Southern Nuclear

Cheryl A. Gayheart
Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5316 tel
205 992 7795 fax

cagayhea@southernco.com

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ATTN: Document Control Desk
Washington, DC 20555-0001

Edwin I. Hatch Nuclear Plant - Units 1 & 2
Revision 37 to the Updated Final Safety Analysis Report, Fire Hazard Analysis Changes, Technical Specification Bases Changes, Technical Requirements Manual Changes, License Renewal 10 CFR 54.37(b) Changes, 10 CFR 50.59 Summary Report, and Revised NRC Commitments Report

Ladies and Gentlemen:

In accordance with 10 CFR 50.4(b) and 50.71(e), Southern Nuclear Operating Company (SNC) hereby submits Revision 37 to the Edwin I. Hatch Nuclear Plant Units 1 and 2 (HNP) Updated Final Safety Analysis Report (UFSAR). The revised HNP Units 1 and 2 UFSAR pages, indicated as Revision 37, reflect changes through August 31, 2019.

The HNP Units 1 and 2 Technical Specifications, Section 5.5.11, "Technical Specifications (TS) Bases Control Program," provides for changes to the Bases without prior NRC approval. In addition, TS Section 5.5.11 requires that Bases changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Pursuant to TS 5.5.11, SNC hereby submits a complete copy of the HNP TS Bases. The revised HNP TS Bases pages, indicated as Revision 104 for Unit 1 and Revision 116 for Unit 2, reflect changes to the TS Bases through September 26, 2019.

In accordance with Regulatory Issue Summary (RIS) 2001-05, "Guidance on Submitting Documents to the NRC by Electronic Information Exchange or on CD-ROM," all of the current pages of the HNP UFSAR, the HNP UFSAR reference drawings, the TS Bases, the Technical Requirements Manual (TRM), and the Fire Hazard Analysis (FHA) are being submitted on CD-ROM in portable document format (PDF). The revised HNP TRM pages, indicated as Revision 115 for Unit 1 and Revision 120 for Unit 2, reflect changes to the TRM through August 31, 2019. The revised HNP FHA, indicated as Revision 37, also reflects changes through August 31, 2019.

In accordance with 10 CFR 50.59(d)(2), SNC hereby submits the 10 CFR 50.59 Summary Report containing a brief description of any changes, tests, or experiments, including a summary of the safety evaluation of each.

In accordance with NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, SNC reviewed its Commitment Database and identified no commitment changes for the applicable reporting period (July 1, 2018 to August 31, 2019).

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SNC conducted a review of HNP plant changes for 10 CFR 54.37(b) applicability and identified no components that were determined to meet the criteria for newly identified components as clarified by RIS 2007-16, Revision 1, "Implementation of the Requirements of 10 CFR 54.37(b) for Holders of Renewed Licenses."

Enclosure 1 provides a table of contents with associated file names for the set of two CD-ROMs (Enclosure 2). Enclosure 3 provides the 10 CFR 50.59 Summary Report.

This letter contains no NRC commitments. If you have any questions, please contact Jamie Coleman at (205) 992-6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 26th day of September 2019.

Respectfully submitted,

Cheryl Gayheart
Regulatory Affairs Director

CAG/TLE/scm

Enclosures:
1. CD-ROM Table of Contents
2. CD-ROMs (2 discs)
3. 10 CFR 50.59 Summary Report

cc: Regional Administrator, Region II (w/o enclosures)
Senior NRR Project Manager – Hatch (w/o enclosures)
Senior Resident Inspector – Hatch (w/o enclosures)
INPO Emergency Management Manager (Enclosure 2, CD ROMs, only)
RType: CHA02.004

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Enclosure 1
CD-ROM Table of Contents

SEQ	FILENAME CONTENT	EXTENSION
	DISC 1	
	NRC File Nomenclature	.doc
001	HATCH FSAR_U1 UNIT 1 Active Page List Table of Contents Chapters 1 thru 14 Appendices A thru K, Supplement Ka, M, N & R	.pdf
002	HATCH FSAR_U2_APL, TOC, CH1 THRU CH4 UNIT2 Active Page List Table of Contents Chapters 1 thru 4	.pdf
003	HATCH FSAR_U2_CH5 THRU CH7 UNIT 2 Chapters 5 thru 7	.pdf
004	HATCH FSAR_U2_CH8 THRU CH 18, APP A UNIT 2 Chapters 8 thru 18 Appendix A	.pdf
005	HATCH BASES Units 1 and 2 Technical Specifications Bases	.pdf
006	HATCH TRM UNIT 1 PT 1	.pdf
007	HATCH TRM UNIT 1 PT 2	.pdf
008	HATCH TRM UNIT 2	.pdf
009	HNP FHA	.pdf

SEQ	FILENAME CONTENT	EXTENSION
	DISC 2	
010	HATCH FSAR REF DWGS PT 1 A-21603 – H-11606	.pdf
011	HATCH FSAR REF DWGS PT 2 H-11607 – H-16002	.pdf
012	HATCH FSAR REF DWGS PT 3 H-16003 – H-16174	.pdf
013	HATCH FSAR REF DWGS PT 4 H-16176 – H-16339	.pdf
014	HATCH FSAR REF DWGS PT 5 H-16512 – H-19941	.pdf
015	HATCH FSAR REF DWGS PT 6 H-19942 – H-21114	.pdf
016	HATCH FSAR REF DWGS PT 7 H-22250 – H-24748	.pdf
017	HATCH FSAR REF DWGS PT 8 H-24749 – H-26036	.pdf
018	HATCH FSAR REF DWGS PT 9 H-26037 – H-26102	.pdf
019	HATCH FSAR REF DWGS PT 10 H-26103 – S-15290	.pdf
020	HATCH FSAR REF DWGS PT 11 S-15304 – S-55894	.pdf
021	HATCH FSAR REF DWGS PT 12 S-53448 – S-56429 Part 1	.pdf
022	HATCH FSAR REF DWGS PT 13 S-56429 Part 2 – SX-28760	.pdf

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Enclosure 2
CD-ROMs (2 discs)

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Enclosure 3
10 CFR 50.59 Summary Report

10 CFR 50.59 Summary Report

Activity: DCP SNC105747

Title: U1 Refueling Bridge Replacement

10 CFR 50.59 Evaluation Summary:

This activity replaces the existing Hatch Nuclear Plant (HNP) Unit 1 Stearns-Roger refueling platform with a new Westinghouse refueling machine (RFM). The new RFM has the capability of using the new automatic and semi-automatic software control for fuel movement to position the fuel assemblies to their specified locations in addition to the current manual control. The logic of the BWR standard interlocks between the RFM and the Reactor Control System is moved from circuitry hardware to the software of the Programmable Logic Control (PLC) of the new RFM.

The replacement of the Unit 1 refueling platform does not affect the ability of the plant to safely shut down after accidents nor increase the consequence of accidents evaluated in the HNP Updated FSAR.

Activity: DCP SNC494055

Title: Units 1&2 IRM Signal Filter Circuit Improvement

10 CFR 50.59 Evaluation Summary:

This activity modifies the Unit 1 and Unit 2 Intermediate Range Monitor (IRM) neutron detector signal processing circuit to filter short-lived, spurious signal spikes and prevent false high-high neutron flux trips and unnecessary reactor SCRAMs. The Unit 1 and Unit 2 IRM noise rejection must be improved to upgrade the barrier to false high-high flux trips and unnecessary reactor SCRAMs.

Capacitor C2 (1.0µF) in the Mean Square Analog module's output section is replaced with a 100µF capacitor. In addition, Capacitor C1 (10µF), Resistor R7, and the K10 relay contact effectively are removed from the output circuit. These modifications result in an increase in the Mean Square Analog module output signal time constant from 110 milliseconds on Ranges 1 through 6 and 10 milliseconds on Ranges 7 through 10 to 1000 milliseconds (1.0 second) on all ranges.

The increased time constant results in a lag in the Mean Square Analog module output signal value relative to the neutron detector input signal value such that the IRM scram setpoint id reached 0.719 seconds later relative to the current time constant. This delay results in reactor power reaching almost 39% of rated thermal power before the IRM high flux scram signal is generated. An analysis shows the peak fuel enthalpy for the control rod withdrawal event increases from 59.6 cal/gm to 111 cal/gm due to the increase in the time constant, but remains well below the fuel cladding damage limit of 170 cal/gm. Accordingly, the design function of the IRM system as described in the Updated Final Safety Analysis Report will continue to be met with the increase in the Mean Square Analog module output signal time constant to one second.

Activity: DCP SNC873725

Title: SAT 1E Open Phase Protection Trip Enable

10 CFR 50.59 Evaluation Summary:

This activity installs an open phase protection system on the high side neutral of SAT 1E that allows for an open phase condition to trip the high side breaker. The OPP system has been evaluated based on the requirements of 10 CFR 50.59 and following the guidance provided in NEI 96-07 and NEI 01-01. The addition of the OPP system and its method of evaluation does not result in:

- More than minimal increase in the frequency of occurrence or consequences of an accident previously evaluated;
- More than a minimal increase in the frequency of occurrence or consequences of a malfunction of an important-to-safety SSC;
- The creation of an accident of a different type or possibility for a malfunction of an important-to-safety SSC with a different result than any previously evaluated;
- Any impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment;
- A departure from a method of evaluation used in establishing design bases or in safety analysis.

Activity: DCP SNC881203

Title: SAT 1C and SAT 1D Open Phase Protection Trip Enable

10 CFR 50.59 Evaluation Summary:

This activity adds a new function to the OPP panels and existing protective relaying for each SAT-1C, SAT-1D, SAT-2C, and SAT-2D. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of each transformer. Each OPP panel is an addition to existing protective relays which lockout (isolate) the same transformers. These transformers are the offsite source of power to the onsite ac power distribution system during normal start-up, normal shutdown, and emergency shutdown. When the nuclear plant is producing power, the separate unit auxiliary transformers (UAT -1NXAA, UAT-1NXAB, UAT-2NXAA, and UAT-2NXAB), sourced from the main generator, provide the nonemergency power to the plant. Thus, isolation of SAT-1C, SAT-1D, SAT-2C, or SAT-2D while the nuclear plant is producing power will not result in complete loss of the non-emergency onsite ac power distribution system, as automatic fast transfers exist between the normal feed from SAT-2D (SAT-1D) to the backup feed from SAT-2C (SAT-1C). If both SAT-2D and SAT-2C (SAT-1D and SAT-1C) are locked out, then offsite ESF power is lost.

Energizing these transformers generates a large inrush current. To avoid false indication of an OPC and inadvertent transformer lockout, procedures for energizing SAT-1C, SAT-1D, SAT-2C, and SAT-2D must be modified to turn off the OPP panel manually when energizing the monitored transformer and re-energized manually after energizing the transformer.

Modification of the OPP system at the Hatch Nuclear Plant has been evaluated based on the requirements of 10 CFR 50.59 and following the guidance provided in NEI 96-07 and NEI 01-01. The conclusion of the Evaluation is that the proposed activities may be implemented under 10 CFR 50.59 without requiring prior USNRC review or approval.

Activity: FDC-H-17-009

Title: 10 CFR 50.59 Evaluation for GNF3 Fuel Introduction for Reloads

10 CFR 50.59 Evaluation Summary:

Plant Hatch Units 1 and 2 are transitioning from the GNF2 (10x10) fuel design to the more advanced GNF3 (10x10) fuel design for reloads, beginning with Hatch-2 Cycle-26.

The change in fuel design is documented in Fuel Design Change FDC-H-17-009. Based upon the 10 CFR 50.59 Screening, the introduction of GNF3 fuel at Hatch resulted in Questions 1 and 3 of the Screening answered as "Yes", therefore requiring this aspect of the proposed activity being addressed in this 10 CFR 50.59 Evaluation.

GNF3 fuel bundles are designed and manufactured by Global Nuclear Fuel (GNF) to have the same form, fit, and function as earlier fuel designs used for Hatch, and also manufactured by GNF. All of the licensing criteria for fuel, as specified by GESTAR-II, have been demonstrated (including the cycle-specific analyses performed for each reload), therefore ensuring that the design functions of the fuel and reactor core, containment, and ECCS systems are not adversely affected.

Since a GNF3 fuel assembly has physical characteristics (external dimensions, weight, material composition, etc.) that are fully compatible with the existing plant equipment, will be handled in the same manner, and weighs less than the initial core fuel assemblies, the frequency of occurrence of the fuel handling accident will not be increased. Since all other accidents described in the FSAR are initiated by operator error or equipment failure or malfunction outside the fuel and core, use of a new fuel bundle design, which meets all applicable design and licensing requirements for fuel, does not have any effect on the frequency of occurrence of those accidents or the malfunction of other SSCs.

The increase in the GNF3 source term compared to the base GE14 source term is bounded by the AST design basis source term, which used the base GE14 source term plus 10% additional conservatism. Therefore, the radiological consequences of accidents or malfunctions in the current design basis dose analyses remain valid.

GNF3 fuel bundles have physical characteristics (external dimensions, weight, material composition, etc.) which are fully compatible with the existing plant equipment and systems, and will be handled in the same manner. The behavior of GNF3 fuel in the core has been properly evaluated. Therefore, using GNF3 fuel bundles will not increase the possibility of an accident of a different type or result in a malfunction of an SSC important to safety with a different result.

Results for the ECCS-LOCA analysis based on GNF3 showed an increase in the maximum local cladding oxidation. However, the results remain within the design basis limits. The

increased GNF3 core decay heat was explicitly evaluated relative to impacts to containment and the results show that allowable design basis limits are not exceeded or altered for GNF3 introduction. Prior to each reload, cycle-specific power distribution limits for each fuel type in the core are established to assure that the margin of safety for fuel cladding integrity will not be reduced. In addition, cycle specific analyses confirm that the ASME overpressure protection criteria will not be exceeded during the limiting pressurization event. Therefore, the use of the GNF3 fuel design does not have an impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment.

Therefore, the responses to Questions 1 through 6 all result in "NO" answers, the response to Question 7a is "YES" but the response to Question 7b is "NO".

GEH has prepared and obtained NRC approval of the TRACG-LOCA licensing topical report (LTR) that describes the TRACG-LOCA methodology. The TRACG-LOCA methodology has been applied at Hatch within the applicable limitations and conditions of the NRC-approved methodology as documented in Reference 9. Therefore, the use of TRACG-LOCA for ECCS-LOCA analysis at Hatch is not a departure from a method of evaluation described in the UFSAR that would require prior NRC approval, because this method of evaluation is approved by the NRC for the intended application.

Therefore, the response to Question 8 results in a "NO" answer.

Based on the above, a license amendment from the NRC is not required before the activity may be implemented.

Activity: LDCR 2019-003

Title: Revise Hatch Unit 1 FSAR Table C.3-1 to reflect new calculated stress values for the "Top Guide-Highest Stress Beam" and "Top Guide Beam End Connections" due to indications

10 CFR 50.59 Evaluation Summary:

During the Unit 1 2018 Spring Refueling Outage (1R28), VT-1 best effort inspections were performed on the Top Guide Beam Connection. Indications were found on the top guide beam connections as described in INR H1R28 IVVI-18-06.

Edwin I. Hatch Nuclear Plant contracted General Electric Hitachi Nuclear Energy (GEH) to perform a structural evaluation on the top guide beam connection condition. Based on the GEH evaluation, which becomes the new analysis-of-record for stresses on the "Top Guide-Highest Stress Beam" and "Top Guide Beam End Connections", it was concluded that the top guide and its beam connections could perform their safety-related function in the as-found condition. As such, this activity considers continued use (as-is) of the top guide from a licensing basis perspective.

Although new calculated stress values are generated, the method of evaluation, based on ASME Boiler and Pressure Vessel Code Section III for type 304 stainless steel plate, is

unchanged. In other words, the applicable load combinations and acceptance criteria presented in UFSAR Table C.3-1 for these components continues to be used as the design basis. Markups of the UFSAR for this activity as provided in LDCR 2019-003 update the calculated (actual) stresses the for the "Top Guide-Highest Stress Beam" and "Top Guide Beam End Connections" indicated in Table C.3-1 of the UFSAR while also clarifying that the GEH evaluation is the current analysis-of-record for these components.

The conclusion of this 10 CFR 50.59 evaluation is that the activity may be implemented without requiring prior USNRC review and approval.