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

Ladies and Gentlemen:

In accordance with the 10 CFR 50.4(b) and 50.71(e), Southern Nuclear Operating Company (SNC) hereby submits Revision 34 to the Edwin I. Hatch Nuclear Plant (HNP) Updated Final Safety Analysis Reports (UFSAR). The revised HNP UFSAR pages, indicated as Revision 34, reflect changes through July 2016.

The HNP Unit 1 and Unit 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 82 for Unit 1 and Revision 93 for Unit 2, reflect changes through July 2016.

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 the current pages of the HNP UFSAR, TS Bases, Technical Requirements Manual (TRM), and Fire Hazard Analysis (FHA) are being submitted on CD-ROM in portable document format (PDF) with non-proprietary browser included. The revised HNP TRM pages, indicated as Revision 103 for Unit 1 and Revision 106 for Unit 2, reflect changes through July 2016. The revised HNP Unit 1 and 2 FHA, indicated as Revision 34, reflects changes through July 2016.

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

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In accordance with NEI 99-04, "Guideline for Managing NRC Commitment Changes," Revision 0, SNC is required to submit a Revised NRC Commitments Report. There were no commitment changes made from March 2015 through June 2016; therefore, there is no Revised Commitments Report included in this submittal.

SNC conducted a review of HNP Units 1 and 2 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 CD-ROMs (Enclosure 2). Enclosure 3 provides the 10 CFR 50.59 Report.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Mr. J. T. Wheat states he is the Nuclear Licensing Manager for Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

J. T. Wheat
Nuclear Licensing Manager

Sworn to and subscribed before me this 25th *day of* August *, 2016.*

Notary Public

My commission expires: 1/2/2018

JTW/GLS

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

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO (w/o enclosures)
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Mr. D. R. Vineyard, Vice President – Hatch (w/o enclosures)
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(enclosure 2 – CD ROMs only)

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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 UNIT 2 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	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 PART 1	.pdf
007	HATCH TRM UNIT 1 PART 2	.pdf
008	HATCH TRM UNIT 2	.pdf
009	HATCH FHA	.pdf
	DISC 2	
010	HATCH FSAR_REF DWGS PART 1 A-21603 – H-11606	.pdf
011	HATCH FSAR_REF DWGS PART 2 H-11607 – H-16002	.pdf

SEQ	CONTENT (FILENAME)	EXTENSION
012	HATCH FSAR_REF DWGS PART 3 H-16003 – H-16174	.pdf
013	HATCH FSAR_REF DWGS PART 4 H-16176 – H-16339	.pdf
014	HATCH FSAR_REF DWGS PART 5 H-16512 – H-19941	.pdf
015	HATCH FSAR_REF DWGS PART 6 H-19942 – H-21114	.pdf
016	HATCH FSAR_REF DWGS PART 7 H-22250 – H-24748	.pdf
017	HATCH FSAR_REF DWGS PART 8 H-24749 – H-26036	.pdf
018	HATCH FSAR_REF DWGS PART 9 H-26037 – H-26102	.pdf
019	HATCH FSAR_REF DWGS PART 10 H-26103 – S-15290	.pdf
020	HATCH FSAR_REF DWGS PART 11 S-15304 – S-40969	.pdf
021	HATCH FSAR_REF DWGS PART 12 S-53448 – S-56429 Part 1	.pdf
022	HATCH FSAR_REF DWGS PART 13 S-56429 Part 2 – SX-28760	.pdf
	NRC File Nomenclature	.doc

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

Enclosure 2

CD-ROMs (2) NRC Submittal

Edwin I. Hatch Nuclear Plant – Units 1 & 2
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Enclosure 3

10 CFR 50.59 Report

10 CFR 50.59 Summary Report

Activity: Design Change SNC548828

Title: Unit 1 Panel 1H11-P601 Recorder Replacement

10 CFR 50.59 Evaluation Summary: This activity is a DCP that replaces existing analog paper chart recorders in the Unit 1 MCR panel 1H11-P601 with digital paperless recorders. The DCP also removes the vibration recorder that is no longer used and replaces it with a cover plate. These replacement recorders are used in the RHR system to display and record flowrate information of the RHR discharged line by throttling the Full Flow Test Line valves. The RHR Discharged Header Flow recorders are safety related. The replacement recorders use a microprocessor and firmware (software) to function, and thus are considered a digital upgrade. Since the two RHR Discharged Header Flow recorders that display flowrate information are qualified to Regulatory Guide 1.97 Category 1 requirements with functionality controlled through software, these recorders introduce the possibility of having a common mode failure mechanism. The guidance of NEI 01-01 has been used to evaluate these recorders in respect to new digital upgrades. The likelihood of an occurrence of a recorder malfunction is minimal due to the V&V process used and operating history and there are no new system malfunctions with a different result than that evaluated in the Updated FSAR. Furthermore, the new recorders will meet current performance requirements, improve the human/machine interface, and will be used in the same way as the replaced recorders with the same information. Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.

Activity: Design Change SNC548830

Title: Make Temporary Modification (TM) SNC454213 permanent – Crosstie of Unit 1 Makeup Demineralized Water System to Unit 1 Condensate Supply to Unit 1 Radwaste Building.

10 CFR 50.59 Evaluation Summary: This DCP will make TM SNC545213 a permanent plant modification. When the existing supply piping upstream of the 1P11-F021 valve was capped for tritium testing, there was no condensate supply water to the Unit 1 Radwaste building. With the installation of the TM, a connection was made between the makeup demineralized water piping and the existing condensate to radwaste supply water piping in the Unit 1 Radwaste building. The supply piping upstream of the 1P11-F021 valve has been reinstalled and the flow of condensate supply water to the Unit 1 Radwaste has been restored by DCP SNC456832.

The demineralized water system does not provide any functions that are necessary to safely shut down the reactor, maintain the plant in a safe shutdown state, or mitigate the consequences of an accident. The condensate transfer system is not safety related and is not required to mitigate the consequences of any accident previously evaluated in the UFSAR. The condensate transfer system and the demineralized water system are not initiators for any accidents. Any new accident scenarios created for the condensate transfer system and the demineralized water system are bounded by those previously evaluated in the UFSAR. Any malfunction of the crosstie isolation valve or check valves remains the same as previously evaluated in the UFSAR. The demineralized water system and the condensate transfer system are not required to mitigate the consequences of any malfunction of SSCs. The condensate demineralized water system and the condensate transfer system are not credited for maintaining the integrity of the fuel cladding, reactor coolant pressure boundary, or containment.

There is no effect on any design bases or safety analyses. The methods of evaluation used in this activity are the same as was used in the original design.

Therefore, this will not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Updated FSAR, does not result in any increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the UFSAR, does not result in any increase in the consequences of an accident previously evaluated in the Updated FSAR, will not result in any increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the Updated FSAR, will not create the possibility for an accident of a different type than any previously evaluated in the Updated FSAR, will not create the possibility for a malfunction of an SSC important to safety with any other result than any previously evaluated in the Updated FSAR, will have no impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment, and does not result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Activity: FDC-H-13-005; LDCR 2015-017

Title: GNF2 Fuel Design for Reloads

10 CFR 50.59 Evaluation Summary: Plant Hatch Units 1 and 2 are transitioning from the GE14 (10x10) fuel design to the more advanced GNF2 (10 x10) fuel design for core reloads, beginning with Hatch-2 Cycle-24. The change in fuel design is documented in Fuel Design Change FDC-H-13-005. GNF2 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. All of the licensing criteria for fuel, as specified by NEDE-24011-P-A-19, "General Electric Standard Application for Reactor Fuel (GESTAR-11) have been demonstrated (excluding the cycle-specific analyses performed for each reload), therefore ensuring that the design functions of the fuel and reactor core are not adversely affected.

Since a GNF2 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 GNF2 source term compared to the base GE14 source term is bounded by the AST licensing basis source term, which used the base GE14 source term, but added 10% conservatism. Therefore, the radiological consequences of accidents or malfunctions of SSCs are bounded.

GNF2 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 GNF2 fuel in the core has been properly evaluated. Therefore, using GNF 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 GNF2 showed an increase in the peak cladding temperature (PCT) from 1930 °F (GE14, corrected) to 2110 °F, and the maximum local oxidation increased from < 2% to < 6%. However, these results remain within the 10 CFR 50.46 regulatory design basis requirements of 2200 °F for PCT and 17% for maximum local oxidation. Slight changes in the decay heat calculation were evaluated to have insignificant impacts on containment analyses. Prior to each reload, cycle-specific power distribution limits for each fuel type in the core will be established to assure that the margin of safety for fuel cladding integrity will not be reduced as a result of using GNF2. 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 GNF2 fuel design does not have an impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment.

The updated spent fuel pool rack criticality analysis utilized an updated Monte Carlo simulation model which utilizes more recent, conservative standards supplemented by additional NRC guidance. This approach has been previously recognized by the NRC as an industry standard, and with proper validation, has been accepted by the NRC for use in similar applications by other licensees. Therefore, there is not a departure from an approved method of evaluation.

Activity: Design Change SNC114879

Title: Unit 2 EDG LOCA/LOSP Timer Replacement

10 CFR 50.59 Evaluation Summary: This DCP replaces the Rochester Instrument Systems (RiS) digital microprocessor based timing modules Model CS-1601, used in 2A, 1B (when aligned to Unit 2), and 2C Emergency Diesel Generator (EDG) RiS Model 1600 Sequential Loading Timers, with reverse engineered digital field programmable gate array based (FPGA) timing modules manufactured by MPR Associates. This DCP will also retire-in-place the Veeder-Root counters in panels 2H21-P303, 2H21-P304, and 2H21-P305. The counter/display functions of these devices are provided by the replacement digital timing modules. The timing modules are safety related. For this DCP, programmable logic is programmed using the "VHDL" language. Since VHDL has similar design and development issues as software, this activity is considered a digital upgrade and follows the guidance of NEI 01-01. Each timing module starts the associated EDG service water pump after the EDG starts from either a Loss of Coolant Accident (LOCA) or due to a Loss of Off-Site Power (LOSP). The new EDG timing module will perform the same function as the existing modules and was developed and qualified in accordance with regulatory requirements. The replacement module is designed, developed, manufactured, reviewed, verified, validated, environmentally qualified, and tested under a NQA-1 compliant nuclear quality assurance process. The FPGA code was designed, developed, implemented, tested, reviewed, safety analyzed, verified, and validated under a software lifecycle process compliant to NQA-1, under a NQA-1 compliant nuclear quality assurance process, and thus meeting regulatory requirements and expectations. In addition to the normal verification a validation activities which include internal timing verification, the safety critical portions of the programmable logic are designed to be 100% testable and were 100% tested with all possible combinations of externally applied field contact inputs (including internal diagnostic failures) applied to all internal safety critical states. Installing the new FPGA replacement module will not result in more than a minimal increase in frequency of occurrence of an accident, will not result in more than a minimal increase in likelihood of occurrence of a malfunction, will not result in more than a minimal increase in the consequences of an accident or the consequences of a malfunction of an SSC important to safety previously evaluated in the final safety analysis report. Failure analysis has been performed at the module level and did not identify any failure modes different from the existing, implemented RiS design. The programmable logic was implemented under a software-like life cycle process, customized for programmable logic, which incorporates best engineering guidance available from the nuclear industry. The life cycle has been evaluated by the EPRI task force documenting best industrial and thus recommended practice for the US and French nuclear power industries for programmable logic. Based on the programmable logic life cycle processes used and the analyses and reviews performed, the possibility of programmable logic common cause failure has been minimized. Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.

Activity: Design Change 1039002601; LDCR 2013-037

Title: Unit 1 Plant Service Water (PSW) Modifications

10 CFR 50.59 Evaluation Summary: The temporary design change (TM SNC 454213) added a blind plate in place of the 1P41-D166 orifice plate, located in the 1B diesel generator room. The new DCP will remove valves 1P41- F401A and 1P41-F402A and replace them with a spool piece. Additionally a blind plate will be placed where valve 1P41-F402A was located to prevent water from entering the spool piece. The temporary design change also removed a section of the 12" shield piping in the 2G switchgear room in the Diesel Generator Building between the branch tee to the 1A diesel generator and the south wall. Vent valve 1P41-F1173 and drain valve 1P41-F1140 located in the section of 12" shield piping were removed. They were relocated to the 12" shield piping north of the branch tee. The 8" PSW piping inside the shield piping was cut and capped near the tee just past the branch to the 1A EOG room. A spool piece of the 12" shield piping with a cap was welded over the PSW cap. This sealed the PSW piping from the 2G switchgear room. A Foreign Material Exclusion (FME) plate was welded over the end of the remaining abandoned section of the 12" shield piping.

The proposed modification will remove Division I of Unit 1 PSW as a secondary cooling source to the 1B diesel generator. A failure of the Standby Service Water (SSW) pump will therefore result in the 1B diesel generator being tied to Division II of PSW. There is minimal increase in the consequences of that malfunction since the 1B diesel will subsequently be aligned to Division II of Unit 1 PSW with two operable PSW pumps. Even though Division I will not be available, this modification does not introduce an active single failure that results in the loss of cooling to the 1B diesel with the cooling water aligned to Division II of PSW and both PSW pumps available. The loss of the SSW pump will place the plant in a 60 day Technical Specifications Required Action Statement (RAS) per TS Limiting Condition for Operation (LCO) 3.7.3 A. If an additional Unit 1 PSW pump subsequently becomes inoperable with the SSW pump already out of service, the plant will be under a
30 day RAS per TS LCO 3.7.2. If that inoperable PSW pump happens to be a Division II pump, a subsequent failure of the remaining Division II pump during an accident would leave only one diesel generator available. On the other hand, if a Division I PSW pump were to become inoperable simultaneous with the SSW pump out of service, subsequent failure of any PSW pump during the accident scenario would result in two diesel generators available. Both scenarios result in a 30 day TS RAS even though the loss of a Division II PSW pump leads to a more degraded situation. However, even in the more degraded case, one diesel generator would remain available, powering two low pressure ECCS pumps to provide cooling water to the reactor.

The proposed design change does not involve more than a minimal increase in the consequences of a previously evaluated accident and the possibility of an accident of a different type from one previously evaluated is not created.

Activity: Design Change SNC528997

Title: Unit 1 Recorders Replacement for Panel 1H11-P603

10 CFR 50.59 Evaluation Summary: This activity is a DCP that replaces existing analog two pen and digital four pen paper chart recorders in the Unit 1 Main Control Room Reactor Control Console 1H11-P603 with digital paperless recorders with additional input capacity. Additionally, the DCP removes associated signal selector switches and connects all process inputs directly to the recorders. These recorders are used in the Neutron Monitoring System (NMS) to display, trend, and record process information. The NMS recorders display Regulatory Guide 1.97 Type B process variables and meet the requirements of the current licensing basis. The replacement recorders use a microprocessor and firmware (software) to function, and thus are considered a digital upgrade. Since the four NMS recorders that display APRM process information are qualified to Regulatory Guide 1.97 Category 1 requirements with functionality controlled through software, these recorders introduce the possibility of having a common mode failure mechanism. The guidance of NEI 01-01 has been used to evaluate these recorders in respect to new digital upgrades. The likelihood of an occurrence of a recorder malfunction is minimal due to the V&V process used and operating history and there are no new system malfunctions with a different result than that evaluated in the Updated FSAR. Furthermore, the new recorders will meet current performance requirements, improve the human/machine interface, and be used in the same way as the replaced recorders with the same information. Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.

Activity: Design Change SNC539300; LDCR 2014-077

Title: Unit 1 Reliable Hardened Containment Vent Design

10 CFR 50.59 Evaluation Summary: The hardened containment vent system was originally installed under DCR 1H89-278 to provide a vent line to bypass the SGTS from the primary containment directly to the main stack for use in severe accident conditions only. The original design installed a redundant passive isolation valve (1T48-F082) and passive rupture disc (1T48-D346) to isolate the 18 inch vent line during normal operation and Design Basis Analysis (DBA) operating conditions. The rupture disc setpoint (51 psig) was established such that the disc would rupture under pressures below the maximum containment pressure design limit (56 psig), but remain intact when exposed to the pressure (20 psig) expected to develop in the primary containment purge line following a LOCA. The proposed activity will replace the rupture disc (1T48-D346) in the hardened containment vent path with a disc of a lower setpoint to facilitate easier, purposeful bursting of the disc in response to beyond design basis events. The new setpoint will be lower than the pressure expected to develop in the purge line of 20 psig following a LOCA. As such, the hardened containment vent line isolation will no longer be single failure proof. Instead, the upstream isolation valve, 1T48-F082, will be relied upon to serve as a passive barrier to prevent exposing the rupture disc to the 20 psig expected to develop in the 18 inch purge line under LOCA conditions. As a result, isolation valve 1T48-F082 will be added to the Table T8.2-1. "Secondary Containment Devices," in the Technical Requirements Manual (Reference 17).

Isolation valve 1T48-F082 is normally closed, fails closed, and is operated with a key lock switch to prevent inadvertent operator actuation during normal operations and design basis accidents; therefore, it continues to be credited as a passive device, and serves as a barrier to the effects of a LOCA on the downstream rupture disc, particularly pressure, until the primary containment isolation valves on the drywell and wetwell purge lines close. Failure of isolation valve 1T48-F082 (i.e., the valve fails "open") need not be postulated, because this valve can still be credited as a passive device. The new rupture disc setpoint will be set to a value above the pressure calculated (Reference 10) that the disc would realize, while taking into account leakage past isolation valve 1T48-F082 for the 5-second period (Reference 16) of time during a LOCA before the primary containment isolation valves close. The rupture disc will still serve its design function as a zero bypass leakage barrier. The isolation valve (1T48-F082) and rupture disc (1T48-D346), together, maintain their design basis function for pressure boundary and bypass leakage prevention during normal and accident conditions. There is no analysis which uses the rupture disc setpoint in determining consequences of any radiological release.
Therefore, the proposed activity does not:

- Result in more than a minimal increase in the frequency of occurrence or consequences of an accident previously evaluated in the UFSAR;
- Result in more than a minimal increase in the likelihood of occurrence or consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR;
- Create the possibility for an accident of a different type or a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR;
- Have an impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment.

(Note: This is a modification being performed to meet Fukushima order EA-13-109)

Activity: Design Change SNC629425

Title: Unit 2 Outage Recorders Replacement (2H11P650, 2H11P601, 2H11P602)

10 CFR 50.59 Evaluation Summary: This activity replaces/combines existing obsolete GE 521, GE 530, GE 531 and L&N Speedomax series chart style recorders located in Main Control Room (MCR). The recorders are located in panels 2H11-P601, 2H11-P602, and 2H11-P650. These recorders are used in the Nuclear Boiler (B21), Reactor Recirculation (B31), Reactor Heat Removal (E11), Radwaste (G11), Reactor Water Cleanup (G31), Condensate/Feedwater (N21), Circulating Water (N71) and Sampling (P33) Systems to display, trend, record, and alarm process information. The Residual Heat Removal System recorders (2E11-R608A and 2E11-R608B) are Safety-Related and identified as Regulatory Guide (RG) 1.97 Type D variables (RHR flow & LPCI flow) per Updated FSAR Table 7.5-1.

The replacement recorders use a microprocessor and firmware (software) to function, and thus are considered a digital upgrade. Since the two RHR recorders are qualified as safety related with functionality controlled through software, these recorders introduce the possibility of having a common mode failure mechanism. The guidance of NEI 01-01 has been used to evaluate these recorders in respect to new digital upgrades. The likelihood of an occurrence of a recorder malfunction is minimal due to the V&V process used and operating history and there are no new system malfunctions with a different result than that evaluated in the Updated FSAR. Furthermore, the new recorders will meet current performance requirements, improve the human/machine interface, and will be used in the same way as the replaced recorders with the same information. Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.

Activity: Design Change SNC581096

Title: Replace Unit 1 Vital AC Inverter 1R44-S001

10 CFR 50.59 Evaluation Summary: The Vital AC Inverter provides power for vital services for which power interruption should be avoided. These vital services are necessary for the operation of the plant but are not required for plant safety. The new Vital AC system static inverter is rated 76 kVA, 120/240 V, ±1% voltage regulation, and ±0.5% frequency regulation.

The replacement Vital AC Inverter is compatible with the installed environment, including seismic and electromagnetic compatibility (EMC), such that system performance will not be degraded compared to the existing Vital AC Inverter. The replacement Vital AC Inverter will have no adverse impact on the installed environment, including seismic and EMI/RFI emissions. The electrical load of the replacement Vital AC Inverter is within the capabilities of its connected power sources.

The guidance of NEI 01-01 has been used to evaluate the Vital AC Inverter with respect to new digital upgrades. The likelihood of an occurrence of Vital AC Inverter malfunction is minimal due to the V&V process used and acceptance testing. There are no new system malfunctions with a different result than that evaluated in the Updated FSAR. Furthermore, the new Vital AC Inverter will meet the current performance requirements and is used in the same way as the existing Vital AC Inverter while performing the same functions.

Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.

Activity: Design Change SNC629427

Title: Unit 2 Non-outage Recorders Replacement (2H11P650 and 2H11P689)

10 CFR 50.59 Evaluation Summary: This activity replaces/combines existing obsolete GE 531, GE 521, and L&N Speedomax series chart style recorders located in the Main Control Room (MCR). All the recorders are located in panel 2H11-P650 except 2T48R647 which is in panel 2H11-P689. These recorders are used in the Main Steam (N11), Condensate/Feedwater (N21), Turbine Controls (N32), Extraction Steam (N36), Reheat (N38), Generator and Auxiliary (N40), and Containment Purge, Vent & Nitrogen (T48) Systems to display, trend, record, and alarm process information. The Containment Purge, Vent & Nitrogen System recorders (2T48-R601B, 2T48-R609, & 2T48-R647) are Safety-Related and identified as Regulatory Guide (RG) 1.97 instruments per Updated FSAR Table 7.5-1.

Recorders	RG 1.97 Type	Category
2T48-R601B	B & C	1 & 3
2T48-R609	B, C, & D	1 & 2
2T48-R647	A & D	1 & 2

The replacement recorders use a microprocessor and firmware (software) to function, and thus are considered a digital upgrade. Since the three Containment Purge, Vent & Nitrogen System recorders are qualified as safety related with functionality controlled through software, these recorders introduce the possibility of having a common mode failure mechanism. The guidance of NEI 01-01 has been used to evaluate these recorders in respect to new digital upgrades. The likelihood of an occurrence of a recorder malfunction is minimal due to the V&V process used and operating history and there are no new system malfunctions with a different result than that evaluated in the Updated FSAR. Furthermore, the new recorders will meet current performance requirements, improve the human/machine interface, and will be used in the same way as the replaced recorders with the same information. Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.