring strategies in the following section "Implementation and Monitoring Program."

The NRC staff concludes that the licensee's program to control risk is capable of adequately assessing the activities being performed to ensure that high-risk plant configurations do not occur and compensatory actions are implemented if a high-risk plant configuration or condition should occur. As such, the licensee's program provides for the assessment and management of increased risk during maintenance activities as required by the Maintenance Rule 10 CFR 50.65(a)(4) and satisfies the RG 1.177 guidelines for a CRMP for the proposed change.

Implementation and Monitoring Program

RGs 1.174 and 1.177 also establish the need for an implementation and monitoring program to ensure that extensions to TS CTs or bypass test times do not degrade operational safety over time and that no adverse effects occur from unanticipated degradation or common-cause mechanisms. The purpose of an implementation and monitoring program is to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of systems, structures, and components impacted by the change. In addition, the application of the three-tiered approach in evaluating the proposed CT and bypass test times provides additional assurance that the changes will not significantly impact the key principle of defense in depth. In an RAI dated November 5, 2014, the NRC staff requested that the licensee explain how it planned to monitor reliability and availability of the RTS instrumentation. In its RAI response dated December 4, 2014, the licensee stated that it will perform the following surveillance procedures for the RTS instrumentation:

> Trip Actuating Device Operational Test (TADOT) surveillances for each train of RTS instrumentation including testing of the reactor trip bypass breaker, turbine trip relay testing, and automatic shunt trip testing for each train are performed at least once every 9 months for both RTS trains
>
> Response time testing and reactor trip breaker gripper release surveillances for both RTS trains, including automatic undervoltage trip testing and gripper release testing is performed at least once per 18 months

The following Preventative Maintenance (PM) activities are performed for the RTS instrumentation:

> Contingency maintenance support for the reactor trip breaker TADOT surveillance is performed as needed
>
> Thermography inspections to verify the integrity of cabinets, wiring, fasteners, and electrical connections on components associated with the RTS is performed every 26 weeks
>
> Inspection and testing of each RTB and reactor trip bypass breaker is performed every refueling outage
>
> Lubrication and overhaul of each RTB and reactor trip bypass breaker is performed every 9 years

The licensee monitors the reliability and availability of the RTS instrumentation under the Maintenance Rule 10 CFR 50.65(a)(1), which requires a licensee to monitor the performance or condition of systems, structures, and components against licensee-established goals. Based on the above, the licensee satisfies the RG 1.174 and RG 1.177 guidelines for an implementation and monitoring program for the proposed change.

3.2.3 Comparison with Regulatory Guidance

The proposed changes conform to the analysis performed in WCAP-15376-P-A, as accepted by the NRC staff, including limitations and conditions identified in the NRC staff SE. As such, the NRC staff has determined that the STP implementation of the WCAP-15376-P-A conforms to the RG 1.174 and RG 1.177 acceptance guidance for ΔCDF, ΔLERF, ICCDP, and ICLERP.

3.2.4 Conclusion

The NRC staff concludes that the licensee has demonstrated the applicability of the plant-specific aspects of WCAP-15376-P-A to STP, Units 1 and 2, and has met the limitations and conditions as outlined in the staff's SE, which adopted WCAP-15376-P-A specific to TS 3.3.1, "Reactor Trip System Instrumentation." The staff also concludes that the risk impacts for ΔCDF, ΔLERF, ICCDP, and ICLERP, as estimated by WCAP-15376-P-A to be applicable to STP, Units 1 and 2, and within the acceptance guidelines of RG 1.174 and RG 1.177. The licensee showed the applicability of the specified functional units to the WCAP-15376-P-A evaluations and results. The licensee's Tier 2 analysis evaluated concurrent outage configurations and confirmed the applicability of the risk-significant configurations identified by the staff's SE limitations and conditions and topical report analysis to ensure control of these configurations. The licensee's Tier 3 CRMP was found to be consistent with the RG 1.177 CRMP guidelines and the Maintenance Rule 10 CFR 50.65(a)(4)) for the implementation of WCAP-15376-P-A. The licensee monitors the reliability and availability of the RTS under the Maintenance Rule 10 CFR 50.65(a)(1). Therefore, the staff concludes that the licensee's proposed TS revisions are consistent with the CTs, bypass test times, and surveillance test intervals consistent with WCAP-15376-P-A, and meet the staff's SE conditions and limitations for WCAP-15376-P-A applicability. Since the WCAP-15376-P-A and associated NRC staff SE were based on the SRP criteria of maintaining an appropriate level of reliability of the Reactor Protection System instrumentation, the NRC staff concludes that the proposed changes are acceptable.

3.3 Commitments

In its application, the licensee provided the following regulatory commitments as administrative controls that the licensee will put in place to meet the conditions in the SE that approved WCAP-15376-P-A. The licensee stated in its application that these commitments will be put in place during the implementation of the amendment after its approval.

	Commitment	**Scheduled Completion**
1	Procedures will be revised or created to ensure activities that degrade the availability of reactor coolant system pressure relief, auxiliary feedwater flow, ATWS Mitigating System Actuation Circuitry (AMSAC), and turbine trip will not be scheduled when a reactor trip breaker is out-of-service.	Upon amendment implementation
2	Procedures will be revised or created to ensure activities that could degrade other components of the reactor protection system including master relays, slave relays, and analog channels will not be scheduled concurrently with a logic cabinet out of service.	Upon amendment implementation
3	Procedures will be revised or created to ensure activities on electrical systems (e.g. AC and DC power) that support the functionality of the following systems or components will not be scheduled when a reactor trip breaker is unavailable: • Reactor coolant pressure relief, • Auxiliary feedwater, • AMSAC, • Turbine trip protective circuitry, and • Reactor protection system including master relays, slave relays, and analog channels.	Upon amendment implementation

The NRC staff SE that approved WCAP-15376-P-A discussed restrictions on equipment removal when an RTB is out of service. The staff compared the commitments provided in the LAR to the requested administrative restrictions as found in the staff SE that approved WCAP-15376-P-A. As discussed in the Tier 2 analysis *Avoidance of Risk-Significant Plant Configurations* in Section 3.2.2.2 of this SE, the staff determined that the licensee's commitments were consistent with the NRC staff's WCAP-15376-P-A SE. The licensee's administrative processes, including its commitment management program, provide reasonable controls for the licensee's implementation, and subsequent evaluation of any changes to the above regulatory commitments. The NRC staff may choose to verify the implementation of and maintenance of these commitments in a future audit. Based on the staff comparison of the licensee's commitments to the NRC staff's SE approving the WCAP-15376-P-A restrictions, and the discussion in Section 3.2.2.2 of this SE, the NRC staff concludes that the licensee's regulatory commitments are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the 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 published in the *Federal Register* on August 5, 2014 (79 FR 45481). 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.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Deirdre Spaulding-Yeoman, NRR/DE/EICB
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Date: April 29, 2015

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Lisa M. Regner, Senior Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498 and 50-499

Enclosures:
1. Amendment No. 205 to NPF-76
2. Amendment No. 193 to NPF-80
3. Safety Evaluation

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