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

August 19, 2015

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

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNIT 1 – ISSUANCE OF AMENDMENT FOR RESIDUAL HEAT REMOVAL SYSTEM PUMP MOTOR (TAC NO. MF6323)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 176 to Renewed Facility Operating License NPF-68 for the Vogtle Electric Generating Plant (VEGP), Unit 1, in response to your letters dated June 4, July 22 and July 31, 2015. The amendment revises VEGP Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.5.2, "ECCS Operating", such that with the '1A' Residual Heat Removal (RHR) pump inoperable for a motor replacement, the Completion Time of Condition 3.5.2.A changes from 72 hours to 7 days. This TS change would be in effect only for the '1A' RHR pump for the remainder of Cycle 19.

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

Sincerely,

Bob Martin

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

Docket Nos. 50-424

Enclosures:
1. Amendment No. 176 to NPF-68
2. Safety Evaluation

cc w/encls: Distribution via Listserv

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 176
Renewed License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Renewed Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated June 4, July 22 and July 31, 2015, 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 Renewed Facility Operating License No. NPF-68 is hereby amended to read as follows:

C. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 176, 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, shall be implemented within 30 days from the date of issuance, and is subject to implementation of the protections and compensatory measures described in the license amendment request submittals, pursuant to SNC's Nuclear Management Procedure NMP-0S-010, for this one-time change to the VEGP Technical Specifications.

FOR THE NUCLEAR REGULATORY COMMISSION

RJ Pascarelli

Robert Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

ATTACHMENT TO

LICENSE AMENDMENT NO. 176

RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

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

Remove	Insert
License NPF-68, page 4	License NPF-68, page 4
TSs 3.5.2-1	TSs 3.5.2-1

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 176 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

(4) Deleted

(5) Deleted

(6) Deleted

(7) Deleted

(8) Deleted

(9) Deleted

(10) Mitigation Strategy License Condition

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

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

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

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

------------------------------NOTE------------------------------
In MODE 3, either residual heat removal pump to cold legs injection flow path may be isolated by closing the isolation valve to perform pressure isolation valve testing per SR 3.4.14.1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. <u>AND</u> At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	----------NOTE--------- A one-time only change of the Completion Time to 7 days is permitted for the 1A RHR pump motor replacement during Vogtle Unit 1, Cycle 19. The increased Completion Time is applicable only to the 1A RHR pump. ------------------------ 72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 176 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-68

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

DOCKET NO. 50-424

1.0 INTRODUCTION

By application dated June 4, as supplemented by letters dated July 22 and July 31, 2015, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML15155B593, ML15203B112 and ML15212A940, respectively, Southern Nuclear Operating Company (SNC, the licensee) submitted a license amendment request (LAR) to change the Vogtle Electric Generating Plant, Unit 1 (Vogtle) Technical Specifications (TSs).

The supplements dated July 22 and July 31, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 23, 2015 (80 FR 35983).

The amendment revises VEGP Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.5.2, "ECCS Operating", such that with the '1A' Residual Heat Removal (RHR) pump inoperable for a motor replacement, the Completion Time (CT) of Condition 3.5.2.A changes from 72 hours to 7 days. This TS change would be in effect only for the '1A' RHR pump for the remainder of Cycle 19.

2.0 RISK-INFORMED REVIEW

2.1 Regulatory Evaluation

The regulatory guidance on which the NRC staff based its review are:

- Regulatory Guide (RG) 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 2), describes a risk-informed approach, acceptable to the NRC, for

assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such evaluations.

- RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," (Reference 3), describes an acceptable risk-informed approach specifically for assessing proposed one-time TS changes in CTs. This regulatory guide also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 provides the following three-tiered TS acceptance guidelines specific to one-time only CT changes for evaluating the risk associated with the revised CT:

    1. The licensee has demonstrated that the impact of implementation of the one-time only CT change on plant risk is acceptable (Tier 1) based on:

        - Incremental conditional core damage probability (ICCDP) of less than $1.0x10^{-6}$ and an incremental conditional large early release probability (ICLERP) of less than $1.0x10^{-7}$, or

        - ICCDP of less than $1.0x10^{-5}$ and an ICLERP of less than $1.0x10^{-6}$ with effective compensatory measures implemented to reduce the sources of increased risk.

    2. The licensee has demonstrated that there are appropriate restrictions on dominant risk-significant configurations associated with the change (Tier 2), and

    3. The licensee has implemented a risk-informed plant configuration control program. The licensee has implemented procedures to utilize, maintain, and control such a program (Tier 3).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 4), describes one acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Chapter 19, Section 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," Revision 1, of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 5). Guidance on evaluating PRA technical adequacy is provided in Chapter 19, Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 6). More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed

Decisionmaking: Technical Specifications," (Reference 7), which includes CT changes as part of risk-informed decisionmaking. Chapter 19 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 (CDF) or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement; and
- The impact of the proposed change should be monitored using performance measurement strategies.

Title 10 of the Code of Federal Regulations, Section 50.65(a)(4), requires that: "Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety.

2.2 Technical Evaluation

SNC's LAR for VEGP 1, requested approval for a one-time change to LCO 3.5.2 condition A CT. The NRC staff technical evaluation in this safety evaluation (SE) include (a) evaluations of those portions of the licensee's submittal that pertain to risk-informed evaluation of the proposed TS change, i.e., PRA considerations, (b) evaluation of the proposed change's risk impact consistent with RG 1.177, (c) determination of the technical adequacy of the PRA models consistent with the requirements of RG 1.200, and (d) the other applicable aspects of the PRA related to the proposed TS change.

2.2.1 Detailed Description of the Proposed Change

LCO 3.5.2 requires that two Emergency Core Cooling System (ECCS) trains be OPERABLE in Modes 1, 2, or 3. An ECCS train consists of a centrifugal charging system, a Safety Injection system, and a RHR system. Condition 3.5.2.A requires that, if one of the required trains is inoperable, and that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, then the inoperable train must be restored to OPERABLE status in 72 hours. Otherwise, the reactor must be taken to Mode 3 in 6 hours and to Mode 4 in 12 hours. The proposed change revises the CT of Condition 3.5.2.A from 72 hours to 7 days. This proposed one-time change will apply only to 1A RHR pump and will only be used on VEGP 1, prior to the Cycle 19 shutdown.

The licensee stated in the LAR that replacing the 1A RHR pump motor would require decoupling the motor, removal of electrical and cooling water connections, removal of physical interferences within the pump room, removal of the motor from the room, installing the new motor in the pump room, mounting the motor to the pump casing, making all the electrical and cooling water connections, running the motor uncoupled, coupling the motor to the pump, alignment, and finally performing a coupled run of the motor and pump. These activities are estimated by the licensee to take longer than the 72 hour CT of LCO Condition 3.5.2.A. The licensee further stated that it is preferable to perform the pump motor replacement prior to the shutdown for the refueling outage to reduce worker radiation doses. Prior to the outage, radiation from water in the RHR system has decayed for almost an entire fuel cycle and, therefore, the dose rate within the pump room is reduced. However, if this work is done during the outage, the RHR pump will run immediately prior to the motor replacement and will circulate reactor coolant system water throughout the system, which will increase the dose rate within the pump room. In the licensee's estimate, if the motor replacement is performed during the outage, the total dose to workers is approximately 2 rem. The estimate for performing the work with the unit on-line is estimated to be approximately 1 rem.

2.2.2 Review Methodology

Per SRP Section 19.1 and Section 16.1, the NRC staff reviewed the submittal using the three-tiered approach and two of the five key principles of risk-informed decision making presented in RG 1.174 and RG 1.177 as discussed below.

2.2.2.1 Key Principle 1: Compliance with Current Regulations

This is discussed in Section 3.2.2 of this safety evaluation.

2.2.2.2 Key Principle 2: Evaluation of Defense-in-Depth

This is discussed in Section 3.2.1 of this safety evaluation.

2.2.2.3 Key Principle 3: Evaluation of Safety Margins

This is discussed in Section 3.2.1 of this safety evaluation.

2.2.2.4 Key Principle 4: Change in Risk Consistent with the Commission's Safety Goal Policy Statement

The evaluation presented below addresses the NRC staff's philosophy of risk-informed decision making: that when proposed changes result in a change in CDF or risk, the increase should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. The NRC staff evaluation of key principle 4 for the proposed one-time TS change is described below.

Tier 1: PRA Capability and Insights

The first tier of RG 1.177 evaluates the impact of the proposed change on plant operational risk. The Tier 1 review involves: (1) evaluation of the technical adequacy of the VEGP probabilistic risk assessment (PRA) model and its application to the proposed change, and (2) evaluation of the PRA results and insights based on the licensee's proposed change.

PRA Quality

RG 1.174 states that, "[t]he scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process." The technical adequacy of the PRA must be compatible with the safety implications of the TS change being requested and the role that the PRA plays in justifying that change. That is, the more the potential change in risk or the greater the uncertainty in that risk from the requested TS change, or both, the more rigor that must go into ensuring the technical adequacy of the PRA. This applies to Tier 1, and it also applies to Tier 2 and Tier 3 to the extent that a PRA model is used.

RG 1.200, Revision 2 (Reference 8) describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results such that the PRA can be used in regulatory decisionmaking for light-water reactors. RG 1.200, Revision 2, clarifies the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard to be ASME/ANS RA-Sa-2009, "Addenda to ASME RA-S-2008, Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," (Reference 9). The ASME/ANS PRA standard provides technical supporting requirements in terms of three Capability Categories (CCs). The intent of the delineation of the CCs within the Supporting Requirements (SRs) is generally that the degree of scope and level of detail, the degree of plant specificity, and the degree of realism increase from CC I to CC III. In general, the NRC staff anticipates that current good practice, i.e., CC II of the ASME/ANS standard, is the level of detail that is adequate for the majority of applications.

The VEGP PRA model is composed of an Internal Events PRA (including internal flooding), and a Fire PRA. Because the licensee requests a one-time only CT change, the Vogtle 1 & 2 Configuration Risk Management Program (CRMP) model, an at-power, zero-maintenance model, is used for evaluation of ICCDP and ICLERP, as accepted by RG 1.177. The Vogtle 1 & 2 CRMP model combines the baseline internal events (including internal flooding) PRA model and the baseline internal fire PRA model into a single top model. The CRMP model can be configured to apply seismic CDF and large early release frequency (LERF) factors based on bounding analysis of seismic risk. The licensee stated in the LAR that the process of creating the CRMP model from the baseline PRA models ensures preservation of the CDF and LERF quantitative results, conservation of the quality of the peer-reviewed PRA models, and accommodation of changes in risk due to time-of-year, time-of-cycle and configuration-specific considerations.

Internal Events PRA (Including Internal Flooding)

The licensee stated in the LAR that the Vogtle 1 & 2 at-power internal events PRA model (including flooding) and internal fire PRA model conform to the ASME PRA standards at CC II. The licensee further stated that the Vogtle 1 & 2 internal events (including internal flooding) model has been subjected to several peer reviews and self-assessments including the most recent peer review performed in accordance with the 2007 version of the ASME PRA standard (Reference 10). Details about the peer-review findings and their disposition were provided to the NRC as a part of the Vogtle LAR to transition to a Risk Informed Completion Times (RICTs) by implementing NEI 06-09, Revision 0, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines" (Reference 11) and licensee's responses to Request for Additional Information (RAIs) (Reference 12), for that LAR review. As discussed in References 11 and 12, in May 2009, a full scope peer review of the VEGP internal events PRA was performed against the requirements of ASME PRA Standard RA-Sb-2005 (Reference 13) and RG 1.200, Revision 1. The results identified three "not met" facts and observations (F&Os), as shown in Reference 11, Table E2.2. All other internal events supporting requirements were judged to meet at least CC II. In addition, a gap assessment was performed between ASME PRA Standard ASME RA-Sc-2007 as clarified by RG 1.200, Revision 1, and ASME/ANS PRA Standard ASME/ANS RA-Sa-2009, as clarified by RG 1.200, Revision 2. This assessment was provided in Table E2.3 of the Vogtle LAR for using RICTs (Reference 11) and updated in response to RAIs for that LAR review.

The NRC staff evaluated the PRA quality information provided by the licensee in the LAR for using RICTs, as supplemented by responses to RAIs in Reference 12 and reviewed the peer review F&Os for the Internal Events PRA and the Fire PRA. A summary of the NRC staff's evaluation of the licensee's resolutions of F&Os designated as "Not Met" are provided below. F&O HR-G6-01 is related to not performing a reasonableness check as a part of the human reliability analysis (HRA) Revision 4 update. The ASME standard states that the peer review should include the human failure events (HFEs) and compare those to their final human error probabilities (HEPs) relative to each other to check their reasonableness. The NRC staff finds the licensee's resolution appropriately dispositions the F&O for the application because the licensee performed a reasonableness check for all HRAs to resolve this F&O. F&O QU-D3-01 is related to not providing sufficient evidence of a comparison study of their PRA results against the results from similar plants. VEGP performed a comparison study between their facility and two other facilities as discussed in Reference 11, Table E2.2. The NRC staff finds that the resolution appropriately dispositions this F&O for the application because the licensee performed a comparison study to resolve the F&O.

F&O LE-G5-01 is related to not identifying limitations in the LERF analysis that would impact applications. In the F&O resolution (Reference 11, page E2-9), VEGP showed that the LERF scenarios sufficiently covered the LERF scenarios that would be required by the ASME PRA Standard.

The NRC staff finds that the Vogtle internal PRA has been peer reviewed in accordance with ASME PRA Standard RA-Sb-2005 (Reference 13) and RG 1.200 Revision 1, and that the licensee has performed a gap assessment against the current PRA standard and RG 1.200 Revision 2. The NRC staff finds that the licensee adequately dispostioned the three F&Os to support the internal PRA technical adequacy for the proposed TS change.

Fire PRA

The licensee stated in the LAR that the VEGP Fire PRA model was reviewed in 2012 by the Pressurized Water Reactors Owners Group (PWROG) against the ASME/ANS Standard, RG 1.200, Revision 2, which addresses ASME/ANS RA-Sa-2009, and the Topical Report NEI 07-12 (Reference 14).

The licensee provided findings generated from the Vogtle Fire PRA model peer-review and the licensee's dispositions of those findings in Table E2.5 of RAI responses related to Vogtle LAR for using RICTs (Reference 12). The licensee stated in the same document that all SRs are being met at CC II or better, with the exception of FSS-E3 and PP-B5. As a part of the NRC's review of Vogtle LAR for using RICTs, the NRC staff reviewed those findings and licensee's disposition of each finding to assess the technical adequacy of the Fire PRA for the Vogtle RICT program. In the LAR for the proposed one-time TS change, the licensee also confirmed that open findings were reviewed and none impact the use of the model for evaluations related to RHR.

In response to PRA RAIs related to the Vogtle RICT LAR (Reference 12), SNC identified four Fire PRA methods which may not be consistent with NRC-accepted methods. By letter dated July 22, 2015 and in response to PRA RAI 1, the licensee stated that the CRMP model used to calculate risk results includes methods that are not accepted by the NRC (Reference 15). By letter dated July 31, 2015 (Reference 16), the licensee stated that the model has been refined to incorporate NRC-accepted methods and provided risk results using the refined model.

The NRC staff finds that the Fire PRA has been adequately peer reviewed against the current version of the PRA standard and RG 1.200. Furthermore, the NRC staff finds that, based on the NRC staff's review of Fire PRA F&Os and their dispositions for the Vogtle RICT LAR and the licensee's review of Fire PRA F&Os to evaluate their impact on this application, the licensee has adequately dispositioned the F&Os to support the use of the model for the proposed one-time TS change.

Seismic and Other External Hazards

Attachment 1 to Enclosure 2 of the LAR presents a quantitative analysis to provide a bounding assessment of the seismic risk impact on this application. This analysis estimates bounding seismic CDF, evaluates potential risk increases due to out-of-service equipment and qualitatively evaluates a bounding LERF contribution. The seismic CDF calculated for this application is $2x10^{-5}$/yr, which is based on the 1994 Lawrence Livermore National Laboratory study (Reference 17) and is more conservative than the 1989 Electric Power Research Institute study (Reference 18) and the 2008 United States Geological Survey study Appendix D (Reference 19). This value represents the convolution of the VEGP seismic hazard curve with an assumed limiting plant fragility based on the high confidence of low probability of failure (HCLPF) of 0.3g, as reported in the VEGP Individual Plant Examination of External Events (IPEEE) (Reference 20). A seismic LERF of $2x10^{-7}$/yr is used for this application based on the conditional large early release probability of 0.01 from the VEGP internal events PRA.

SNC addressed other external hazards in the Enclosure 2 of the LAR. The licensee stated that VEGP IPEEE study and its subsequent update assess the vulnerability of the site to other external hazards. The licensee applied preliminary screening criteria, which resulted in further evaluation of seven external hazards using bounding analyses. These external hazards include aircraft impact, extreme winds and tornadoes, external flooding including intense local precipitation, industrial and military facility accidents, pipelines accidents, and turbine-generated missiles. The licensee concluded that the bounding analyses demonstrated that these external hazards do not pose a credible threat to Vogtle 1 and 2.

Regulatory Position 2.3.2 of RG 1.177 states that the scope of the analysis should include all hazard groups (i.e., internal events, internal flood, internal fires, seismic events, high winds, transportation events, and other external hazards) unless it can be shown that the contribution from specific hazard groups does not affect the decision. The NRC staff finds that the licensee followed RG 1.177 by providing a bounding quantitative analysis of seismic impact, performing a qualitative bounding analyses of other external hazards and determining that those hazards do not impact this application.

Sensitivity and Uncertainty Analyses

Regulatory Position 2.3.5 of RG 1.177 states that the risk resulting from TS CT changes is relatively insensitive to uncertainties because uncertainties associated with CT changes tend to similarly affect the base case and the change case. The licensee stated in Enclosure 2 to the LAR that the CT change subjects the plant to a variation in exposure to the same risks and no new initiating events or failure modes are introduced and, therefore, the PRA models are able to accurately predict the change in CDF and LERF for the variation in exposure. The NRC staff finds that the licensee's assessment of sensitivity and uncertainty is consistent with RG 1.177.

PRA Results and Insights

The licensee used Vogtle 1 & 2 CRMP model to provide a quantitative analyses of internal events, internal floods, internal fires, and seismic events to evaluate the impact of the proposed change on plant risk. The licensee demonstrated that the TS CT change has only a small quantitative impact on the plant risk.

To determine ICCDP and ICLERP, the licensee calculated CDF and LERF for a baseline, zero-maintenance case and for a preventive maintenance (PM) case. Regulatory Position 2.3.4 of RG 1.177 allows using a zero-maintenance state instead of nominal expected equipment unavailabilities, if the licensee provides an explanation stating so in the submittal. The licensee stated in the LAR that as the proposed change is a one-time planned event, the specific configuration can be closely controlled and a zero-maintenance model is more appropriate than using an average maintenance model.

The licensee stated in the LAR that the VEGP 1 & 2 CRMP model was configured as follows for the PM case:

- 1A RHR pump failure to start and failure to run events are changed from their nominal values to failed (1.0),
- seismic risk factors are enabled, and

- RHR common cause factors are set to zero.

The licensee's treatment of common cause failure (CCF) is acceptable, as Section A-1.3.2.2 of RG 1.177 states that the term for CCF of the redundant components can be equated to zero for a short-duration PM (i.e., when the component is down for PM and the duration of the PM is much less than the test interval).

The licensee reported the following results by calculating ICCDP and ICLERP that reflect the entire seven day completion time for the 1A RHR pump outage (Reference 16). Implementation of risk management actions were not credited in licensee's evaluation.

ICCDP = 7.27×10^{-6} (Acceptance Guideline: 1×10^{-5} with effective compensatory measures implemented to reduce the sources of increased risk)
ICLERP = 2.08×10^{-7}(Acceptance Guideline: 1×10^{-6} with effective compensatory measures implemented to reduce the sources of increased risk)

The NRC staff finds that the licensee meets appropriate risk measures specific to one-time only CT changes considering the compensatory measures discussed in the following sections of this SE, and is therefore acceptable.

Tier 2: Avoidance of Risk-Significant Plant Configurations

RG 1.177, Tier 2 states, in part, that: "The licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change."

Based on configuration-specific insights provided by the Vogtle 1 & 2 CRMP software tool, the licensee performed analyses to identify risk-significant combinations of equipment out-of-service during the time the 1A RHR pump is out-of-service and identified further compensatory actions and restrictions for entry into PM to avoid high risk equipment out-of-service combinations during that time. Attachments 2-A through 2-E to Enclosure 2 of the LAR identify top 10 CDF and LERF cutsets for both baseline and PM cases. In addition, in Enclosure 2 to the LAR, the licensee provided a list of systems, structures and components (SSCs) that must be functional during the proposed PM. Finally, the licensee provided a list of SSCs that will be protected and a list of fire zones for which maintenance activities involving unavailability of fire protection equipment (detection, suppression or fire barriers), hot work, or introduction of transient combustible materials will be restricted. Protecting the identified SSCs and restricting maintenance activities in the identified fire zones will maintain the functionality of the safety significant SSCs.

The NRC staff finds that the licensee provided adequate analyses of risk-significant configurations while the 1A RHR pump is out-of-service and identified appropriate compensatory actions that can mitigate corresponding increases in risk. Therefore, the NRC staff concludes that the licensee's analysis of risk significant combinations and identification of compensatory actions are consistent with RG 1.177 and provide reasonable assurance that risk-significant plant equipment outage configurations will not occur during the PM.

Tier 3: Risk-Informed Configuration Risk Management

RG 1.177, section 2.3, states, in part, that: "Tier 3 is the establishment of an overall configuration risk management program (CFMP) to ensure that other potentially lower probability, but nonetheless risk-significant, configurations resulting from maintenance and other operational activities are identified and compensated for." RG 1.177, section 2.3, further states that: "The licensee program for compliance with 10 CFR 50.65(a)(4) ensures that the risk impact of out-of-service equipment is appropriately assessed and managed."
The licensee stated in Enclosure 2 to the LAR that SNC has an established configuration risk management program that implements Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.65(a)(4) requirements. The Vogtle 1 & 2 CRMP software tool is used to perform real time calculations of CDF and LERF impacts to internal event (including flooding) and internal fire hazards and to assess and manage the risk associated with planned maintenance activities and unplanned operational events. The licensee further explained that the CRMP software tool provides risk insights that enable development of effective, configuration-specific, risk management actions as required by CRMP procedures. The CRMP software tool also assists in development of risk management actions by providing configuration-specific information, such as important mitigating components to maintain functional and initiating events to avoid. Finally, the licensee stated that SNC procedures and sub-tier instructions and guidelines provide guidance and requirements for creating and maintaining the SNC PRA models, the CRMP model, and the CRMP software tool for implementation of 10 CFR 50.65(a)(4).

Based on the above, the staff finds the licensee's Tier 3 program for complying with 10 CFR 50.65(a)(4) is consistent with the guidance of Section 16.1 of the SRP and RG1.177 and, thus, is acceptable.

Summary

The licensee has demonstrated that the technical adequacy and scope of its PRA models can support the LAR for the proposed one-time completion time change to LCO 3.5.2.A. The risk metrics used to support the LAR are consistent with the RG 1.177. The NRC staff finds that the licensee followed the three-tiered approach outlined in RG 1.177 to evaluate the risk associated with proposed TS CT change and, therefore, the proposed one-time TS change satisfies the fourth key safety principle of RG 1.177.

2.2.2.5 Key Principle 5: Performance Measurement Strategies – Implementation and Monitoring Program

RG 1.174 and RG 1.177 establish the need for an implementation and monitoring program to ensure that no adverse safety degradation occurs because of the changes to the TS. 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.

RG 1.177 states that the licensee is to use a three-tiered approach in implementing the proposed TS CT changes. Application of the three-tiered approach is in keeping with the fundamental principle that the proposed change is consistent with the defense-in-depth philosophy. Application of the three-tiered approach provides assurance that defense-in-depth will not be significantly impacted by the proposed change. Furthermore, RG 1.177 states that,

to ensure that extension of a TS CT does not degrade operational safety over time, the licensee should ensure, as part of its Maintenance Rule program (10 CFR 50.65), that when equipment does not meet its performance criteria, the evaluation required under the Maintenance Rule includes prior related TS changes in its scope.

The licensee provided a brief evaluation of the proposed TS change against the three tiered approach in Section 3.3.1 of Enclosure 1 to the LAR. In addition, in Section 3.3.2 of Enclosure 1 to the LAR, the licensee stated that, as the proposed change is a one-time change, the additional out-of-service time for the 1A RHR pump is not expected to adversely affect the performance of this pump or of the ECCS. Nevertheless, the licensee confirmed that the RHR pumps are monitored under the VEGP Maintenance Rule Program. If the pre-established reliability or availability performance criteria for the RHR pumps are exceeded, they are evaluated for 10 CFR 50.65(a)(1) actions, which requires increased management attention and goal setting in order to restore their performance to an acceptable level.

The NRC staff concludes that the implementation and monitoring program for the proposed TS change described by the licensee satisfies the fifth key safety principle of RG 1.177.

2.2.3 Summary

The NRC staff finds that the risk impact of licensee's request for a one-time extension of the CT of Condition 3.5.2.A from 72 hours to seven days, as estimated by ICCDP and ICLERP, is consistent with the acceptance guidelines specified in RG 1.177 and the staff guidance outlined in Sections 19.1 and 16.1 of NUREG-0800. The licensee's methodology for assessing the risk impact is accomplished using PRA models of sufficient scope and technical adequacy based on a review of the model consistent with the guidance of RG 1.200, Revision 2. For external hazards which do not have PRA models, the licensee used bounding analyses. The NRC staff finds that the licensee has followed the three-tiered approach and performance monitoring programs outlined in RG 1.177.

3.0 REACTOR SYSTEMS

3.1 Regulatory Evaluation

3.1.1 System Description

The RHR system for VEGP-1 functions as follows: (1) Removes heat energy from the reactor core and the reactor coolant system (RCS) during plant cooldown and refueling operations; (2) Transfers refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations; and (3) The RHR system is a part of the safety injection (SI) system for emergency reactor core cooling and long-term recovery. The RHR system is thus a subsystem of the ECCS, covered by LCO 3.5.2.

The function of the ECCS is to provide core cooling and negative reactivity to ensure that the reactor core is protected after any of the following accidents: (1) Loss of coolant accident (LOCA), coolant leakage greater than the capability of the normal charging system; (2) Rod

ejection accident; (3) Loss of secondary cooling accident, including uncontrolled steam release or loss of feedwater; and (4) Steam generator tube rupture (SGTR).

To satisfy LCO 3.5.2, two redundant 100% capacity ECCS trains are required to ensure that sufficient ECCS is available, assuming a single failure affecting either train. One ECCS train consists of a centrifugal charging system, an SI system, and an RHR system. Furthermore, the ECCS systems at VEGP1 are interconnected such that the operator is able to use components from opposite trains to achieve the required 100% flow to the core.

3.1.2 Proposed Change

LCO 3.5.2 requires two ECCS trains to be OPERABLE for MODES 1, 2, and 3. Condition A states that if one or more trains are inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available, than the required action is to restore the train(s) to an OPERABLE status with a completion time of 72 hours.

The proposed Note would be added to the completion time stating the following:

> A one-time only change of the Completion Time to 7 days is permitted for the 1A RHR pump motor replacement during Vogtle 1, Cycle 19. The increased Completion Time is applicable only to the 1A RHR pump.

3.1.3 Applicable Regulatory Requirements

Regulatory Guide 1.177, "An Approach for Plant Specific Risk Informed Decision Making: Technical Specifications," to assess the nature and impact of proposed TS changes by considering engineering issues and applying risk insights.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.36(c)(3)(ii) states that a technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:

> Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3.2 Technical Evaluation

3.2.1 Summary of Technical Information Provided by Licensee

The licensee states in the application that while the 1A RHR pump is out of service, the other RHR system, 1B, will remain OPERABLE. In addition to the 1B RHR system remaining OPERABLE, both high head systems and both SI systems will be OPERABLE. Should an event occur that requires the initiation of ECCS, the ECCS will be capable of performing its safety function of providing adequate core cooling and negative reactivity to protect the reactor core, assuming no additional failures. This is inherent in TS Condition 3.5.2.A such that a combination of equipment must be maintained OPERABLE such that 100% of the required

ECCS flow remains available. If the required 100% flow is not available, the plant must be taken into MODE 3 within 6 hours.

For the extension of CT to 7 days, defense-in-depth is maintained, in part, for the following reasons:

- Prior to removing the RHR pump from service for the purpose of the motor replacement, any assumptions made (e.g., plant equipment configurations) for the risk informed analysis will be verified as necessary to ensure that the conclusions of the risk informed analysis remain valid;

- Every effort will be made to ensure that the 1A RHR pump is not removed from service for the motor replacement during a period of time of impending inclement weather;

- Should an event occur with the 1A RHR pump out of service and an accompanying failure of either a centrifugal charging pump or an SI pump, 100% ECCS flow could still be delivered to the reactor, as a result of the interconnected ECCS systems.

The proposed TS change is consistent with the principle that sufficient safety margins are maintained based on the following:

Codes and standards (e.g., American Society of Mechanical Engineers (ASME), Institute of Electrical and Electronic Engineers (IEEE) or alternatives approved for use by the NRC) are met. The proposed change is not in conflict with approved codes and standards relevant to the RHR system.

The ECCS has sufficient capacity to function for the LOCA, Rod Ejection Accident, loss of secondary cooling and the steam generator tube rupture (SGTR) event. Assuming no additional failures, the Final Safety Analysis Report (FSAR) acceptance criteria for these events will be met should such an event occur during the time that the 1A RHR pump is out of service.

3.2.2 Summary of NRC Staff Review

The licensee performed an evaluation in accordance with RG 1.177, (Reference 3). The TS revision represents a risk informed licensing change that increases the required completion time for TS 3.5.2.A. The NRC staff evaluated the licensee's proposed change with respect to RG 1.177, Element 2, "Engineering Evaluation," specifically Regulatory Positions 2.1, "Compliance with Current Regulations," 2.2, "Traditional Engineering Considerations," and 2.3, "Evaluation of Risk Impact."

Regulatory Position 2.1 – Compliance with Current Regulations

LCO 3.5.2 assures that the analytic assumptions remain conservative based on requirements for equipment operability pertaining to the ECCS system. As required by LCO 3.5.2, both trains of the ECCS must be operable. The implementation of the proposed Note of the 7-day CT does not change this requirement for LCO 3.5.2. Based on the considerations discussed above, the

NRC staff determined that the implementation of the proposed Note adequately meets the intent of Regulatory Position 2.1 of RG 1.177, which references 10 CFR 50.36, by not changing the requirements of LCO 3.5.2. The staff concludes that the implementation of the proposed Note described by the licensee satisfies the first key safety principle of RG 1.177.

Regulatory Position 2.2 – Traditional Engineering Considerations

The licensee states in the application that defense-in-depth is maintained for the reasons discussed in Section 3.1, Summary of Technical Information Provided by Licensee, of this safety evaluation. Defense-in-depth is maintained by the risk informed analysis identifying compensatory actions that will be put into place to both decrease the likelihood of failure of a second ECCS system and an initiating event during the time the 1A RHR pump is out of service. The licensee states in the application that safety margins are maintained for the reasons discussed in Section 3.2.1, "Summary of Technical Information Provided by Licensee," of this safety evaluation. The maintenance of sufficient safety margins is demonstrated by the compliance with approved codes and standards relevant to the RHR system and Final Safety Analysis Report (FSAR) acceptance criteria relevant to the RHR system. The staff concludes that the approved codes and standards relevant to the RHR system and associated FSAR sections satisfies the third key safety principle of RG 1.177.

RG 1.177 identifies a three tier approach for the evaluation and implementation of risk informed TS changes. Tier 2, Avoidance of Risk-Significant Plant Configurations, states that the licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out of service consistent with the proposed TS change.

Plant configuration

SNC evaluated the plant configuration for the one-time 1A RHR 7-day CT and determined the following SSCs for Vogtle 1 must remain functional:

- Train 'B' RHR;
- Auxiliary Component Cooling Water (ACCW) System;
- Train 'B' Nuclear Service Cooling Water (NSCW) System;
- Train 'B' Safety Injection System; and
- Train 'A', 'B', and 'C' Auxiliary Feedwater (AFW) System.

Protection of SSCs

The following SSCs will be protected in accordance with the SNC equipment protection program:

- SSCs for VEGP1 listed to remain operable/functional prior to entry into the one-time CT;
- 4160VAC [alternating current voltage] Electrical Bus 1BA03;
- 480VAC Switchgear 1BB06 and 1BB16;
- 480VAC Motor Control centers 1BBB and 1BBD;
- 125VDC Bus 1BD1;

- 125VDC Panel 1BY2B;
- AC Inverter 1BD1I12; and
- Fire pumps.
- In addition, for fire zone maintenance activities involving unavailability of fire protection equipment, hot work or the introduction of transient combustible materials will be restricted.

The staff concludes that the plant configuration and protection of SSCs satisfies the second key safety principle of RG 1.177.

The licensee considered Regulatory Positions 2.2.1, Defense in Depth, and 2.2.2, Safety Margins, of RG 1.177. Based on the considerations discussed above, the NRC staff determined that the proper compensatory measures are in place to both decrease the likelihood of failure of a second ECCS system by the plant configuration measures noted above and to decrease the likelihood of an initiating event during the time the 1A RHR pump is out of service by the Protection of SSCs measures listed above.

Request for Additional Information – Compensatory Measures

In a letter dated July 22, 2015, the licensee responded to the NRC staff's RAI regarding the Vogtle 1 compensatory measures for the 1A RHR pump motor replacement with the proposed CT of 7 days as opposed to 72 hours.

The NRC staff requested that the licensee provide the steps taken to ensure the 1A RHR pump is not removed from service for the motor replacement during a period of time of impending inclement weather. The licensee responded that in accordance with SNC procedures, the plant will assess the potential for severe weather conditions and the impact on the proposed schedule for out-of-service activities. This assessment will take place prior to the removal of the pump from service, with the potential to reschedule the RHR pump being out of service based on established compensatory measures. In the case of unanticipated severe weather during the out-of-service duration, efforts would be taken to ensure other equipment or components important to safety remain in-service. Based on the considerations discussed above, the NRC staff determined that the steps taken by the licensee to ensure that the 1A RHR pump is not removed from service during a period of impending inclement weather are acceptable via pre-established SNC procedures and implemented compensatory measures.

The NRC staff requested that the licensee provide the steps taken to ensure that the 1B RHR pump will remain OPERABLE if the 1A RHR pump is out of service. The licensee responded that the 1B RHR pump will remain OPERABLE in accordance with a site procedure. The 1B RHR pump will become "Protected Equipment" during the time the 1A RHR pump is removed from service. Per the site procedure, the following actions will be taken to ensure 1B RHR pump operability: (1) Work will be restricted on the 1B RHR pump and its associated breakers and switchgear; (2) Signage will be put up to increase the sensitivity of personnel in the areas of these components and point out the additional requirements that have been put in place to maintain the 1B RHR pump OPERABLE; (3) The protected status of the 1B RHR pump will also be included on the daily status reports and will be included in crew briefings; and (4) Permission from the Shift Manager will be required to work on or near the protected equipment, which will

only be authorized if the work is required to restore a Key Safety Function on the protected equipment. Based on the considerations discussed above, the NRC staff determined that the steps taken to ensure that the 1B RHR pump will remain OPERABLE if the 1A RHR pump is out of service are acceptable via pre-established site procedures that list the required actions to ensure 1B RHR pump operability.

3.3 Summary

The NRC staff evaluated the licensee's proposed revisions to TS 3.5.2, "ECCS Operating." Based on the topics evaluated in this safety evaluation section 3.0, the NRC staff concludes that the proposed revision to TS 3.5.2 is acceptable from a reactor systems perspective.

4.0 PRA OPERATIONS AND HUMAN FACTORS

4.1 Regulatory Evaluation

The regulatory requirements and guidance which the NRC staff considered in its review of PRA operations and human factors aspects of this LAR are:

a. 10 CFR 50.120, "Training and qualification of nuclear power plant personnel"

b. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition", specifically: Chapter 13.5.2.1, Rev. 2, "Operating and Emergency Operating Procedures," and Chapter 18, Rev. 2, "Human Factors Engineering."

c NUREG-1764, Rev. 1, "Guidance for the Review of Changes to Human Actions"

d. NUREG-0700, Rev. 2, "Human-System Interface Design Review Guidelines"

e. NUREG-0711, Rev. 3, "Human Factors Engineering Program Review Model"

f. 10 CFR Part 50, Appendix A, GDC 19 "Control room,"

4.2 Technical Evaluation

4.2.1 Description of Operator Action(s) and Assessed Safety Significance

In its LAR the licensee proposes to implement the following compensatory actions consistent with controlled procedure NMP-OS-010 to restrict dominant risk-significant configurations associated with the change and to ensure 1B RHR pump operability:

- Work will be restricted on the 1B RHR pump and its associated breakers and switchgear.

- Signage will be put up to increase the sensitivity of personnel in the areas of these components and point out the additional requirements that have been put in place to maintain the 1B pump operable.
- The protected status of the 1B RHR pump will also be included on daily status reports and will be included in crew briefings.
- Permission from the Shift Manager will be required to work on or near the protected equipment, which will only be authorized if the work is required to restore a Key Safety Function on the protected equipment.

In accordance with the generic risk categories established in Appendix A to NUREG-1764, these compensatory actions are considered "of low risk-importance" due to the fact that they are controlled by the site procedure NMP OS-010 "Protected Train/Division and Protected Equipment Program". It is highly unlikely that operators would directly violate a controlled procedure that is the subject of daily briefings. Because of the low potential risk, the NRC staff performed a "Level Three" review, i.e., the least stringent of the graded reviews possible under the guidance of NUREG-1764.

4.2.2 Staffing

Normal staffing and qualification are not affected by the proposed LAR. The NRC staff concludes that there are no new or additional qualifications required to recognize and avoid the prohibited actions and configurations.

4.2.3 Human-System Interface Design

Human-System Interface (HSI) design, including the design of the Safety Parameter Display System (SPDS) will not be affected by the proposed LAR. Signage will be added to alert personnel to the protected equipment status of the 1B RHR Pump. The NRC staff finds this to be acceptable.

4.2.4 Procedure Design

No new or revised procedures are necessary to support the proposed LAR. Procedure NMP OS-010 "Protected Train/Division and Protected Equipment Program" which defines the operator compensatory actions is currently in effect. To implement these compensatory actions, operations will issue a standing order in accordance with SNC procedure NMP-OS-007-003, "Plant Operating Orders", prior to the 1A RHR pump being removed from service. The NRC staff finds this approach to controlling operator actions consistent with NUREG-0711 and therefore, acceptable.

4.2.5 Training Program and Simulator Design

SNC staff have been trained to use applicable procedures and the Vogtle 1 & 2 CRMP software tool to assess and manage the following:

- risk increases associated with planned maintenance activities prior to performance of the activities, and
- risk increases associated with unplanned operational events.

Additionally, daily briefings will be held to remind personnel of the protected status of the 1B RHR pump. These actions will remind personnel of:

- important mitigating components to maintain functional,
- initiating events to avoid,
- important fire zones containing important mitigating components, and
- important fire zones in which fire can generate a risk significant initiating event.

Changes to the simulator are not necessary since neither the plant design nor the functional requirements of LCO 3.5.2 are being changed.

The NRC staff finds this approach to training to be consistent with NUREG-0711 and 10 CFR 50.120, "Training and qualification of nuclear power plant personnel," and acceptable to support this proposed LAR.

4.2.6 Human Factors Verification and Validation

Because existing, approved procedures will be used and because no operator actions were added or changed, verification and validation are not necessary. This is consistent with the guidance of NUREG-0711 and is acceptable to the NRC staff.

4.2.7 Summary

Based on the statements provided by SNC, as summarized in sections 4.2.1 through 4.2.6 above, i.e., that appropriate administrative controls will be applied using approved, controlled procedures, that there are appropriate restrictions on dominant risk-significant configurations associated with the change by using a risk-informed plant configuration control program, and that trained, qualified operators will perform the compensatory actions using the existing, familiar interfaces, the NRC staff concludes that the proposed LAR is acceptable from the human performance point of view.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

As stated in the *Federal Register* Notice published on June 23, 2015 (80 FR 35978), the amendment may be issued before the expiration of the 60 day hearing opportunity period if the Commission makes a final determination that the amendment involves no significant hazards consideration. The NRC's regulations in 10 CFR 50.92 state that the NRC may make a final determination that a license amendment involves no significant hazards consideration 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; or (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. As required by 10 CFR 50.91(a), the licensee, in its application dated June 4, 2015, provided its analysis of the issue of no significant hazards consideration, using the standards in 10 CFR 50.92. The NRC staff's evaluation of the issue of no significant hazards consideration is presented below.

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

   Response: No

   The emergency core cooling systems (ECCS), including the Residual Heat Removal system, are designed for the mitigation of design basis accidents or transients, such as a Loss of Coolant Accident (LOCA). They are not designed, nor do they serve, for the prevention of those events. Consequently, the proposed amendment does not increase the probability of a previously evaluated accident occurring.

   Should an accident occur during the period of time that the RHR pump is out of service, the remaining ECCS components would serve to provide the minimum amount of flow assumed in the accident analyses. Even assuming failure of a charging pump or an SI system on either of the trains, sufficient ECCS flow would still be provided to the reactor vessel to mitigate the consequences of the event. Furthermore, a risk-informed analysis performed in support of this amendment request demonstrates that the consequences of an accident are not significantly increased. As such, the proposed change does not involve a significant increase in the probability or consequences of a previously evaluated accident.

   Also, appropriate compensatory measures will be implemented during the time of the extended Completion Time for the RHR pumps. These actions are intended to decrease the chances of an initiating event occurring during the time of the extended CT and also to minimize the chances of losing any ECCS components.

   For the above reasons, the proposed changes will not result in 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 accident from any accident previously evaluated?

   Response: No.

   Replacement of the 1A RHR pump motor for the extended Completion Time period does not introduce any new or unanalyzed modes of operation. The replacement of the pump motor does not involve any unanalyzed modifications to the design or operational limits of the RHR system. Therefore, no new failure modes or accident precursors are created due to the motor replacement during the extended Completion Time. For the reasons noted above, the proposed change will not create the possibility of a new or different accident from any previously evaluated.

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

    Response: No.

    The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment. The performance of these fission product barriers will not be significantly affected by the proposed change. The risk implications of this amendment request were evaluated and found to be acceptable.

    During the extended Completion Time for the 1A RHR pump, the ECCS will remain capable of providing adequate flow to the reactor vessel to mitigate the consequences of a design basis event such as LOCA. Also, compensatory actions will be put in place to minimize the probability of an initiating event during the extended CT period as well as to minimize the chances of a loss of one of the remaining ECCSs. A risk informed analysis has also been performed which shows that the incremental plant risk has increased by an acceptable amount.

    For the reasons noted above, there is no significant reduction in a margin of safety.

Based on the above evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Georgia was notified on June 25, 2015, of the proposed issuance of the amendments. The State officials 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. 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 made a final determination that no significant hazards consideration is involved for the proposed amendment as discussed above in Safety Evaluation Section 5.0. Accordingly, the amendment meets 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

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2. Regulatory Guide 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 31, 2011. (ADAMS Accession No. ML100910006)

3. Regulatory Guide 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," May 31, 2011. (ADAMS Accession No. ML100910008)

4. Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014)

5. NUREG-0800, Standard Review Plan 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," November 2002

6. NUREG-0800, Standard Review Plan Section 19.1, Revision 2, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," June 2007. (ADAMS Accession No. ML071700657)

7. NUREG-0800, Standard Review Plan Section 16.1, Revision 1, "Risk-Informed Decision Making: Technical Specifications," March 2007. (ADAMS Accession No. ML070380228)

8. Regulatory Guide 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014)

9. ASME/ANS PRA Standard ASME/ANS RA-Sa-2009, Addenda to ASME RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications"

10. ASME PRA Standard ASME RA-Sc-2007, Addenda to ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"

11. Ajluni, Mark J., SNC, to NRC, "License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0, 'Risk-Informed Technical Specifications Initiative 4b,' Risk-Managed Technical Specifications (RMTS) Guidelines," September 13, 2012 (ADAMS Accession No. ML12258A055)

12. Pierce, C.R., SNC, to NRC, "Response to Request for Additional Information on Plant Vogtle License Amendment Request to Revise Technical Specifications to Implement NEI 06-09, Revision 0, 'Risk-Informed Technical Specifications Initiative 4b,' Risk-Managed Technical Specifications (RMTS) Guidelines," July 17, 2014 (ADAMS Accession No. ML14198A574)

13. ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum B to ASME RA-S-2002, ASME, New York, New York, December 30, 2005

14. NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Draft Version H, Revision 0, Nuclear Energy Institute, Washington, DC, November 2008

15. Pierce, C.R., SNC, to NRC, "Vogtle Electric Generating Plant- Unit 1 Response to Request for Additional Information on License Amendment Request for Residual Heat Removal Pump Motor," July 22, 2015 (ADAMS Accession No. ML15203B112)

16. Pierce, C.R., SNC, to NRC, "Vogtle Electric Generating Plant- Unit 1 Supplement Response to Request for Additional Information on License Amendment Request for Residual Heat Removal Pump Motor," July 31, 2015 (ADAMS Accession No. ML15212A940)

17. A.1-4. NUREG-1488, 1994, "Revised Livermore Seismic Hazard Estimates for Sixty-Nine Nuclear Power Plant Sites East of the Rocky Mountains," U.S. Nuclear Regulatory Commission, Washington, D.C.

18. A.1-3. EPRI NP-6395-D, 1989, "Probabilistic Seismic Hazard Evaluation at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue," Electric Power Research Institute, Palo Alto, CA

19. Generic Issue 199 (GI-199), "Implications of Updated Probabilistic Seismic Hazard Estimates In Central And Eastern United States On Existing Plants, Safety/Risk Assessment," August 2010 (ADAMS Accession No. ML100270639 and ML100270756)

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August 19, 2015

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SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNIT 1 – ISSUANCE OF AMENDMENT FOR RESIDUAL HEAT REMOVAL SYSTEM PUMP MOTOR (TAC NO. MF6323)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 176 to Renewed Facility Operating License NPF-68 for the Vogtle Electric Generating Plant (VEGP), Unit 1, in response to your letters dated June 4, July 22 and July 31, 2015. The amendment revises VEGP Technical Specifications (TS) Limiting Condition for Operation (LCO) 3.5.2, "ECCS Operating", such that with the '1A' Residual Heat Removal (RHR) pump inoperable for a motor replacement, the Completion Time of Condition 3.5.2.A changes from 72 hours to 7 days. This TS change would be in effect only for the '1A' RHR pump for the remainder of Cycle 19.

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

Sincerely,

/RA/

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

Docket Nos. 50-424

Enclosures:
1. Amendment No. 176 to NPF-68
2. Safety Evaluation

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