December 3, 2003

Mr. J. T. Gasser
Vice President
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNIT 2 RE: ISSUANCE OF AMENDMENT (TAC NO. MC1256)

Dear Mr. Gasser:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 108 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Unit 2 (VEGP-2). The amendment consist of changes to the Technical Specifications (TS) in response to your application dated November 5, 2003.

The amendment would extend the surveillance interval for the Train B Memories Test portion of the ACTUATION LOGIC TEST for: (1) Power Range Block (Switch position 1), (2) Intermediate Range Block (Switch position 2), (3) Source Range Block (Switch positions 3 and 4), (4) Safety Injection (SI) Block, Pressurizer (Switch positions 5 and 6), (5) SI Block, High Steam Pressure Rate (Switch positions 7 and 8), (6) Auto SI Block (Switch position 9), and (7) Feedwater Isolation on P14 or SI (Switch positions 10 and 11). In addition to the functions listed above, the amendment would extend the surveillance interval for the portions of the ACTUATION LOGIC TEST for Feedwater Isolation on P14 or SI that pass through the memories circuits and the Power Range block of the Source Range Trip test for the VEGP-2 Train B Solid State Protection System (SSPS). Because the above-described surveillances will become due multiple times before the end of the current fuel cycle, and the Memories Test Switch is not functioning, you requested an exigent TS change in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.91(a)(6), to extend the surveillance interval of the above-described tests to the next refueling outage at the end of Cycle 10 or the next Unit 2 shutdown to MODE 5, whichever comes first.

The operating history at VEGP has demonstrated the SSPS to be highly reliable. A review of the results of the actuation logic testing has not revealed any logic failures. In accordance with Regulatory Guide 1.177, "An approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," the risk increase associated with the proposed change to the Surveillance Requirements (SR) is very small. In addition, there are measures that can be taken to mitigate a malfunction in the Train B actuation logic should it occur. Operation under the proposed change, as demonstrated by the risk-informed evaluation, discussed in the enclosed Safety Evaluation, presents the safest course of action. Therefore, the NRC staff finds that the proposed changes to SR 3.3.1.5 and SR 3.3.2.2 are acceptable. Southern Nuclear Operating Company has committed to implement the proposed temporary TS changes in accordance with the three-tiered approach described in Section 3.2 of the enclosed Safety Evaluation. The NRC staff concludes that the results and insights of the risk assessment support the proposed temporary surveillance testing interval extensions.

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/***

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-425

Enclosures: 1. Amendment No. 108 to NPF-81
2. Safety Evaluation

cc w/encls: See next page

in accordance with the three-tiered approach described in Section 3.2 of the enclosed Safety Evaluation. The NRC staff concludes that the results and insights of the risk assessment support the proposed temporary surveillance testing interval extensions.

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/***
Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-425

Enclosures: 1. Amendment No. 108 to NPF-81
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

PUBLIC
PDII-1 R/F
RidsNrrDlpmLpdii (EHackett)
RidsNrrDlpmLpdii-1 (JNakoski)
RidsNrrPMFRinaldi
RidsNrrLACHawes
RidsOgcRp
RidsAcrsAcnwMailCenter
G. Hill (2 Hard Copies)
TBoyce (DRIP/IROB)
RidsRgn2MailCenter (BBonser)
SRhow
CSchulten
WBeckner
SPeters
MReinhart
EMarinos
NSaltos

ADAMS Accession No.: ML033380002 *See previous concurrence NRR-058

| OFFICE | PDII-1/PM | PDII-1/LA | OGC* | EEIB/SC* |
|---|---|---|---|---|
| NAME | FRinaldi:sab | CHawes | RHoefling | EMarinos |
| DATE | 11/25/03 | 12/02/03 | 12/02/03 | 11/25/03 |

| OFFICE | SPSB/SC* | IROB/SC* | PDII-1/SC |
|---|---|---|---|
| NAME | MReinhart | TBoyce | JNakoski |
| DATE | 11/26/03 | 12/01/03 | 12/02/03 |

OFFICIAL RECORD COPY
C:\ORPCheckout\FileNET\ML033380002.wpd

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. 108
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 November 5, 2003, 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. 108, 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 as soon as possible following issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

***/RA/***

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification
Changes

Date of Issuance: December 3, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

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

| Remove | Insert |
|---|---|
| 3.3.1-10 | 3.3.1-10 |
| ----------- | 3.3.1-10a |
| ----------- | 3.3.1-10b |
| 3.3.2-7 | 3.3.2-7 |
| ---------- | B 3.3.1-57a |
| ---------- | B 3.3.1-57b |
| B 3.3.2-46 | B 3.3.2-46 |

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

DOCKET NO. 50-425

## 1.0 INTRODUCTION

On October 26, 2003, Southern Nuclear Operating Company, Inc., et al. (the licensee or SNC) encountered problems with a test switch (Memories Test Switch) for the Vogtle Electric Generating Plant, Unit 2 (VEGP-2), while performing Technical Specifications (TS) surveillance tests stated in Surveillance Requirements (SR) 3.3.1.5 and 3.3.2.2. As a result of the malfunction of the Memories Test Switch, these actuation logic surveillance tests could not be completed. The inability to complete these tests affected equipment in the VEGP-2 Train B Solid State Protection System (SSPS) actuation logic. The SSPS performs the decision logic required for reactor trip or engineered safeguard feature actuation. Following this malfunction, the Memories Test Switch was placed in the "Off" position for SSPS Train B of VEGP-2, and all actuation logic is operable with all permissives and blocks in their correct state. The only impact of this malfunction on the plant safety is that the Memories Test Switch is unavailable for use to perform subsequent surveillance tests of the VEGP-2 SSPS Train B per SR 3.3.1.5 and SR 3.3.2.2 (on a 31-day staggered test basis). The testing of the VEGP-2 SSPS Train A actuation logic is not affected by this condition.

The inability to complete all steps of surveillance tests SR 3.3.1.5 and SR 3.3.2.2 on October 26, 2003, has been addressed by a Notice of Enforcement Discretion (NOED) granted by the NRC staff's verbal approval on November 4, 20003, and its letter of November 7, 2003, documenting the NRC staff's verbal approval. This NOED extended the testing interval of the non-completed steps by 28 days to allow time for an exigent TS change to be processed. However, the 31-day staggered tests of SSPS Train B actuation logic will become due three times before the end of the current Unit 2 fuel cycle (planned for Spring 2004).

By letter dated November 5, 2003, SNC proposed an exigent TS license amendment to change SR 3.3.1.5 and SR 3.3.2.2 for the VEGP-2. The proposed changes would extend the surveillance interval for the Memories Test portion of the ACTUATION LOGIC TEST for: (1) Power Range Block (Switch position 1), (2) Intermediate Range Block (Switch position 2), (3) Source Range Block (Switch positions 3 and 4), (4) Safety Injection (SI) Block, Pressurizer (Switch positions 5 and 6), (5) SI Block, High Steam Pressure Rate (Switch positions 7 and 8), (6) Auto SI Block (Switch position 9), and (7) Feedwater Isolation on P14 or SI (Switch positions 10 and 11). In addition to the functions listed above, the licensee has requested an extension of the surveillance interval for the portions of the ACTUATION LOGIC TEST for Feedwater

Isolation on P14 or SI that pass through the memories circuits and the Power Range block of the Source Range Trip test for the VEGP-2 Train B SSPS. Because the above-described surveillances will become due multiple times before the end of the current fuel cycle, and the Memories Test Switch is not functioning, SNC requested an exigent TS change in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.91(a)(6) to extend the surveillance interval of the above-described tests to the next refueling outage at the end of Cycle 10 or the next Unit 2 shutdown to MODE 5, whichever comes first..

## 2.0 REGULATORY EVALUATION

SNC requests changes to the testing requirements of SSPS Train B actuation logic for VEGP-2. The proposed changes involve only the extension of the surveillance interval for tests that cannot be performed when the Memories Test Switch is not functional. These changes affect SR 3.3.1.5 (related to Reactor Trip System instrumentation) and SR 3.3.2.2 (related to Engineered Safety Feature Actuation System (ESFAS) instrumentation) and are summarized below.

### Change to SR 3.3.1.5

The surveillance interval for (a) the Memories Test portion of the ACTUATION LOGIC TEST and (b) the test of the Power Range Block of the Source Range Neutron Flux Trip Block will be extended to the end of the VEGP-2 end-of-cycle 10 refueling outage or the next VEGP-2 shutdown to MODE 5, whichever comes first.

### Change to SR 3.3.2.2

The surveillance interval for (a) the Memories Test portion of the ACTUATION LOGIC TEST and (b) the portions of the ACTUATION LOGIC TEST for Feedwater Isolation on P-14 or SI that pass through the memories circuits will be extended to the end of the Unit 2 end-of-cycle 10 refueling outage or the next Unit 2 shutdown to MODE 5, whichever comes first.

## 3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the proposed changes on each of the functions that cannot be tested when the Memories Test Switch is not functional. The functions that are tested by the use of the Memories Test Switch include several permissives and blocks as well as the feedwater isolation from safety injection or steam generator high-high level, as discussed below. SNC's evaluation of the proposed TS changes utilizes traditional engineering analyses (Deterministic Evaluation) as well as the risk-informed approach (Probabilistic Evaluation).

## 3.1 Deterministic Evaluation

### Power Range Low Setpoint Trip Block

The Nuclear Instrumentation System (NIS) Power Range - Low Setpoint Reactor Trip prevents inadvertent power excursions caused by reactivity accidents such as uncontrolled rod withdrawals, boron dilutions, and excessive heat removal from steam line breaks or excess feedwater additions. The operators manually block this trip during power ascension after reaching the P-10 setpoint. When descending in power, this trip is automatically unblocked when the power goes below the P-10 setpoint.

Memories Test Switch position 1 allows testing of the seal in circuit associated with blocking and unblocking of this trip. Currently, VEGP-2 is operating above the P-10 setpoint, which means that if the seal-in-circuit failed, a reactor trip would be initiated. As of this time, VEGP-2 has not tripped. On the other hand, testing of this seal-in-circuit becomes more important when reducing power below the P-10 setpoint. The licensee must verify that the Power Range Low Setpoint Trip becomes unblocked below this power level to initiate protection from the above mentioned reactivity accidents. If the licensee can not verify the removal of the trip setpoint block, by procedure, they must shut the plant down to Mode 3. Since the VEGP-2 procedures take the plant to a subcritical condition, the staff finds that adequate measures can be taken should a failure of the seal in circuit occur.

### Intermediate Range Block

Similarly, the NIS Intermediate Range High Flux Reactor Trip prevents inadvertent power excursions caused by reactivity accidents such as uncontrolled rod withdrawals, boron dilutions, and excessive heat removal from steam line breaks or excess feedwater additions. The operators manually block this trip during power ascension after reaching the P-10 setpoint. When descending in power, this trip is automatically unblocked when the power goes below the P-10 setpoint.

Memories Test Switch position 2 allows testing of the seal-in-circuit that is associated with blocking and unblocking of this trip. Currently, VEGP-2 is operating above the P-10 setpoint, which means that if the seal-in-circuit failed, a reactor trip would be initiated. As of this time, VEGP-2 has not tripped. On the other hand, the unblocking of the trip becomes very important during plant shutdown, i.e. below P-10. When dropping below P-10, the licensee must verify that the Intermediate Range High Flux Reactor Trip becomes unblocked to initiate protection from the above mentioned reactivity accidents. If the licensee can not verify the removal of the trip setpoint block, by procedure, SNC must shut the plant down to Mode 3. Since the VEGP-2 procedures take the plant to a subcritical condition, the NRC staff finds that adequate measures can be taken should a failure of the seal in circuit occur.

### Source Range Neutron Flux Trip Block

The NIS Source Range High Flux Reactor Trip prevents inadvertent power excursions caused by reactivity accidents such as uncontrolled rod withdrawals, boron dilutions, and excessive heat removal from steam line breaks or excess feedwater additions. The operators manually block this trip during power ascension after reaching the P-6 setpoint.

Memories Test Switch positions 3 and 4 allow testing of the seal-in-circuit associated with blocking and unblocking of this trip. Currently, Vogtle, Unit 2, is operating above the P-6 and P-10 setpoints. If the P-10 circuit were to fail and an operator were to unblock the source range trip function, a reactor trip would be initiated. These two independent failures are highly unlikely. If the plant dropped to operation at a power level between P-6 and P-10 and above the source range trip value, a failure of the seal-in-circuit would cause a reactor trip.

On the other hand, the unblocking of the trip becomes very important during plant shutdown, i.e. below P-6. When dropping below P-6, the licensee must verify that the Source Range High Flux Reactor Trip becomes unblocked to initiate protection from the above mentioned reactivity accidents. If the licensee can not verify the removal of the trip setpoint block, they must manually reset the source range trip function. If this action fails, the licensee would open the reactor trip breakers. Since the VEGP-2 operators can remove power from the reactor trip breakers, thus preventing startup, the staff finds that adequate measures can be taken should a failure of the seal-in-circuit occur.

SI Block, Pressurizer

The Low Pressurizer Pressure Safety Injection signal is an ESFAS function that helps mitigate Loss-of-Coolant Accidents. However, to support plant shutdown, SI can be manually blocked when the pressure goes below the P-11 setpoint. The SI function will be automatically enabled when the pressurizer pressure goes above the P-11 setpoint.

Memories Test Switch positions 5 and 6 allow testing of the seal-in-circuit associated with blocking of the Pressurizer Low Pressure Safety Injection when below P-11. Currently, this function is enabled at VEGP-2. In the case of a failure to manually block this signal because of an equipment malfunction, the licensee can block the SI actuation by placing the bypass test instrumentation (BTI) switches associated with pressurizer low pressure SI in bypass. Since the VEGP-2 operators can block this SI actuation below P-11, the staff finds that adequate measures can be taken should a failure of the seal-in-circuit occur.

SI Block, High Steam Pressure Rate

The Low Steamline Pressure SI Actuation and steamline isolation functions to mitigate the consequences of a steamline break downstream of the isolation valves. To support plant shutdown, this setpoint can be manually blocked when the pressure drops below the P-11 setpoint. Blocking of this function below P-11 automatically enables the High Steam Pressure Rate steamline isolation. Above P-11, on the other hand, the Low Steamline Pressure SI Actuation and steamline isolation functions are automatically enabled.

Memories Test Switch positions 7 and 8 allow testing of the seal in circuit associated with blocking of the SI actuation and steamline isolation. In the case of a failure to manually block this signal because of an equipment malfunction, the licensee can block the SI actuation by placing the BTI switches associated with Low Steamline Pressure in bypass. However, if the seal in circuit does not work, the automatic steamline isolation on steam pressure rate function will be inoperable. To compensate, operators can monitor the pressure rate bistable status lights and manually actuate the steamline isolation. Since the VEGP-2 operators can block the SI actuation below P-11 and can manually isolate the steamline, the staff finds that adequate measures can be taken should a failure of the seal in circuit occur.

Auto SI Block

The Automatic SI block functions to prevent re-actuation of SI as long as a reactor trip P-4 signal exists. The P-4 signal can be cleared by closing the reactor trip breakers. Switch position 9 allows testing of the seal in circuit associated with blocking of the automatic SI after SI has been actuated and after it has been reset. If this seal-in-circuit fails and does not block SI re-actuation as indicated by the Auto SI Blocked permissive lamp in the control room the licensee can take several actions. First, they could clear the SI signal by blocking the pressurizer pressure SI signal and main steamline low pressure SI signal if below the P-11 setpoint using the main control board block switches or by placing the tripped SI bistables in bypass using the BTI switches. Then the licensee could reset the SI signal.

Since the VEGP-2 operators can block the SI signals by alternate means, thus preventing SI re-actuation, the staff finds that adequate measures can be taken should a failure of the seal-in-circuit occur.

Feedwater Isolation on P-14 or SI

The Steam Generator Hi-Hi Level signal or an SI signal will initiate feedwater isolation. Once it is initiated, the feedwater isolation valves and the main and bypass feedwater control valves isolation signal will remain sealed in as long as a reactor trip P-4 signal exists. Isolating the main feedwater helps prevent overcooling of the primary side.

Switch positions 10 and 11 allow testing of the seal in circuit associated with the seal in of the feedwater isolation in the presence of P-4. If a feedwater isolation does not occur or a valve closure is not sealed in, the operators can manually close any open feedwater isolation valve from the control board. The current reactor trip and SI procedures direct operators to verify that feedwater isolation has occurred. If it has not, the procedure then directs the operators to shut the isolation valves. Since the VEGP-2 operators can manually close the feedwater isolation valves, the staff finds that adequate measures can be taken should a failure of the seal-in-circuit occur.

3.2 Probabilistic Evaluation

SNC has submitted risk information in support of the proposed TS changes. Such information considers the impact of the proposed changes on each of the functions that cannot be tested when the Memories Test Switch is not functional. The functions that are tested by the use of the Memories Test Switch have been identified in Section 3.1 of this Safety Evaluation. These functions include several permissives and blocks as well as the feedwater isolation from safety injection or steam generator high-high level.

The risk impact associated with the postponement of the surveillance testing of permissives and blocks was qualitatively assessed to be very small. The assessment was based on (a) the fact that all the permissive and block functions are in their correct state for power operation, (b) the worst consequence of a permissive or block function failure is a reactor trip, and (c) a reactor trip requires an additional failure, such as operator failure to follow procedures. The NRC staff finds that any risk increase associated with such scenarios is very small and well below what is considered acceptable by the guidance provided in Regulatory Guide (RG) 1.174 “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The risk impact associated with the extension of the surveillance interval of the feedwater isolation (from safety injection or steam generator high-high level) function was assessed quantitatively by assuming (conservatively) that the Train B feedwater isolation signal (actuation and seal-in) is completely lost for the duration of the proposed extension of the surveillance interval. This assumption, which was made for the purpose of simplifying the analysis, is very conservative. The real impact of the proposed change is to increase the failure probability of the Train B feedwater isolation function rather than render it unavailable. Since it is assumed that the Train B feedwater isolation function is unavailable, the three-tiered approach documented in RG 1.177, "An approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," can be followed in evaluating the risk information submitted by the licensee (i.e., the risk information is, conservatively, evaluated as a request for a completion time extension instead of a request for a surveillance interval extension). The first tier of the three-tiered approach includes the assessment of the risk impact of the proposed change for comparison to acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174. In addition, the first tier aims at ensuring that the plant risk does not increase unacceptably during the period the equipment is taken out of service. The second tier addresses the need to preclude potentially high risk configurations that could result if equipment other than to that associated with the change are taken out of service simultaneously. The third tier addresses the establishment of an overall configuration risk management program (CRMP) for identifying risk significant configurations resulting from maintenance or other operational activities and taking appropriate compensatory measures to avoid such configurations.

Quality of Risk Assessment

The dominant accident sequences contributing to the risk increase are all initiated by a secondary side break followed by failure to isolate the faulted steam generator (SG) from the intact SGs in order to reduce the cooldown rate associated with the secondary break. The isolation of the faulted SG requires feedwater isolation as well as closure of the main steam isolation valves and termination of auxiliary feedwater flow. The impact of the proposed TS change is modeled in the risk assessment as a degradation of the feedwater isolation function because the Train B feedwater isolation signal is (conservatively) assumed unavailable but the Train A feedwater isolation signal is available (subject, obviously, to normal random failures). However, core damage would not occur as long as the reactor coolant system boundary remains intact (i.e., no loss of coolant) and the high pressure safety injection is successful.

The licensee also considered accident sequences initiated by a SG tube rupture (SGTR) that also require feedwater isolation in order to isolate the faulted SG. However, the risk increase associated with these sequences was not assessed because in the VEGP-2 probabilistic risk assessment (PRA) the automatic feedwater isolation signal is not modeled (only manual isolation of the faulted SG is credited). This shows that there is no significant risk increase associated with SGTR accidents.

The NRC staff has evaluated the quality of the PRA models, major assumptions and data used in the risk assessment by comparing it to applicable findings from the staff's review of the PRA (developed as part of licensee's individual plant evaluation) as well as to findings for similar plants and found them acceptable. Furthermore, the NRC staff has utilized the submitted risk information to investigate how sensitive the risk assessment results and conclusion are to important modeling assumptions and data. These sensitivity studies have indicated that the conclusion of the risk assessment regarding the acceptability of the proposed change would not change even when additional bounding assumptions are made.

Risk Impact of the Proposed Change (Tier 1)

An acceptable approach to risk-informed decision making is to show that the proposed change to the licensing basis meets several key principles (RG 1.174). One of these principles is to show that the proposed change results in an increase in risk, in terms of core damage frequency (CDF) and large early release frequency (LERF), that is small and consistent with the Commission’s Safety Goal Policy Statement. Acceptance guidelines for meeting this principle are presented in RG 1.174. Although the RG 1.174 refers to permanent changes to the licensing basis while the licensee proposes a temporary change to allow a one-time extension, guidance provided in RG 1.174 can be used to show that the proposed change results in an increase in risk that is small and consistent with the Commission’s Safety Goal Policy Statement.

The licensee used its PRA model of the plant to calculate risk increases due to the proposed extension of the surveillance interval of the feedwater isolation (from safety injection or steam generator high-high level) function. Both the incremental conditional core damage probability (ICCDP) and the incremental conditional large early release probability (ICLERP) were assessed. These quantities are a measure of the increase in probability of core damage and large early release, respectively, during an outage assumed to last for the entire duration allowed by the proposed change.

ICCDP: Less than 1E-9

ICLERP: Less than 1E-11

These values are much smaller than the acceptance guidance criteria of 5E-7 for ICCDP and 5E-8 for ICLERP, respectively, outlined in RG 1.177.

Because the proposed TS changes are one-time changes, the increase in CDF is numerically equal to the assessed ICCDP value. Similarly, the increase in LERF is numerically equal to the assessed ICLERP value:

- The mean CDF of VEGP-2 will increase by less than 1E-9/year (during the one-year period that the proposed TS changes will be implemented).

- The mean LERF of VEGP-2 will increase by less than 1E-11/year (during the one-year period that the proposed TS changes will be implemented).

According to the guidelines of RG 1.174, the estimated increases in the mean values of CDF and LERF are of low risk significance.

Avoidance of High Risk Plant Configurations (Tier 2)

The licensee used its PRA and qualitative risk arguments to identify dominant accident sequences contributing to the increase in risk associated with the proposed TS changes. This process did not identify any high risk configurations or conditions that must be avoided during implementation of the proposed TS changes at VEGP-2. However, this process identified measures (listed in pages E2-7 to E2-9 of its submittal) that can be taken to ensure safe plant operation should a malfunction occur. Since such measures are part of required procedures, no additional monitoring or compensatory measures are necessary.

Risk-Informed Configuration Risk Management (Tier 3)

The intent of the risk-informed configuration risk management is to ensure that plant safety is maintained and monitored during an extended outage. A formal commitment to maintain a CRPM is required on the part of a utility prior to implementation of a risk-informed TS whenever such TS is entered and risk-significant components are taken out of service. SNC has programs in place for VEGP-2 to comply with 10 CFR 50.65(a)(4), to assess and manage risk from proposed maintenance activities. These programs can support licensee decisionmaking regarding the appropriate actions to control risk whenever a risk-informed TS is entered.

The licensee has committed to implement the proposed temporary TS changes in accordance with the three-tiered approach described above. The NRC staff concludes that the results and insights of the risk assessment support the proposed temporary surveillance testing interval extensions.

## 4.0 STATEMENT OF EXIGENT CIRCUMSTANCES

The regulations at 10 CFR 50.91 provide special exceptions for the issuance of amendments when the usual 30-day public notice cannot be met. One type of special exception is an exigency. An exigency exists when the staff and the licensee need to act quickly and time does not permit the staff to publish a *Federal Register* notice allowing 30 days for prior public comment, and the staff also determines that the amendment involves no significant hazards consideration. In accordance with 10 CFR 50.91(a)(6)(i)(A), the staff issued a Federal Register notice (68 FR 65092, November 18, 2003) providing notice of an opportunity for hearing and allowing two weeks from the date of the notice for prior public comments and proposed finding of no significant hazards consideration.

The licensee requested that the NRC issue this amendment on an exigent basis because the condition was not created by its failure to take timely action. On October 26, 2003, SNC encountered problems with a test switch at VEGP-2 while performing TS surveillance tests (SR 3.3.1.5 and SR 3.3.2.2). As a result of the malfunction of the Memories Test Switch, these actuation logic surveillance tests could not be completed. VEGP-2 is currently operating under a Notice of Enforcement Discretion granted verbally on November 4, 2003, and in writing on November 7, 2003, that will expire on December 3, 2003.

Since VEGP-2 is scheduled for a refueling outage in the Spring 2004, SNC requested an exigent amendment to allow plant operation until the beginning of its Cycle 10 Refueling Outage or the next VEGP-2 shutdown to MODE 5, whichever comes first. The licensee could not have anticipated the need for a license amendment under these circumstances that would allow for the 30 day comment period. Additionally, the proposed amendment involves no significant hazards as specified in 10 CFR 50.92. Further, the licensee maintains that precluding the possibility of an unnecessary plant transient by implementing this one-time exigent amendment will minimize potential safety consequences and operational risks, which is in the best interests of the overall health and safety of the public.

On the basis of the above discussion, the NRC staff has determined that exigent circumstances exist and that the licensee used its best efforts to make a timely application and did not cause the exigent situation.

## 5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The amendment has been evaluated against the three standards in 10 CFR 50.92(c). In its analysis of the issue of no significant hazards consideration, as required by 10 CFR 50.91(a), the licensee has provided the following:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not physically alter any plant structures, systems or components. The SSPS [Solid State Protection System] at VEGP [Votgle Electric Generating Plant] has a history of high reliability. In addition, similar changes to the surveillance interval for actuation logic testing for Westinghouse SSPS actuation logic has been approved by the NRC with their approval of WCAP-15376 and Technical Specification Task Force (TSTF) 411. Therefore[,] there will not be a significant increase in the probability of an accident previously evaluated. There will not be a significant increase in the consequences of any accident previously evaluated as a result of this Technical Specification amendment because the incremental condition large early release probability is very small in accordance with the criteria of Regulatory Guide 1.177. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change involves an extension of a previously determined acceptable surveillance interval. The proposed change does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures. In addition, compensatory actions will be in place which will offset the very small increase in risk. Therefore, the requested

Technical Specification amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety.

   The extended surveillance interval for the SSPS ACTUATION LOGIC TEST has been shown to have a very small impact on plant risk using the criteria of Regulatory Guides 1.174 and 1.177. In addition, compensatory actions in place will be in place in the case of a failure of the functions listed above. Therefore, the enforcement discretion does not involve a significant reduction in a margin to safety.

The NRC staff has reviewed the licensee's analysis, and based on its revie, has determined that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff finds that the amendmend request involves no significant hazards consideration.

## 6.0 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.

## 7.0 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 (68 FR 65092). 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.

## 8.0 CONCLUSION

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

Principal Contributors: S. Rhow
N. Saltos
S. Peters
F. Rinaldi

Date: December 3, 2003

cc:

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

Mr. Skip Kitchens
General Manager, Vogtle Electric
Generating Plant
Southern Nuclear Operating
Company, Inc.
P. O. Box 1600
Waynesboro, Georgia 30830

Mr. J. D. Woodard
Executive Vice President
Southern Nuclear Operating
Company, Inc.
P. O. Box 1295
Birmingham, Alabama 35201-1295

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

Mr. Reece McAlister
Executive Secretary
Georgia Public Service Commission
244 Washington St., S. W.
Atlanta, Ga. 30334

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

Attorney General
Law Department
132 Judicial Building
Atlanta, Georgia 30334

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

Arthur H. Domby, Esquire
Troutman Sanders
NationsBank Plaza
600 Peachtree Street, NE.
Suite 5200
Atlanta, Georgia 30308-2216

Resident Inspector
Vogtle Plant
8805 River Road
Waynesboro, Georgia 30830

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