September 1, 2006

Mr. D. E. Grissette
Vice President
Southern Nuclear Operating
Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING THE EXTENSION OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM SURVEILLANCE TEST INTERVALS AND COMPLETION TIMES (TAC NOS. MC5863 AND MC5864)

Dear Mr. Grissette:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 145 to Facility Operating License NPF-68 and Amendment No. 125 to Facility Operating License NPF-1 for the Vogtle Electric Generating Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 27, 2005, as supplemented by letters dated September 30, 2005, and January 25 and May 5, 2006.

The amendments revise the surveillance requirements (SRs) and action statements for the TS 3.3.1, "Reactor Trip System (RTS) Instrumentation;" TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation;" TS 3.3.6, "Containment Ventilation Isolation Instrumentation;" and TS 3.3.8, "High Flux at Shutdown Alarm," for each unit.

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

Sincerely,

/RA/

Christopher Gratton, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 145 to NPF-68
2. Amendment No. 125 to NPF-81
3. Safety Evaluation

cc w/encls: See next page

Mr. J. T. Grissette
Vice President
Southern Nuclear Operating
Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

September 1, 2006

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING THE EXTENSION OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM SURVEILLANCE TEST INTERVALS AND COMPLETION TIMES (TAC NOS. MC5863 AND MC5864)

Dear Mr. Grissette:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 145 to Facility Operating License NPF-68 and Amendment No. 125 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 27, 2005, as supplemented by letters dated September 30, 2005, and January 25 and May 5, 2006.

The amendments **revise the surveillance requirements (SRs) and action statements for the TS 3.3.1, "Reactor Trip System (RTS) Instrumentation;" TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation;" TS 3.3.6, "Containment Ventilation Isolation Instrumentation;" and TS 3.3.8, "High Flux at Shutdown Alarm,"** for each unit.

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

Sincerely,

/RA/

Christopher Gratton, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 145 to NPF-68
2. Amendment No. 125 to NPF-81
3. Safety Evaluation

cc w/encls: See next page

<u>DISTRIBUTION:</u> See next page
Package Number: ML062360588
License Amendment Number: ML062360587
Tech Spec Number: ML

OFFICE	LPL2-1/PM	LPL2-1/LA	DSS/EICB	DRA/APLA	DIRS/ITSB	OGC(nlo w/comments)	LPL2-1/BC
NAME	CGratton	MO'Brien	AHowe	MRubin	TKobetz	SHamrick	EMarinos
DATE	08/30/06	08/30/06	08/09 /06	07/27/06	08/04 /06	08/22 /06	09/01/06

OFFICIAL RECORD COPY

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENT NOS.145 AND 125 REGARDING THE EXTENSION OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM SURVEILLANCE TEST INTERVALS AND COMPLETION TIMES (TAC NOS. MC5863 AND MC5864)

Date: September 1, 2006

DISTRIBUTION:

Public	RidsAcrsAcnwMailCenter
LPL2-1 R/F	GHill(4 hard copies)
RidsNrrDorlLpl2-1(EMarinos)	RidsNrrDirsItsb(TKobetz)
RidsNrrPMCGratton(hard copy)	RidsRgn2MailCenter(SSchaefer)
RidsNrrLAMO’Brien(hard copy)	RidsNrrDorlDpr
RidsOgcRp	MRubin, NRR
AHowe, NRR	SRhow, NRR
CDoutt, NRR	CSchulten, NRR

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.145
License No. NPF-68

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

    A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated January 27, 2005, as supplemented by letters dated September 30, 2005, and January 25 and May 5, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

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

    Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 145 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 1, 2006

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated January 27, 2005, as supplemented by letters dated September 30, 2005, and January 25 and May 5, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

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

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 125 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
  Changes

Date of Issuance: September 1, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 145

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

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

Remove Pages	Insert Pages
Licenses	Licenses
License No. NPF-68, page 4	License No. NPF-68, page 4
License No. NPF-81, page 4	License No. NPF-81, page 4
TSs	
3.3.1-7	3.3.1-7
3.3.1-10	3.3.1-10
3.3.1-10a	-
3.3.1-10b	-
3.2.2-7	3.3.2-7
3.3.6-4	3.3.6-4
3.3.8-2	3.3.8-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 125 TO FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By application dated January 27, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML050320191), as supplemented by letters dated September 30, 2005 (ML052730528), and January 25 (ML060260056) and May 5, 2006 (ML061280041), Southern Nuclear Operating Company, Inc. (SNC, the licensee) requested changes to the technical specifications (TSs), for Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle). The supplements provided additional information that clarified the application, but did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 8, 2005 (70 FR 67751).

1.1 Proposed License Amendment

The proposed amendment adopts changes approved by the NRC staff in its letter dated December 20, 2002, wherein it approved the Westinghouse Commercial Atomic Power (WCAP) topical report number 15376 (WCAP-15376) for referencing in licensing applications (Reference 1). The proposed change revises TS limiting conditions for operation (LCO) 3.3.1, 3.3.2, 3.3.6, and 3.3.8. Specifically, the reactor trip breaker (RTB) bypass time is revised from 2 hours to 4 hours, the RTB completion time (CT) from 1 hour to 24 hours, and the surveillance test interval (STI) from 2 months to 4 months. The STIs for the logic cabinets are changed from 2 months to 6 months, the master relays from 2 months to 6 months, and the analog channels from 3 months to 6 months. Although not part of the TSs, the licensee provided associated bases changes associated with the proposed TS changes.

The proposed changes are implemented per Technical Specification Task Force (TSTF) traveler number 411, revision 1, (TSTF-411), as approved by the NRC staff's letter dated August 30, 2002, (Reference 2).

The licensee stated that the proposed changes will reduce the required testing of the reactor protection system (RPS) and the engineered safety features actuation system (ESFAS) components without significantly impacting system reliability. The proposed change will also reduce the potential for reactor trips and actuation of engineered safety features associated

with the testing of these components. The CT for the RTBs will provide additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with RTB TSs. Additionally, an extended RTB CT will provide consistency with the logic cabinet CT, as implemented by WCAP-14333, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," dated October 1998, as approved by NRC letter dated July 15, 1998.

1.2 Related NRC Actions

This license amendment is not related to, or in response to, any ongoing NRC activities (e.g., generic letters).

1.3 Background

1.3.1 Technical Specifications Improvement Project

Since 1983, NRC and industry representatives (e.g., the Westinghouse Owners Group (WOG)) have worked to develop guidelines for improving nuclear power plant TS content and quality. In August 1983, an NRC task group was formed to investigate problems with surveillance testing required by TSs and to recommend approaches to make improvements. The results of the task group study were published in November 1983 in NUREG-1024, "Technical Specifications – Enhancing Safety Impact." NUREG-1024 recommended that the staff (1) review the bases for TS test frequencies, (2) ensure that TS-required tests promote safety and do not degrade equipment, and (3) to review surveillance tests so that they do not unnecessarily burden personnel.

In December 1984, the Technical Specifications Improvement Project (TSIP) was established to provide a framework to address the recommendations of NUREG-1024, and for rewriting and improving the TSs. The NRC developed criteria, as described in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," (58 *Federal Register* (FR) 39132), to determine which of the design conditions and associated surveillances should be located in the TSs as an LCO. The results of this program were reported in NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements [SRs]," dated December 1992. NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 0, the improved standard technical specifications (STS) for Westinghouse plants, were issued using the recommendations for defining the scope of TSs for these plants. Four additional criteria were subsequently incorporated into the regulations by an amendment to Title 10 of the *Code of Federal Regulations*, Part 50, Section 50.36 (10 CFR 50.36) (60 FR 36953). Generic Letter (GL) 93-05, dated September 27, 1993, was prepared to provide guidance to licensees for preparing license amendment requests to implement the recommendations as line-item TS improvements.

1.3.2 Technical Specification Optimization Program (TOP)

The TSIP study formed the basis for the WOG TOP. The WOG TOP evaluated changes to STIs and allowed outage times (AOTs) for the analog channels, logic cabinets, master and slave relays, and RTBs. The methodology evaluated increases in surveillance intervals, test and maintenance out-of-service times, and the bypassing of portions of the RPS during test and maintenance. In 1983, the WOG submitted WCAP-10271-P, "Evaluation of Surveillance

Frequencies and Out-of-Service Times for the Reactor Protection Instrumentation System," which provided a methodology to be used to justify revisions to a plant's RPS TSs. The WOG stated in WCAP-10271 that plant staff devoted significant time and effort to perform, review, document, and track surveillance activities that, in many instances, may not be required on the basis of the high reliability of the equipment. The justification for the changes was the small impact that these changes would have on plant risk.

In WCAP-10271, the WOG performed fault tree analyses to calculate the reactor trip unavailability, with consideration for surveillance intervals and test and maintenance times. The sensitivity to variations in surveillance intervals and test and maintenance times was also evaluated with respect to maintaining or revising current surveillance intervals. The WOG concluded that the results of the analyses for the RPS were adequate to justify a revision of the STSs. WCAP-10271, including Supplement 1, was accepted, with conditions, by NRC letters dated February 21, and July 24, 1985 (References 3 and 4), in which the NRC staff approved the following changes for plant-specific TSs:

- Increase the surveillance interval for reactor trip system (RTS) analog channel operational tests from once per month to once per quarter.
- Increase the time in which an inoperable RTS analog channel may be maintained in an untripped condition from 1 hour to 6 hours.
- Increase the time an inoperable RTS analog channel may be bypassed to allow testing of another channel in the same function from 2 hours to 4 hours. Also, the channel test may be done in the bypass mode, leaving the inoperable channel in a tripped condition.
- Allow testing of the RTS analog channels in a bypass condition instead of a tripped condition.

On February 22, 1989 (Reference 5), the NRC staff issued a safety evaluation report (SER) for WCAP-10271, Supplement 2, that approved similar relaxations for the ESFAS. An additional supplemental SER was issued on April 30, 1990 (Reference 6), that provided consistency between RTS and ESFAS STIs and CTs.

Subsequent to the approval of WCAP-10271 and its supplements, the WOG submitted WCAP-14333-P, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," dated May 1995. The NRC staff accepted WCAP-14333 by letter dated July 15, 1998. The purpose of this topical report was to provide justification for the following additional TS relaxations beyond those approved in WCAP-10271.

- Increase the bypass times and CTs for both the solid-state and relay protection system RPS and ESFAS designs.
- For the analog channels, increase the CT from 6 hours to 72 hours and the bypass time from 4 hours to 12 hours.
- For the logic cabinets, master relays, and slave relays, increase the CTs from 6 hours to 24 hours.

- Revise the action statement for an inoperable slave relay to increase the CT for maintenance to 24 hours, with an additional 6 hours for the mode change.

- For cases where the logic cabinets and trip breakers both cause their train to be inoperable when in test or maintenance, allow the reactor trip breakers to be bypassed for the period of time equivalent to the bypass time for the logic cabinets, provided that both are tested at the same time.

Finally, WCAP-15376 was submitted to the NRC staff on November 8, 2000, and was approved by NRC letter dated December 20, 2002 (Reference 1). The purpose of WCAP-15376 was to provide a technical justification for additional extensions to the STI for components of the RPS and ESFAS, in addition to those already approved in WCAP-10271 and WCAP-14333. The components specifically evaluated were the analog channels, logic cabinets, master relays, and RTBs. WCAP-15376 evaluated both the solid state protection system (SSPS) and the relay protection system. In the case of Vogtle, both WCAP-10271 and WCAP-14333 have been incorporated into the Vogtle TSs by previous NRC-approved license amendment requests.

1.3.3 Technical Specification Task Force Changes

For WCAP-15376, the NRC has approved TS changes to incorporate the extended test bypass times, CTs, and STIs for certain RTS/ESFAS functions referenced in TSTF-411, Revision 1. The TS changes applicable to plants with an SSPS such as Vogtle, are the following.

- Increase the logic cabinet STI from 1 month to 6 months.

- Increase the master relay STI from 2 months to 6 months.

- Increase the STIs for analog channels from 3 months to 6 months.

- Increase the STIs for RTBs from 2 months to 4 months.

- Increase the RTB test bypass time from 2 hours to 4 hours and the RTB CT from 1 hour to 24 hours.

1.4 Surveillance Test Intervals and Completion Times

A CT extension increases the unavailability of a component due to the increased time the component is down for maintenance. For CTs, the designated CTs may not provide adequate time for repair, but longer CTs may incur a relatively larger risk. There are two components to the risk impact: (1) the single event risk when the CT is invoked and the component is down for maintenance, and (2) the yearly risk contribution based on the expected number of times the CT will be implemented. The yearly CT risk contribution is reflected in the *frequency* per year, based on adjusting the component unavailability due to estimated yearly mean outage time. The yearly CT risk impact is represented by the core damage frequency (CDF) and large early release frequency (LERF) metrics referenced on Regulatory Guide (RG) 1.174. The single event risk is represented by the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP) metrics referenced in RG 1.177 and reflects the *probability* of core damage or large early release during the period a component is down for maintenance.

By contrast, STIs are intervals for scheduled periodic surveillance tests, as required by the TSs. Such tests are performed to ensure that safety-related equipment continues to be operable and failures are detected, thereby limiting the fault exposure time. The primary risk contribution attributed to increasing an STI comes from the increased probability that the component will be inoperable during the surveillance interval. The extension of an STI affects the yearly risk, which is represented by CDF and LERF, in several ways:

- Reduces the risk by decreasing the number of test-caused reactor trips by limiting the opportunity for test-caused errors. This occurs simply because increasing the STI decreases the amount of testing over a given time.

- Reduces the risk by decreasing the unavailability of the component due to testing by reducing the test frequency.

- Increases the risk by increasing the fault exposure time as stated above. This is attributable to the fact that the increased STI increases the interval during which equipment is subject to failure while in standby. As the fault exposure time increases, there is a greater probability that component failures will occur while in standby.

2.0 <u>REGULATORY EVALUATION</u>

2.1 Description of Structures, Systems, and Components

The proposed TS modifications affect the RPS (RTS and ESFAS). The RTS is designed to shutdown the reactor when a limit to permissible operation is reached. The ESFAS is designed to actuate for transients that challenge the normal control and heat removal systems.

The RPS comprises several major functions including instrumentation, logic, reactor trip, and ESFAS actuation. Instrumentation consists of sensors, power supplies, signal processing, and bistable outputs, and typically consists of three or four channels. The logic (i.e., logic cabinets) includes two parallel logic blocks consisting of two trains (A and B) of RPS logic where the input coincidence for various trip functions is determined. The RPS logic provides two parallel outputs for ESFAS actuation. Each output is actuated by its associated RPS logic train, which initiates the ESFAS function through master and slave relays.

Additionally, two parallel actuation paths are provided from the RPS logic to the RTBs. Normally, an RTB receives its signal from its associated RPS logic train. Bypass breakers are provided for when a breaker is out of service. In this configuration, the bypass breaker is associated with the logic train of the operable RTB. The RPS utilizes two normally closed RTBs and two normally open bypass breakers. Train A RPS logic actuates RTB A and train B logic actuates RTB B. Opening of either RTB will disconnect power from the control rods, causing a reactor trip.

The Vogtle RPS utilizes the SSPS for the logic portion of the RPS.

2.2 Applicable Regulations

Pursuant to 10 CFR 50.36, "Technical specifications," a licensee's TSs must have SRs relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operations are within safety limits, and that the LCOs will be met. The SRs may include mode restrictions based on the safety aspects of conducting the surveillances in excluded modes.

Although the rule does not specify specific TS requirements, implicit within 10 CFR 50.36 is the requirement for required actions for an LCO not being met and that the test bypass times, CTs, and STIs specified in the TSs must 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 RTS/ESFAS functions affected by these proposed TS changes will perform their required safety functions in accordance with the design-basis accidents in Chapter 15 of the licensee's Final Safety Analysis Report, based on the proposed test bypass times, CTs, and STIs.

The Maintenance Rule, 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants,” requires licensees to monitor the performance or condition of structures, systems, and components (SSCs) against licensee-established goals, in a manner sufficient to provide reasonable assurance that SSCs are capable of fulfilling their intended functions as applicable to the implementation and monitoring program guidance of RG 1.174, Section 2.3 and RG 1.177, Section 3. In addition, 10 CFR 50.65(a)(4), as it relates to the proposed CT extension, requires the assessment and management of the increase in risk that may result from the proposed maintenance activity.

According to 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 13, “Instrumentation and control,” appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

According to GDC 21, “Protection system reliability and testability,” the protection system shall be designed for high functional reliability and inservice testability, commensurate with the safety functions to be performed. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Finally, 10 CFR 50.90, “Application for amendment of license or construction permit,” addresses the requirements for a licensee desiring to amend its license, which includes the TSs.

2.3 Applicable Regulatory Criteria/Guidelines

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Chapter 19.0, “Use of Probabilistic Risk Assessment (PRA) in Plant-Specific, Risk-Informed Decisionmaking: General Guidance,” of the NRC Standard Review Plan (SRP), NUREG-0800. More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, “Risk-Informed Decisionmaking: Technical Specifications,” which includes CT changes as part of risk-informed decisionmaking. Chapter 19.0 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

- The proposed change is consistent with the defense-in-depth philosophy.

- The proposed change maintains sufficient safety margins.

- When proposed changes increase core damage frequency or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

- The impact of the proposed change should be monitored using performance measurement strategies.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment [PRA] in Risk-Informed Decisions on Permanent Plant-Specific Changes to the Licensing Basis," dated November 2002, describes a risk-informed approach, acceptable to the NRC, for licensee's to assess the nature and impact of proposed licensing basis changes by considering engineering issues and applying risk insights.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated August 1998, identifies an acceptable risk-informed approach including additional guidance geared toward the assessment of proposed TS CT changes. Specifically, RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed CT TS change as shown below.

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out-of-service, as represented by ICCDP and ICLERP. Tier 1 also addresses PRA quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Cumulative risk of the present TS change in light of past applications or additional applications under review is also considered along with the uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The purpose of this evaluation is to ensure that there are appropriate restrictions in place such that risk-significant plant equipment outage configurations will not occur when equipment associated with the proposed CT is implemented.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk-significant configurations that

may not have been considered when the Tier 2 evaluation was performed. Compared with Tier 2, Tier 3 provides additional coverage to ensure risk-significant plant equipment outage configurations are identified in a timely manner and that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (10 CFR 50.65(a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing, and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The CRMP is to ensure that equipment removed from service prior to or during the proposed extended CT will be appropriately assessed from a risk perspective.

More specific methods and guidelines acceptable to the staff are also outlined in RG 1.177 for assessing risk-informed TS changes. Specifically, RG 1.177 provides recommendations for utilizing risk information to evaluate changes to TS CTs and surveillance test intervals with respect to the impact of the proposed change on the risk associated with plant operation.

RG 1.174 and RG 1.177 also describe acceptable implementation strategies and performance monitoring plans to help ensure that the assumptions and analysis used to support the proposed TS changes will remain valid. The monitoring program should include means to adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's evaluation for the proposed licensing basis change. RG 1.174 states that monitoring performed in accordance with the Maintenance Rule, 10 CFR 50.65, can be used when the monitoring performed under the maintenance rule is sufficient for the SSCs affected by the risk-informed application.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee's analysis in support of its proposed license amendment, which is described in the original submittal dated January 27, 2005, as supplemented by letters dated September 30, 2005, January 25, and May 5, 2006.

3.1 Detailed Description of the Proposed Change

The proposed changes in surveillance test intervals and the RTB CT are based on WCAP-15376, as approved in Reference 1 and TSTF-411, Revision 1, as approved in Reference 2.

The proposed changes revise LCOs 3.3.1, "Reactor Trip System Instrumentation," LCO 3.3.2, "Engineered Safety Feature Actuation System Instrumentation," LCO 3.3.6, "Containment Ventilation Isolation Instrumentation," and LCO 3.3.8, "High Flux at Shutdown Alarm."

The proposed changes, applicable to SSPS plants, are summarized in the table below.

RPS Component	STI	CT	Bypass Time for Testing

	Current (Month)	Proposed (Month)	Current (Hour)	Proposed (Hour)	Current (Hour)	Proposed (Hour)
Logic Cabinets	2	6	No Change Requested		No Change Requested	
Master Relays	2	6				
Analog Channels	3	6				
Reactor Trip Breakers	2	4	1	24	2	4

3.2 Review of Methodology

Per SRP Chapter 19 and Section 16.1, the NRC staff reviewed the submittal using the three-tiered approach and the five key principles of risk-informed decisionmaking presented in RG 1.177.

3.3 Key Information Used in the Review

The key information used in the staff's risk evaluation is contained in Enclosure 1A of the license amendment request, as supplemented by the request for additional information (RAI) responses dated September 30, 2005, January 25, and May 5, 2006, TSTF-411, and the conditions and limitations identified in the NRC staff's SE for WCAP-15376 (Reference 1). The NRC staff also referenced previous staff SEs related to WCAP-14333 and WCAP-10271 and the licensee's individual plant examination (IPE) and individual plant examination of external events (IPEEE) and the associated staff SERs.

3.4 Comparison Against Regulatory Criteria/Guidelines

The NRC staff's evaluation of the licensee's proposed amendment to extend STIs, bypass times, and CTs using the three-tiered approach and the five principles outlined in RGs 1.174 and 1.177 are presented in the following sections.

3.4.1 Traditional Engineering Evaluation

The NRC staff previously performed a generic evaluation of WCAP-15376-P, Rev 0, and documented its results in Reference 1. The NRC staff's review of the changes found that WCAP-15376 was consistent with the accepted guidelines of RG 1.174 and RG 1.177, and NRC staff guidance as outlined in NUREG-0800, "Standard Review Plan." From traditional engineering insights, the NRC staff found that the proposed changes in WCAP-15376 continue to meet the regulations, have no impact on the defense-in-depth philosophy, and would not involve a significant reduction in the margin of safety.

3.4.2 Risk Evaluation

The changes proposed by the licensee employ a risk-informed PRA approach using risk insights to justify changes to TS CTs and STIs. The risk metrics ΔCDF, ΔLERF, ICCDP, and

ICLERP used by the licensee to evaluate the impact of the proposed changes are consistent with those presented in RGs 1.174 and 1.177. The subsections below discuss the applicability of WCAP-15376 to Vogtle and the associated reviews.

3.4.2.1 NRC Staff's Technical Evaluation

3.4.2.1.1 Applicability of WCAP-15376 to Vogtle.

To determine whether WCAP-15376 is applicable to Vogtle, the licensee addressed the conditions and limitations of the staff's SE (Reference 1) and the implementation guidance developed by the WOG 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 the WCAP generic evaluation assumptions were also applicable to Vogtle.

The licensee's evaluation of each of the WCAP-15376 conditions and limitations is discussed below.

1. A licensee is expected to confirm the applicability of the topical report to their plant, and to perform a plant-specific assessment of containment failures and address any design or performance differences that may affect the proposed changes.

To demonstrate the applicability of WCAP-15376 to Vogtle, the licensee performed a comparison of WCAP-15376 generic assumptions and data with Vogtle plant-specific parameters. This comparison included the base component bypass, test, and maintenance intervals to those assumed by WCAP-15376. The evaluation also included confirmation that procedures are in place for operator actions assumed by the generic analysis and are applicable to Vogtle. The contribution from anticipated transient without scram (ATWS) events were also confirmed to be consistent with that assumed in the generic analysis. The applicability of the reactor trip actuation signals in the topical report were also confirmed for Vogtle. Component failure probabilities were compared to the topical report assumptions and found to be applicable to Vogtle. The applicability of WCAP-15376 is discussed further in Section 3.4.2.1.2, "Tier 1: PRA Capability and Insights."

In addition, SE Condition 1 stated that a plant-specific assessment of containment failure should be performed for plants referencing WCAP-15376, based on the topical report assumption that the only contributions to LERF would come from containment bypass events and core damage events with the containment not isolated. This assumption was based on the WCAP-15376 PRA model. Since the topical report was based on Vogtle with a large dry containment, the assumptions of the WCAP-15376 analysis are applicable to Vogtle. In addition, LERF-related findings from the Vogtle industry peer review were incorporated into the Vogtle PRA revision used for the implementation of WCAP-15376. Based on the above, SNC has met Condition 1 of the NRC staff's SE for WCAP-15376 for Vogtle.

2. Address the Tier 2 and Tier 3 analyses including risk-significant configuration insights and confirm that these insights are incorporated into the plant-specific CRMP.

Based on the evaluation presented under Section 3.4.2.1.3, “Tier 2 - Avoidance of Risk-Significant Plant Configurations," and Section 3.4.2.1.4, “Tier 3 - Risk-Informed Configuration Risk Management,” of this SE, the NRC staff considers Condition 2 satisfied for Vogtle.

3. The risk impact of concurrent testing of one logic cabinet and associated RTB needs to be evaluated on a plant-specific basis to ensure conformance with the WCAP-15376 evaluation, including RG 1.174 and RG 1.177 guidance.

The licensee stated that the generic analysis presented in WCAP-15376 is applicable to Vogtle. However, concurrent testing of one logic cabinet and associated RTB was not specifically evaluated or precluded by WCAP-15376. Based on this, the NRC staff questioned the applicability of the topical report when in this particular maintenance configuration. In response to a staff RAI, the WOG provided risk estimates for this more limiting configuration. The resulting generic ICCDP estimate was found to be within the acceptance guidelines of RG 1.177. Based on the applicability of WCAP-15376 to Vogtle and an ICCDP estimate within the acceptance guidelines of RG 1.177, the NRC staff considers Condition 3 satisfied.

4. To ensure consistency with the reference plant, the model assumptions for human reliability in WCAP-15376 should be confirmed to be applicable to the plant-specific configuration.

SNC confirmed that the assumptions regarding human reliability used in WCAP-15376 were applicable to Vogtle. This review concluded that operator actions were identified and that plant procedures are available consistent with the assumptions in WCAP-15376. However, SNC noted that the topical report recognized operator action to achieve reactor trip by interrupting power to the motor-generator sets. This operator action is not credited in the Vogtle PRA. The licensee stated that Vogtle has a higher probability of recovery failure should the automatic reactor trip fail, but also noted that Vogtle has a lower frequency for ATWS transient initiators than the generic analysis. SNC stated that with the combined effects, the applicability of WCAP-15376 to Vogtle is maintained. In addition, the licensee stated that operators would be sent to locally open the RTBs in the event of an ATWS. Based on the above, NRC the staff considers Condition 4 of the SE to be satisfied.

5. For future digital upgrades with increased scope, integration and architectural differences beyond that of Eagle 21, the NRC staff finds that the generic applicability of WCAP-15376 to a future digital system is not clear and should be considered on a plant-specific basis.

Because the Vogtle RPS and ESFAS design is based on the SSPS, this condition is not currently applicable to the implementation of WCAP-15376 at Vogtle.

6. WCAP-15376 included an additional condition based on the WOG response to an NRC staff RAI that committed each plant to review their plant-specific setpoint calculation methodology to ensure that the extended STIs do not adversely impact the plant-specific setpoint calculations and assumptions for instrumentation associated with the extended STIs.

The WOG response to the NRC RAI noted that plant-specific RPS and ESFAS setpoint uncertainty calculations and assumptions, including instrument drift, will be reviewed to

determine the impact of extending the surveillance interval of the channel functional test from 92 days to 184 days.

SNC has a process for trending technical specification instrument performance. The trending information is obtained from calibration data sheets. The recorded “as found” and “as left” values are placed into a trending database and evaluated by the system engineer. This assures that the values remain within the drift allowances used in the setpoint methodology. If pre-determined limits are exceeded, the condition is reported in the plant Condition Reporting and Tracking System. The pre-determined limits are set within the allowable values specified in TS Tables 3.3.1-1 and 3.3.2-1. The system engineer’s review of channel operational test (COT) trending data has indicated that the instrument loops will remain within existing TSs and uncertainty calculation limits should the COT frequency change to 184 days.

This review noted that a comparison of the “as found” to previous “as left” data resulted in drift magnitudes that are well within the process rack operability criteria specified by Vogtle procedures and consistent with operability criteria for a healthy channel as defined by the process rack vendor. The licensee did not note any significant adverse trends or history of frequent process rack re-calibration in this review for the current COT frequency. Based on review of the information discussed above, the NRC staff concludes that the extension of the COT frequency to 184 days will not affect the assumptions of the setpoint calculation methodology and the expected process rack drift will remain consistent with current operability criteria. On this basis, the changes justified in WCAP-15376 can be applied to Vogtle.

3.4.2.1.2 Tier 1: PRA Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk based on the Vogtle/WCAP-15376 PRA model. The Tier 1 staff review involves two aspects: (1) evaluation of the validity of the PRA and its application to the proposed changes, and (2) evaluation of the PRA results and insights stemming from its application.

PRA Quality

The objective of the PRA quality review is to determine whether the Vogtle/WCAP-15376 PRA used in evaluating the proposed RTS and ESFAS STI and CT extensions are of sufficient scope, detail, and technical adequacy for this application. In this case, WCAP-15376 provided a generic PRA model used for the evaluation. This generic model and the WCAP-15376 evaluation were found by the staff to be acceptable on a generic basis in the SE dated December 20, 2002. The staff reviewed the information provided in the proposed license amendment request, as well as the findings of the staff’s WCAP-15376 SE, and peer review results. The WCAP-15376 conditions identified by the staff were considered limiting by SNC in that they do not require the plant-specific use of the Vogtle PRA and plant-specific estimates of ΔCDF, ΔLERF, ICCDP, or ICLERP in the incorporation of WCAP-15376 at Vogtle. However, a review of the scope, detail, and technical adequacy of the licensee’s PRA and the licensee’s cross comparisons to the representative WCAP-15376 PRA model help demonstrate the plant-specific applicability of the proposed amendment request to extend the Vogtle RPS and ESFAS STIs and CTs.

WCAP-15376 used the Vogtle PRA model to estimate the impact on risk of the proposed changes. The Vogtle PRA was developed in response to Generic Letter (GL) 88-20, "Individual

Plant Examination for Severe Accident Vulnerabilities." The NRC staff concluded that the Vogtle IPE and IPEEE met the intent of GL 88-20. The Vogtle PRA model was previously utilized in WCAP-14333 to provide the basis for extending CTs and bypass times for the RPS and ESFAS. Since the requested surveillance interval and CT requests in WCAP-15376 are similar in scope to those requested by WCAP-14333, the WOG utilized a similar model for WCAP-15376. WCAP-15376 provides various insights as to the appropriateness of the Vogtle model in support of the proposed TS changes stating that the model provides sufficient detail to perform the analysis, the Vogtle model includes anticipated transients without scram (ATWS) and this model allows operator action to be credited as backup to automatic actuation.

Based on the fact that the WCAP-15376 PRA model is similar to the models used for previous WOG topical report evaluations (including WCAP-14333), the NRC staff found the quality of the representative model to be sufficient for the evaluation of the proposed changes. For the plant-specific case, SNC addressed the SER conditions and performed additional plant comparisons to the generic analysis demonstrating the applicability of the generic analysis to Vogtle.

Peer Review

The Vogtle PRA underwent industry peer review by the WOG peer review team in December 2001. The licensee stated that no Level “A” facts and observations (F&Os) were identified during the peer review. Following the peer review, SNC updated their PRA model to address three findings considered the most significant among the Level “B” F&Os. The licensee, in its RAI response, also addressed the impact of the remaining Level “B” F&Os and their impact on the proposed RPS and ESFAS CT and STI extensions. The Level “C” and “D” findings were not evaluated, consistent with the understanding that Level “C” and “D” findings are not expected to significantly impact results or conclusions or are editorial in nature. The licensee concluded that the remaining Level “B” F&Os would not impact the conclusions of WCAP-15376 and its implementation at Vogtle.

In addition to the peer review, the NRC staff reviewed the results of the Vogtle PRA model as part of the benchmarking of the Vogtle Significance Determination Process notebook. This review was conducted by the staff and its contractors at SNC in Birmingham, Alabama, during March of 2002. The NRC staff’s Phase 2 Notebook benchmarking report stated that benchmarking improved the notebook, reducing the under-estimations and improving the consistency of the results. The benchmark report also noted that the licensee’s PRA staff was knowledgeable of the plant model and that the licensee provided very helpful comments that improved the quality and content of the notebook.

PRA Update/Procedures

The Vogtle PRA was initially developed for the IPE and, as stated by SNC, represented the as-built, as-operated plant at that time. The Vogtle PRA was later updated from the large event tree methodology to the linked fault tree methodology and converted to the computer aided fault tree analysis software and designated as Revision 0, dated March 1998. Revision 0 added design features to credit the Plant Wilson combustion turbine facility. Revision 1, dated

August 1999, added additional new operator actions. Revision 2, dated January 2000, incorporated updated basic event plant data and incorporated plant design changes through April 1998. The Vogtle PRA was then updated November 11, 2003, to Revision 2cy to resolve three Level "B" items, that were identified in the industry peer review. The IPEEE has not been updated since it was submitted for NRC review on November 1, 1995, as supplemented September 10, 1997.

The Vogtle PRA is documented in accordance with procedure PS-PRA-008, Version 1, "PRA Calculations - Preparation and Revision," and PS-PRA-009, Version 1, "PRA Calculation Administration." The licensee stated that the PRA revision schedule is controlled by procedure PS-PRA-001, Version 2, "Generation and Maintenance of Probabilistic Risk Assessment Models and Associated Updates." PRA input data is reviewed approximately every 3 years and the PRA updated in accordance with the schedule determined subsequent to the above scoping review.

PRA Results and Insights

WCAP-15376 evaluated the increase in STIs for the analog channels, logic cabinets, master relays, and the RTBs along with an increase in the RTB CTs. The generic model used data reported in previous studies for WCAP-10271 and WCAP-14333, data from NUREG/CR-5500, Volume 2, "Reliability Study: Westinghouse Reactor Protection System, 1984 - 1985," dated April 1999, and plant surveys. The base case assumed in WCAP-15376 used the proposed CTs and bypass times approved for WCAP-14333. The STI times were not revised for WCAP-14333 and, therefore, are also representative of WCAP-10271. The WCAP-15376 PRA model was also based on the Vogtle PRA, as modified by the topical report. The model was based on a plant with an SSPS instead of a relay protection system. WCAP-15376 concluded that the signal unavailability values for a relay plant were consistently smaller than those for a plant equipped with an safe shutdown protection system (SSPS). Therefore, the analysis for an SSPS plant was considered bounding for a relay plant.

The WCAP-15376 risk impacts were found to be within the RG 1.174 acceptance guidelines of less than 1E-6 for ΔCDF and less than 1E-7 for ΔLERF for the proposed STIs, bypass times, and CTs. The ICCDP for the increased RTB CT was also found to be within the RG 1.177 acceptance guideline of less than 5E-7. The ICCDP estimates presented in the topical report considered only an RTB out of service with the associated bypass breaker available. However, based on WCAP-14333 accepting the case where an RTB and its associated logic cabinet can be out of service simultaneously, the above ICCDP estimates were not applicable for all maintenance configurations. The WOG, through RAI responses to WCAP-15376, verified that, for the more limiting configuration of an RTB and logic cabinet out of service simultaneously, the ICCDP estimate was still within the acceptance guidelines of RG 1.177. The ICLERP estimate was also within the RG 1.177 acceptance guideline of 5E-8.

The implementation guideline developed by the WOG provides guidance on meeting the WCAP-15376 condition to ensure that the model assumptions for human reliability are consistent with the reference plant and applicable to the plant-specific case. The licensee's response to the WOG implementation guidance confirms that plant procedures and operator actions assumed for the reference plant are also applicable to the Vogtle plant-specific case.

Cumulative Risk

The above ΔCDF and ΔLERF estimates were developed using the STIs, bypass times, and CTs from WCAP-14333 as the base case. The cumulative CDF risk was also evaluated from the pre-TOP STIs and CTs to WCAP-15376. In this case, the cumulative impact on CDF for 2-out-of-4 logic was found to be within the RG 1.174 acceptance guidelines of less than 1E-6, representing a very small change. The cumulative impact on CDF for 2-out-of-3 logic was slightly above the RG 1.174 acceptance guideline for a very small change but within the acceptance guidelines for a small change.

Cumulative risk of the present TS change in light of past applications or additional applications under review were also considered by SNC. The licensee identified four previous risk-informed license amendment requests submitted under RG 1.174 and RG 1.177 and provided their respective ΔCDF and ΔLERF estimates. The cumulative risk estimates of the referenced amendments were within RG 1.174 acceptance guidelines for a small change. Considering the cumulative risk estimates given in WCAP-15376, it is expected that the overall cumulative risk for Vogtle will remain within the RG 1.174 acceptance guidelines for a small change. However, WCAP-15376 is being implemented as a bounding analysis, therefore, the Vogtle PRA was not specifically updated for this amendment request and as a result the cumulative risk estimates of WCAP-15376 and the plant-specific amendments are not specifically reflected in the Vogtle PRA. Still, it is expected that the cumulative change in CDF and LERF will remain within the RG 1.174 acceptance guidelines for a small change based on the generic analysis results of WCAP-15376 and the applicability of WCAP-15376 to Vogtle.

External Events

The licensee evaluated the proposed RPS and ESFAS STI and CT extensions for their potential impact on external events including fire, seismic events, and high winds, floods and other (HFO) events. These events are discussed below.

Fires

SNC has a fire PRA for Vogtle Units 1 and 2 that was originally developed as part of the IPEEE. Based on the IPEEE, CDF due to fires was estimated to be 1.01E-5/year or 22.7 percent of internal events CDF. The Vogtle IPEEE concluded that there were no significant vulnerabilities from fire-initiated events and did not identify any plant or procedural enhancements due to fire event risk. The IPEEE specifically identified the top five dominant fire zones as the main control room, 4160 volt train A and train B switchgear rooms, the lower cable spreading room, and the train B electrical penetration area. The first four scenarios developed for these fire zones involve a loss of offsite power. Except for a fire in the train B electrical penetration area, which does not credit recovery actions in the analysis, all the above scenarios credit manual action to mitigate the event. A further review of the top ten fire risk scenarios also show manual recovery actions credited to mitigate the event.

The licensee performed an evaluation to determine the impact of the proposed extended STIs and CTs on the Vogtle fire PRA. This evaluation looked at the top three fire zones, which were the main control room, and the 4160 volt train A and B switchgear rooms. The licensee confirmed that for these sequences, the Vogtle fire risk analysis credits operator action to mitigate the above sequences. The licensee stated that because the operator actions are not impacted by RPS or ESFAS unavailability, the impact of the proposed extended STIs, bypass times, and CTs on the fire risk would be small. Although the licensee only evaluated the top

three fire zones, an NRC staff review of the top 10 fire zones showed similar fire zone results. The licensee did not evaluate the impact of the proposed STIs, bypass times, and CTs with respect to the impact on the fire zone screening criteria (i.e., whether the extended STIs, bypass times, and CTs would screen in additional fire zones). However, based on the fire zones identified by the IPEEE, the impact of extended STIs, RTB bypass times, and CTs, should have a small impact on fire risk. Based on the above, the licensee's proposed extended STIs, bypass times, and CTs show a small impact on fire risk and are acceptable to the staff.

Seismic Events

SNC did not develop a seismic PRA for Vogtle Units 1 and 2, but rather employed a seismic margins assessment for the IPEEE submittal. Based on the IPEEE seismic margin evaluation and the probability of an earthquake greater than that assumed by the seismic margin evaluation during the proposed CTs the seismic contribution to core damage due to the increased CT and STIs is expected to be minimal.

The licensee evaluated low ruggedness relays as part of the IPEEE. The principal method used by the IPEEE for screening was to review the Vogtle Equipment Qualification Data Packages for the suspect relays. Based on this review, SNC concluded that there are no low-ruggedness relays used in safety-related equipment at Vogtle. The NRC staff's evaluation of the IPEEE concluded that the method used by the licensee for relay chatter evaluation appeared to be consistent with the recommendations of NUREG-1407 for focused-scope plants. Based on the above, the NRC staff considers the impact on the licensee's proposed extended STIs, bypass times, and CTs to be insignificant.

High Winds, Floods, and Other (HFO) External Events

Based on the IPEEE, all HFO events were screened according to the 1975 SRP screening criteria. The licensee did not identify any plant vulnerabilities and did not identify improvements associated with HFO events. The licensee concluded that the proposed RPS and ESFAS extended CTs and STIs remain acceptable with respect to HFO external events.

Based on the above, the NRC staff finds that the analysis of the reference plant has been shown to be applicable to the plant-specific case and supportive of the proposed STI and CT extensions requested for Vogtle, and, therefore, is acceptable.

Shutdown Risk (RTB extended CT)

WCAP-15376 stated that transition risk for the RTBs would be decreased with incorporation of a longer CT since transitions to lower modes would be decreased with the longer CT. WCAP-15376 stated that the transition risk would be comparable to the risk increase caused by the requested CT for the RTBs. However, the NRC staff noted that the additional benefit to transition risk would only occur when unscheduled corrective maintenance could not be completed within the TS CT. For failures occurring during an RTB surveillance, transition risk should be considered, but this has limited impact on the WCAP-15376 analysis. However, WCAP-15376 stated that the intent of the increased RTB CT was to provide operational flexibility to conduct testing of the logic cabinet in conjunction with corrective maintenance of an

RTB. This configuration is a higher risk configuration than the one RTB inoperable configuration evaluated by WCAP-15376. Therefore, the shutdown and transition risk evaluation presented by WCAP-15376 was not representative of the configuration risk likely to be encountered on a plant-specific basis. In response to an NRC staff RAI, the WOG provided risk estimates for this more limiting configuration. The resulting generic ICCDP estimate was found to be within the acceptance guidelines of RG 1.177.

3.4.2.1.3 Tier 2 - Avoidance of Risk-Significant Plant Configurations

As stated in the NRC staff's SE for WCAP-15376, 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 the evaluation performed for WCAP-15376, the following configuration restrictions were identified in the NRC staff's SE for when an RTB and/or a logic cabinet is out of service.

- With an RTB out of service, systems designed to mitigate an ATWS event should be available, including the RCS pressure relief, auxiliary feedwater (AFW) flow, ATWS mitigation system actuation circuitry (AMSAC), and turbine trip. As a result, WCAP-15376 stated that activities that degrade the availability of auxiliary feedwater, RCS pressure relief (pressurizer power-operated relief and safety valves), AMSAC, or turbine trip should not be scheduled when an RTB is out of service.

- Because there is increased dependence on the available reactor trip train when one logic cabinet is removed from service, activities that could degrade other components of the RPS (including master relays, slave relays, and analog channels) should not be scheduled when a logic cabinet is out of service.

- WCAP-15376 also noted that activities involving electrical support systems for the identified equipment should not be scheduled during RTB maintenance.

The licensee also stated in their RAI response that work or testing that removes an SSPS train from service includes the following restrictions.

- No other scheduled work should be scheduled that degrades the availability of AMSAC, AFW, power-operated relief valves (PORVs), RCS code safeties, or turbine trips for ATWS mitigation.

- One complete train of the emergency core cooling system (ECCS) must be available for automatic actuation.

- No work should be scheduled on the other SSPS train or its RPS/ESF inputs.

- No work should be scheduled on electrical and cooling systems for the three systems or functions shown above.

The licensee evaluated concurrent component outage configurations and confirmed the applicability of the Tier 2 restrictions identified by WCAP-15376 and the licensee for Vogtle will revise procedure 00354-C accordingly to address the above restrictions prior to WCAP-5376 implementation. Based on the above, the NRC staff finds the licensee's Tier 2 analysis

supports the implementation of WCAP-15376 at Vogtle and satisfies the second condition of the NRC staff's SE for WCAP-15376.

3.4.2.1.4 Tier 3 - Risk-Informed Configuration Risk Management

The licensee stated that Tier 3 is addressed by a risk-informed on-line risk management process that conforms to the requirements of the Maintenance Rule (10 CFR 50.65), specifically 10 CFR 50.65(a)(4). The risk management process is controlled and implemented through Vogtle maintenance scheduling procedure 00354-C. This procedure provides for the assessment and management of the risk of various plant configurations during power operation as required by the Maintenance Rule, RG 1.177, and RG 1.182, which endorses NUMARC 93-01, Section 11. The licensee stated that the Vogtle CRMP included the following elements.

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

- Provisions for performing an assessment prior to entering the LCO condition for preplanned activities.

- Provisions for performing an assessment after entering the LCO condition for unplanned entry into the LCO condition.

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

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

A review of recent inspection reports that evaluated the licensee's assessment of plant risk, scheduling, and configuration control for selected planned and emergent work activities found them acceptable and monitored in accordance with the requirements of 10 CFR 50.65(a)(4) and plant procedures. Based on the licensee's conformance to the requirements of 10 CFR 50.65, RG 1.182, and the guidelines of RG 1.177, the NRC staff finds the licensee's Tier 3 program adequate to support the implementation of WCAP-15376 and satisfies Condition 2 of the staff's SE to WCAP-15376 with regards to Tier 3.

3.4.3 Implementation and Monitoring Program

RG 1.174 and RG 1.177 also establish the need for an implementation and monitoring program to ensure that extensions to TS CT or STIs do not degrade operational safety over time and that no adverse degradation occurs from changes in the licensing basis due to unanticipated degradation or common cause mechanisms. An implementation and monitoring program is intended to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change.

The licensee stated that they plan to monitor RTS and ESFAS protection system equipment unavailability and component failures to ensure consistency with the WCAP-15376 modeling assumptions. The licensee will document these failures in the site corrective action program.

Should the equipment unavailability or reliability fail to meet established monitoring criteria, an engineering assessment will determine required corrective action.

The licensee also confirmed that the expected drift allowances for protection system instrument channels with extended STIs are bounded by the supporting setpoint uncertainty calculations. As above, should repeated occurrences of abnormal rack drift occur, the plant corrective action program will determine the action required to restore channel performance.

The licensee is developing programs to encompass the above monitoring and has identified this as a regulatory commitment.

3.4 Comparison With Regulatory Guidance

The proposed changes are based on TSTF-411, Revision 1, “Surveillance Test Interval Extensions for Components of the Reactor Protection System” (WCAP-15376), and the analysis performed in WCAP-15376, as approved by the NRC staff including limitations and conditions identified in the NRC staff’s SE for WCAP-15376. As such, the implementation of the topical report at Vogtle is within the RG 1.174 and 1.177 acceptance guidance for ΔCDF, ΔLERF, ICCDP, and ICLERP.

3.5 Staff Findings and Conditions

The NRC staff finds that the licensee has demonstrated the applicability of WCAP-15376 to Vogtle and has met the conditions and limitations as outlined in the NRC staff’s SE. The risk impacts for ΔCDF, ΔLERF, ICCDP, and ICLERP as estimated by WCAP-15376 were found to be within the acceptance guidelines for RG 1.174 and RG 1.177. The plant-specific functional units were shown to be applicable to the topical report 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 topical reports to ensure these configurations are controlled. The licensee’s Tier 3 CRMP and implementation and monitoring plan were found to be consistent with the RG 1.177 CRMP and implementation and monitoring plan guidelines. Therefore, the NRC staff finds that the TS revisions proposed by the licensee are consistent with the extended STIs and CT approved for WCAP-15376 and that they meet the NRC staff’s SE conditions and limitations for WCAP-15376.

The following conditions will be incorporated into work planning procedures by the licensee consistent with WCAP-15376 and the NRC staff’s SE.

> The licensee will implement administrative controls to include the following restrictions when an RTB or logic cabinet is removed from service.

- Activities that could degrade the availability of the AFW, RCS pressure relief (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when an RTB is out of service
- Activities that could degrade other components of the RPS, including master relays or slave relays and activities that cause analog channels to be unavailable should not be scheduled when a logic cabinet is unavailable.

- Activities on electrical systems that support the systems or functions listed in the first two bullets should not be scheduled when an RTB is unavailable.

The licensee stated in their RAI response dated May 5, 2006, that work or testing that removes an SSPS train from service includes the following restrictions:

- No other scheduled work should be scheduled that degrades the availability of AMSAC, AFW, PORVs, RCS code safeties, or turbine trips for ATWS mitigation.

- One complete train of ECCS must be available for automatic actuation.

- No work should be scheduled on the other SSPS train or its RPS/ESF inputs.

- No work should be scheduled on electrical and cooling systems for the three systems or functions shown above.

The staff finds the TS revisions proposed by the licensee to be consistent with the extended STIs and RTB CT approval in WCAP-15376 and has met the staff's SE conditions for the implementation of WCAP-15376. The staff finds that the WCAP-15376 generic analysis is applicable to the licensee's plant. Therefore, on the basis of the above evaluation, the NRC staff concludes that the proposed amendments to extend RPS and ESFAS STIs and RTB CT and bypass times are acceptable.

4.0 REGULATORY COMMITMENT

The licensee identified the following regulatory commitment in their RAI response of May 5, 2006.

- The licensee will implement a program to monitor RTS and ESFAS protection system equipment unavailability and component failure to ensure consistency with WCAP-15376 modeling assumptions

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

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that

may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (70 FR 67751). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

References

1. Letter, W. H. Ruland, NRC, to R. H. Bryan, Westinghouse Owners Group (WOG) , “Acceptance for Referencing of Topical Report (Westinghouse Commercial Atomic Power) WCAP-15376-P, Rev. 0, “Risk-Informed Assessment of the RTS [reactor trip system] and ESFAS [engineered safety features actuation system] Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times (TAC. No. MB0983)”, dated December 20, 2002, ML023540534.

2. Letter, W. D. Beckner, NRC, to A. Peitrangelo, Nuclear Energy Institute, finding the proposed changes in Technical Specification Change Traveler, TSTF-411, Rev.1 “Surveillance Test Interval Extensions for Components of the Reactor Protection System,” to be acceptable, dated August 30, 2002, ML022460347.

3. Letter, C. O. Thomas, NRC, to J. J. Sheppard, WOG, “Acceptance for Referencing of Licensing Topical Report WCAP-10271, “Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation Systems,”“ dated February 21, 1985.

4. Letter, July 24, 1985, H. R. Denton, NRC to L. D. Butterfield, WOG, providing comments on WOG guidelines for preparing submittals requesting NRC approval of RTS TSs, dated July 24, 1985.

5. Letter, C. E. Rossi, NRC, to R. A. Newton, Westinghouse Owners Group, WCAP-10271, Supplement 2 and WCAP-10271, Supplement 2, Revision1, “Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System,” dated February 22, 1989.

6. Letter, C. E. Rossi, NRC, to G. T. Goering, WOG, “Westinghouse Topical Report WCAP-10271, Supplement 2, Revision 1, Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System,” dated April 30, 1990.

Principle Contributors: Cliff Doutt, NRR
Sang Rhow, NRR

Date: September 1, 2006

Vogtle Electric Generating Plant, Units 1 & 2

cc:

Mr. N. J. Stringfellow
Manager, Licensing
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

Mr. T. E. Tynan, General Manager
Vogtle Electric Generating Plant
Southern Nuclear Operating Company, Inc.
7821 River Road
Waynesboro, GA 30830

Mr. Jeffrey T. Gasser
Executive Vice President
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

Mr. Steven M. Jackson
Senior Engineer - Power Supply
Municipal Electric Authority of Georgia
1470 Riveredge Parkway, NW
Atlanta, GA 30328-4684

Mr. Reece McAlister
Executive Secretary
Georgia Public Service Commission
244 Washington St., SW
Atlanta, GA 30334

Mr. Harold Reheis, Director
Department of Natural Resources
205 Butler Street, SE, Suite 1252
Atlanta, GA 30334

Attorney General
Law Department
132 Judicial Building
Atlanta, GA 30334

Mr. Laurence Bergen
Oglethorpe Power Corporation
2100 East Exchange Place
P.O. Box 1349
Tucker, GA 30085-1349

Arthur H. Domby, Esquire
Troutman Sanders
Nations Bank Plaza
600 Peachtree Street, NE
Suite 5200
Atlanta, GA 30308-2216

Resident Inspector
Vogtle Plant
8805 River Road
Waynesboro, GA 30830

Office of the County Commissioner
Burke County Commission
Waynesboro, GA 30830

Mr. D. E. Grissette, Vice President
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Birmingham, AL 35201