NRR-PMDAPEm Resource

From: Regner, Lisa
Sent: Tuesday, March 31, 2015 3:43 PM
To: Sterling, Lance (lsterling@STPEGS.COM)
Cc: Goetz, Sujata
Subject: Requests: EAL question and Proprietary verification
Attachments: draft RAI 10a sppl - lmr.docx; DRAFT STP TS 3.3.1 SE.docx

Lance, 2 things:

1. EAL Question: the staff needs a justification for question 10 in your latest response (2/26) to the EAL scheme change. Attached is a draft follow-up question for discussion. We'd like to have a call **Thursday, 4/2, between 10 and noon**, if possible. I understand you're all busy with the outage and your FP experts may be on outage duty. Let me know when we can have a call Thursday or sometime in the next week.

2. Proprietary verification: also attached is the draft SE for the TS 3.3.1 RTB AOT changes. Please verify there is no proprietary information from the proprietary WCAP in the SE; sometime this week would be good, but – again – I understand outage restrictions on resources.

Thanks,
Lisa

Hearing Identifier: NRR_PMDA
Email Number: 1981

Mail Envelope Properties (Lisa.Regner@nrc.gov20150331154300)

Subject: Requests: EAL question and Proprietary verification
Sent Date: 3/31/2015 3:43:16 PM
Received Date: 3/31/2015 3:43:00 PM
From: Regner, Lisa

Created By: Lisa.Regner@nrc.gov

Recipients:
"Goetz, Sujata" <Sujata.Goetz@nrc.gov>
Tracking Status: None
"Sterling, Lance (lsterling@STPEGS.COM)" <lsterling@STPEGS.COM>
Tracking Status: None

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Files	**Size**	**Date & Time**
MESSAGE	772	3/31/2015 3:43:00 PM
draft RAI 10a sppl - lmr.docx	39475	
DRAFT STP TS 3.3.1 SE.docx	97086	

Options
Priority: Standard
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DRAFT
REQUEST FOR ADDITIONAL INFORMATION
SOUTH TEXAS PROJECT NUCLEAR OPERATING COMPANY
SOUTH TEXAS PROJECT, UNITS 1 AND 2
LICENSE AMENDMENT REQUEST FOR
EMERGENCY ACTION LEVEL SCHEMECHANGE
DOCKET NUMBERS 50-498 AND 499
(TAC NOS.MF4195 AND MF4196)

By application dated May 15, 2014 (Reference 1), and supplemented by letters dated February 11, 2015, and February 26, 2015 (References 2 and 3, respectively), South Texas Project Nuclear Operating Company (STPNOC), requested prior U.S. Nuclear Regulatory Commission (NRC) approval for proposed changes to the emergency action level (EAL) scheme for the South Texas Project (STP), Units 1 and 2.

The licensee's requested changes to support a conversion from the current EAL scheme to a scheme based on Nuclear Energy Institute (NEI) 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 2012 (Reference 4).

The following request for additional information (RAI) is necessary for the NRC staff to complete its review.

General discussion

STP's response to RAI-10 (Reference 3), Attachment 1, Page 4 of 5 states:

> **STPNOC** has removed containment radiation monitors (RT-8050 and RT-8051) from the RCS [Reactor Coolant System]Barrier Category 3, RCS Activity/Containment Radiation table due to a reduction in the setpointvalue based on calculation STPNOC013-CALC-004, Revision 2. CalculationSTPNOC013-CALC-004 was revised in February 2015 and would have lowered the RT-8050and RT-8051 setpoint from 450 mR/hr to approximately 140 mR/hr above background.STPNOC believes that the proximity of the new setpoint to the background level and the effectof TIC [temperature induced current] precludes the use of these radiation monitors as reliable indications of an RCS breach.STPNOC does not have other Reg. Guide 1.97 radiation monitors in the containment that canfulfill the function of RT-8050 and RT-8051.

The NEI 99-01, Revision 6 (Reference 4), guidance associated with RCS barrier loss (based on section "RCS Activity / Containment Radiation" in Table 9-F-3: *PWR EAL Fission Product Barrier Table*) on page 98 states the following threshold for loss or potential loss of barriersis needed:

> Containment radiation monitor reading greater than (site-specific value)

The loss of RCS barrier based on RCS Activity / Containment Radiation is included to provide a diverse indication that the RCS barrier has been lost. The staffexpects licensees to provide site-specific indications that will promote timely and accurate assessment of barrier status.

RAI-10.a.

STPNOC proposesremoving the Loss of RCS Barrier due to Category 3, RCS Activity / Containment Radiation from the EAL scheme.

Typically, the value associated with this threshold is based upon the maximum allowed amount of fuel damage. Radiation levels above this threshold would therefore be indicative of a loss of the RCS barrier and would require an EAL as determined from the Fission Barrier logic. It is unusual for this value to be as low as described in STP's response, as compared to licensees of a similar design, which, in some instances, can be a threshold as high as 25 R/hr.

a. Provide justification that the calculated value is correct.If a discrepancy is identified,provide a correct value.
b. Provide a Containment Radiation value that would provide an indication of RCS barrier integrity.

REFERENCES

1. Letter from STPNOC to U.S. Nuclear Regulatory Commission – "South Texas Project, Revision to Unit 1 and Unit 2 Emergency Action Levels," dated May 15, 2014 (ADAMS Accession Number ML14164A341 [package]). (correct)

2. Letter from STPNOC U.S. Nuclear Regulatory Commission – "South Texas Project, Units 1 and 2 - Response to Request for Additional Information Regarding License Amendment Request for Emergency Action Level Scheme Change," dated February 11, 2015 (ADAMS Accession No. ML15055A039). (correct)

3. Letter from STPNOC U.S. Nuclear Regulatory Commission – "South Texas Project, Units 1 and 2 - Response to Request for Additional Information Regarding License Amendment Request for Emergency Action Level Scheme Change," dated February 26, 2015 (ADAMS Accession No. ML15068A045). (correct)

4. NEI 99-01, Revision 6, "Development of Emergency Action Levels for Non-Passive Reactors," dated November 21, 2012 (ADAMS Accession No. ML12326A805).(correct)

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 TSswith 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 arerevised 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

cc w/encls: Distribution via Listserv

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
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3. Safety Evaluation

cc w/encls: Distribution via Listserv

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JEvans, NRR/DRA/APLA
DSpaulding-Yeoman, NRR/DE/EICB
CTilton, NRR/DSS/STSB

ADAMS Accession No. ML15075A146 * per SE memo **via email

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA
NAME	JKlos**	LRegner	JBurkhardt
DATE	3/23/15	3/23/15	3/23/15
OFFICE	NRR/DSS/STSB/BC	NRR/DE/EICB/BC	NRR/DRA/APLA/BC
NAME	RElliott*	JThorp*	HHamzehee*
DATE	1/7/15	12/19/14	3/16/15
OFFICE	OGC	NRR/DORL/LPL4-1/BC	NRR/DORL/LPL4-1/PM
NAME		MMarkley	LRegner
DATE			

OFFICIAL RECORD COPY

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) <u>Technical Specifications</u>

   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 90days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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-76and the
Technical Specifications

Date of Issuance:

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.

---

*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 within90days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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-80and the Technical Specifications

Date of Issuance:

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 License 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 LicenseNo. 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

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 operatinglicenses to include TSs as part of the license. The NRC’s regulatoryrequirements related to the content of the TSs are contained in Title 10 of the*Code of FederalRegulations* (10 CFR), Section 50.36, “Technical specifications.”The TS requirementsin 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systemssettings and control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, (5) administrative controls, (6) decommissioning, (7) initialnotification, and (8) written reports.

As stated in 10 CFR 50.36, the licensee’s TSs must have surveillance requirementsrelating 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.

Although 10 CFR 50.36 does not list specific TS requirements, implicit within this rule are the requirementsthat action be taken when a limiting condition for operationis not being met and that the surveillance requirements, bypass test times, and completion times(CTs) specified in the TSs be based on reasonable protection of the public health and safety. Therefore, the NRC staff must be able to conclude that there is reasonable assurance 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 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 13, "Instrumentation and control," which states that the licensee shall provide appropriate controls to maintain these variables and systems within prescribed operating ranges.
- 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 theprotection 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 protectionsystem 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 resultin loss of the protection function.
- GDC 23, "Protection system failure modes," requires the protectionsystem be designed to fail into a safe state or into a state demonstrated to be acceptable onsome 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 extremelyhigh probability of accomplishing their safety functions in the event of anticipated operationaloccurrences.

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 reportWCAP-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 2013 (proprietary).

Additionally, the NRC staff used the acceptance guidelines specified in RegulatoryGuide (RG) 1.174, Revision 2, “An Approach for Using Probabilistic Risk Assessment in Risk-InformedDecisions 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 NuclearPower Plants” (SRP) Section 19.2, “Review of Risk Information Usedto 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 Results for Risk-Informed Activities” (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 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….

3.1 NRC Staff Review of TS3.3.1, Table 3.3-1, Actions

3.1.1 TS 3.3.1, Table 3.3-1, Action 9

TS3.3.1, 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 RTBAOT for one inoperable breaker for up to 24 hours, and the bypass test time for an RTB for up to 4 hours.

The NRC staff determined that the license amendment proposed AOT and bypass test time for the RTBs are appropriate since the times take into account the operability status of the redundant RTBs, the capability of the remaining reactor trip features to provide protection,the time needed for repairs or replacement, and the probability of a design-basis accident occurring during the repair period.

Additionally, the NRC staff reviewed NUREG-1431, Table 3.3-1, Functional Unit 21, “Automatic Trip Logic,” and compared this to the LAR proposed modifications to Action 9, and determined that they were consistent. Therefore, based on the staff’s review of the proposed AOT and bypass trip time and its comparison with NUREG-1431as consistent, the staff concludes the modifications are acceptable.

3.1.2 TS 3.3.1, Table 3.3-1, Action 10

TS3.3.1, 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,” 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. 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 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.

Current TS Table 3.3-1, Action 10 states:

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

The licensee indicated that the proposed amendment modifiesAction 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 entry into an action that requires restoring 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 requirements to the licensee’s proposed modification. In comparing the requirements, the staff noted that that licensee’s proposed CT of 1 hour is more conservative than 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 no changesare proposed to the time to render rods incapable of withdrawal.

Based on the NRC staff's review and evaluation, the staff concludes that the proposed modification to Action 10 is acceptable.

3.1.3 TS 3.3.1, Table 3.3-1, Action 12

TS3.3.1, Table 3.3-1, directs entry into Action 12 in the event one of the diverse trip features of an RTB is inoperable forFunctional 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 12in 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, Functional Unit 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 staff then compared this to the LAR proposed modifications to Action 12.

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 be in at least hot standby within 6 hours (which is the Action 9 requirement that currently Action 12 points to). 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 proposed 6 hours for the proposed Action 12 is more conservative than NUREG-1431's 54-hour completion. 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.

Based on the NRC staff's review and evaluation, the staff concludes that the proposed modification to Action 12 is acceptable.

3.1.4 TS 3.3.1, Table 3.3-1, New Action 12A

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.

New TS Table 3.3-1, Action 12 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 stated in its application that this change is consistent withNUREG-1431.Table 3.3.1-1, Functional Unit 20, "Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms," specifies that for Modes 3, 4, and 5 and with therod 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 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 completion time, the unit must be placed in a Mode inwhich the requirement does not apply. To achieve this status, actions must be initiated within the same 48 hours to ensure that all rods are fully inserted, and the rod control system must be placed in a conditionincapable of rod withdrawal within the next hour. The additional hourprovides sufficient time to accomplish the action in an orderly manner.With rods fully inserted and the rod control system incapable of rodwithdrawal, these functions are no longer required.

The NRC staff also reviewed proposed Action 12A for addressing the condition where one diverse trip feature for an RTBisinoperable when in Modes 3, 4, and 5, and the RTS breakers are in the closed position andthe 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 impactthe operability of a reactor trip breaker during the modes of applicability. Please includediscussion of any TS actions entered, since this New Action 12A is currently not in STP's TSs.

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

> New Action 12A is being proposed to clarify the actions to take to address an inoperablediverse trip feature (reactor trip breaker undervoltage mechanism or shunt tripmechanism) while operating in Modes 3, 4, and 5. New Action 12A would provide thesame option for Modes 3, 4, and 5 that Action 12 provides for Modes 1 and 2.
>
> The RTBsare in the electrical power supply line from the control rod drive motorgenerator set power supply to the Control Rod Drive Mechanisms (CRDMs). Opening ofthe RTBs interrupts power to the CRDMs, which allows the shutdown rods and controlrods 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 outputvoltage signal is removed, the undervoltage coils are de-energized, the breaker trip leveris actuated by the de-energized undervoltage coil and the RTBs and bypass breakersare tripped open. This allows the shutdown rods and control rods to fall into the core. Inaddition to the de-energization of the undervoltage coils, each breaker is also equippedwith a shunt trip device that is de-energized to trip the breaker open upon receipt of areactor trip signal from the SSPS. Either the undervoltage coil or the shunt tripmechanism is sufficient by itself, thus providing a diverse trip mechanism.
>
> With one of the diverse trip features inoperable, the reactor trip breakers would still openin 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 featurewould require entering Action 12 to restore the feature to operable status within 48 hoursor 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 breakersare in the closed position, the inoperability of a diverse trip feature would requireentering Action 10 to restore the inoperable channel to operable status within 48 hoursor open the RTBs within the next hour.
>
> New Action 12A has the same result as the current Action 10. In new Action 12A, if thediverse trip feature is not returned to operable status within 48 hours, the rod controlsystem 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 ensure that all rods are fully inserted, 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, these functions are no longer required.

Therefore, the NRC staff's review of the licensee's proposed Action 12A to the NUREG-1431concludes 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, given the low probability of an event occurring during this interval and considering that the other diverse trip mechanism is operable and adequate to perform the safety function, the NRC staff concludes that the proposed changes areacceptable.

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 thechanges are consistent with NUREG-1431, in which the specificchanges were based on NRC-approved WCAP-15376-P-A. The NRC staff has determinedthe proposed changes to TS 3.3.1 continue to provide reasonable assurance that therequirements are met for GDCs 20, 21, 22, 23, and 29 of Appendix A to 10 CFR 50, and,therefore, are acceptable.

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, as approved by NRC letter dated December 20, 2002(ADAMS Accession No. ML023540534) (referred to as WCAP-15376).

The following table summarizes the proposed WCAP-15376 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 CTsand 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 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 ofApplicability of WCAP-15376 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 December 20, 2002, safety evaluation (SE) 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 to be applicable to STP, Units 1 and 2.

Condition 1:

A licensee should confirm the applicability of the WCAP-15376 analysis for its plant.

The licensee's submittal provided the evaluation for applicability of WCAP-15376. 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, 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:

Under WCAP-15376, the licensee should also 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:

The licensee should evaluate the risk impact of concurrent testing of one logic cabinetand associated RTB on a plant-specific basis to ensure conformance with the WCAP-15376 evaluation, including the guidance of RGs 1.174 and 1.177.

The licensee showed that the generic analysis presented in WCAP-15376 is applicable to STP, Units 1 and 2. WCAP-15376 previously did not specifically evaluate or preclude concurrent testing of one logic cabinet and associated RTB. In response to an NRC staff questionon WCAP-15376, the Pressurized-Water Reactor Owner's Group (PWROG) provided risk estimates for the logic train and associated RTBtrain 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 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:

To ensure consistency with the reference plant, the licensee should confirm that the model assumptions for human reliability in WCAP-15376 are applicable to the plant-specific configuration.

The licensee confirmed that the assumptions regarding human reliability used in WCAP-15376 is applicable to STP, Units 1 and 2. This review concluded that for the operator actions identified in WCAP-15376, plant procedures are available consistent with the assumptions in WCAP-15376. 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 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 at STP, Units 1 and 2.

Condition 6:

WCAP-15376 included an additional condition based on the PWROG response to a staff RAI that committed each plant to review its 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.

Based on the NRC staff's review for Condition 6 (see Sections 3.2.2.2 and Section 3.3 of this SE), the NRC staff considers that Condition 6 is satisfied for STP, Units 1 and 2.

3.2.2 WCAP-15376 Tier 1, 2, and 3 Analysisto 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. 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, which is used in evaluating the proposed RTB CT and test bypass time, is of sufficient scope and detail for this application. WCAP-15376 provided a generic PRA model for the evaluation of the CTand test bypass time. The NRC staff found this generic model and the WCAP-15376 evaluation to be acceptable on a generic basis in the SE datedDecember 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 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 via its SE. The WCAP-15376 conditions and limitations identified by the staff were considered limiting forSTP, in that the WCAP-15376 does not specify the use of the STP PRA or plant-specific estimates of ΔCDF, ΔLERF, ICCDP, or ICLERP in the implementation of theWCAP. However, in its SE for WCAP-15376, 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 PRA parameters to demonstrate the plant-specific applicability of the proposed CT and bypass test times. The licensee comparedlogic 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 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 datedDecember 4, 2014, the licensee demonstrated that the ATWS CDF of 4.3E-08 provided sufficient margin compared to the acceptance guidelines in RG 1.174 of 1E-06. 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, the NRC staff concludes that WCAP-15376 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,[1]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

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[1] 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).

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/Large Early Release Frequency Probabilistic Risk Assessment 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 surveillance 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 RMTSprogram, 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 Observationsrelated to the TS change. Therefore, the STP internal events PRA was determined, by NRC staff, to be of sufficient technical adequacy to support this application.

<u>PRA Results and Insights</u>

Cumulative Risk

WCAP-15376 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/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 implementation. Since the proposed change for STP is from TOP to WCAP-15376, the expected change in cumulative risk is expected to be less than the WCAP-15376 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 presents the external event risk contribution to the proposed TS change.

ExternalEvents Risk Contributions			
INITIATOR	IEFrequency	CDF	%CDF
TornadoInducedFailureofSwitchyardand EssentialCoolingPond	1.22E-06	1.11E-06	18.31
FireZone047ScenarioX	1.46E-05	3.65E-07	6.02
SwitchyardandEssentialCoolingPond Failure D uetoBreachofMainCooling Reservoir	3.20E-07	2.91E-07	4.80
FireZone071ScenarioX	2.34E-07	2.13E-07	3.51
FireZone047ScenarioB	2.72E-03	2.09E-07	3.45
ControlRoomFireScenario18	2.12E-06	9.12E-08	1.50
FireZone047ScenarioBC	3.18E-06	5.91E-08	0.98
SeismicEvent,0.4gAcceleration	7.74E-07	4.04E-08	0.67
ControlRoomFireScenario23	1.61E-06	2.62E-08	0.43
SeismicEvent,0.6gAcceleration	6.14E-08	2.08E-08	0.34
FireZone147Scenario	1.08E-03	1.19E-08	0.20
ExternalFloodingScenarios2Through6	1.05E-08	9.49E-09	0.16
SeismicEvent,0.2gAcceleration	2.89E-06	9.35E-09	0.15
SeismicEvent,0.1gAcceleration	3.02E-05	1.73E-09	0.03
ControlRoomFireScenario10	3.43E-06	1.04E-09	0.02
FloodInducedLOOP-Scenario1	3.20E-06	5.41E-10	0.01
GroupSubtotal	**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/year with the implementation of WCAP-15376. The estimated combined total CDF is estimated to be about 6.06E-6/year (3.6E-6/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/year, the focus should be on finding ways to decrease rather than increase risk. Given the typically conservative nature of the estimation of the fire, seismic, high winds, and flooding risk, the staff concludes that the total CDF is not expected to be higher than 1E-4/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 outofservice in accordance with the proposed TS change.

Based on WCAP-15376 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 finds 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).

The licensee evaluated concurrent component outage configurations and confirmed the applicability of the Tier 2 restrictions. Based on the above, the NRC staff concludes that the licensee’s Tier 2 analysis supports the implementation of WCAP-15376 and satisfies the conditions of the staff’sSE acceptanceof WCAP-15376 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 licenseehas 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 CRMPfor 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 defenseindepth. 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 and

- Response time testing and RTB 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 licensee will also perform the following Preventative Maintenance activities for the RTS instrumentation:

- Contingency maintenance support for the RTB 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 RTBand reactor trip bypass breaker is performed every refueling outage, and

- 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, as accepted by the NRC staff, including limitations and conditions identified in the NRC staff SE. As such, the implementation of WCAP-15376 is within the RG 1.174 and RG 1.177 acceptance guidance for ΔCDF, ΔLERF, ICCDP, and ICLERP.

3.2.4 Conclusion

The NRC staff concludesthat the licensee has demonstrated the applicability of WCAP-15376 to STP, Units 1 and 2,and has met the limitations and conditions as outlined in the staff's SE, which adopted WCAP-15376. The staff also concludes that the risk impacts for ΔCDF, ΔLERF, ICCDP, and ICLERP, as estimated by WCAP-15376 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 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. 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, and meet the staff's SE conditions and limitations for WCAP-15376 applicability. Therefore, based on the above evaluation, the NRC staff concludes that the proposed amendment to extend the RTB CT and bypass test time is acceptable.

3.3 Commitments

The licensee provided the following regulatory commitments as seen in the table below.

	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 discussed restrictions on equipment removal when an RTB is out of service. The staff compared the commitments provided in the LAR to the requested restrictions as found in the staff SE that approved WCAP-15376-P-A. The staff determined that the commitments were consistent with the NRC staff's WCAP-15376 SE and are, therefore, acceptable.

4.0 <u>STATE CONSULTATION</u>

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 <u>ENVIRONMENTAL CONSIDERATION</u>

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.

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