Enclosure 3 contains security-related information and should be withheld under 10 CFR 2.390.
Upon removal of Enclosure 3, this correspondence is suitable for public disclosure.

Southern Nuclear

Regulatory Affairs

3535 Colonnade Parkway
Birmingham, AL 35243
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September 3, 2024

Docket Nos.: 50-321
50-366

NL-24-0334

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant - Units 1 & 2
Revision 40 to the Updated Final Safety Analysis Report, Technical Specifications 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 40 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 40, reflect changes through July 31, 2024.

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 123 for Unit 1 and Revision 135 for Unit 2, reflect changes to the TS Bases through July 31, 2024.

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, and the Technical Requirements Manual (TRM) are being submitted on CD-ROM in portable document format (PDF). The revised HNP TRM pages, indicated as Revision 135 for Unit 1 and Revision 140 for Unit 2, reflect changes to the TRM through July 31, 2024.

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 evaluation of each.

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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 (September 1, 2022 to July 31, 2024).

SNC conducted a review for the period from January 1, 2022 to March 31, 2024 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, public documents, and Enclosure 3, Non-Public Documents). Enclosure 3 contains security-related information. SNC requests that Enclosure 3 be withheld from public disclosure in its entirety in accordance with 10 CFR 2.390(d)(1). Enclosure 4 provides the 10 CFR 50.59 Summary Report.

This letter contains no regulatory commitments. If you have any questions, please contact Mr. Ryan Joyce at (205) 992-6468.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 3rd day of September 2024.

Respectfully submitted,

Jamie Coleman

Jamie M. Coleman
Director, Regulatory Affairs
Southern Nuclear Operating Company

JMC/RMJ

Enclosures:

1. CD-ROM Table of Contents
2. CD-ROM – Public Documents (1 Disc)
3. CD-ROM – Non-Public Documents (Withhold from public disclosure in accordance with 10 CFR 2.390(d)(1)) (1 Disc)
4. 10 CFR 50.59 Summary Report

cc: Regional Administrator, Region II
NRR Project Manager – Hatch
Senior Resident Inspector – Hatch
Director, Environmental Protection Division – State of Georgia
RType: CHA02.004

NL-24-0334

Edwin I. Hatch Nuclear Plant - Units 1 & 2
Revision 40 to the Updated Final Safety Analysis Report,
Technical Specifications 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

Enclosure 1
CD-ROM Table of Contents

SEQ	FILENAME CONTENT	EXTENSION
	DISC 1 PUBLIC DOCUMENTS FOR ENCLOSURE 2	
000	NRC File Nomenclature	.pdf
001	HATCH FSAR_U1 UNIT 1 Active Page List Table of Contents Chapters 1 thru 14 Appendices A thru K, 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

SEQ	CONTENT / FILENAME	EXTENSION

DISC 2
NON-PUBLIC DOCUMENTS FOR ENCLOSURE 3

NL-24-0334

Edwin I. Hatch Nuclear Plant - Units 1 & 2
Revision 40 to the Updated Final Safety Analysis Report,
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Enclosure 2
CD-ROM – Public Documents
(1 Disc)

NL-24-0334

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Enclosure 3
CD-ROM – Non-Public Documents (Withhold from public disclosure in accordance with
10 CFR 2.390(d)(1))
(1 Disc)

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Enclosure 4
10 CFR 50.59 Summary Report

10 CFR 50.59 Summary Report

Activity: LDCR 2022-016

Title: Remove FSAR Requirement for Cask Pit Gate Installation During Transfer Cask Movement

10 CFR 50.59 Evaluation Summary:

This activity removes the site FSAR requirement to install the cask pit gates when moving a dry cask to or from the cask pit. A breach of the cask pit liner drains the SFPs, which is an analyzed scenario in the FSAR. With the cask pit gates removed, a breach of the cask pit liner would allow water to drain from both spent fuel pools (SFPs). Installation and removal of the cask pit gates each consists of two heavy lifts (one gate is installed on each side of the cask pit). Eliminating four total heavy load lifts in the cask pit lowers the possibility of a lifting and rigging event that would cause a breach in the cask pit liner.

This change does not create a more than minimal increase in the frequency of a fuel pool draining event or the likelihood of occurrence of a malfunction of an SSC due to the remaining defense-in-depth measures summarized below:

1. By removing the requirement to install cask pit gates during transfer cask movements, four heavy load lifts in the cask pit will be eliminated per dry cask loaded. A collision between the transfer cask and cask pit is unlikely to occur due to the single-proof-failure design of the Unit 1 overhead crane (OHC) and heavy load handling practices employed during transfer cask lifts. Additionally, the transfer cask is analyzed against tipover during a design basis seismic event while sitting in the cask pit.

2. A collision between the transfer cask and cask pit is unlikely to breach the cask pit liner due to the slow speeds of the U1 OHC and large size of the transfer cask to spread and dissipate impact energy such that no sufficient force develops to penetrate the cask pit liner.

All pre-existing, UFSAR-described, dose analyses have been reviewed. There is no effect on these analyses, either directly or indirectly. Based on this, not installing cask pit gates during transfer cask movement into and out of the cask pit does not create an increase in the consequences of an accident or malfunction of an SSC evaluated in the FSAR.

In the unlikely event of the breach of the cask pit liner, the fuel pools and cask pit will drain at a rate of less than 150 gallons/minute due to the size of the piping in the leak detection system. This means a maximum SFP level drop rate of 6 in/hour. This is less than half of the make-up rate of either pool's fuel pool cooling system. There is a sufficient make-up capacity to prevent SFP level drop. Additionally, there are procedures and equipment on the refuel floor to address a breach in the cask pit liner. There are no new design interfaces created.

Specifically, all system-level interfaces remain unchanged. The change is completely described by existing safety analyses and there is no need for a new analysis. Based on this analysis, not installing cask pit gates during transfer cask movement into and out of the cask pit does not create the possibility for an accident of a different type than evaluated in the site FSAR and does not create a malfunction of an SSC with a result different than analyzed in the FSAR.

This change does not create any new types of SFP drainage or increase SFP drain rate in the event of a cask liner breach. There is no impact to the integrity of the fuel stored in the SFP racks.

This activity removes the site FSAR requirement to install the cask pit gates when moving a dry cask to or from the cask pit. It does not revise or replace a method of evaluation described in the Updated FSAR.

Activity: LDCR22-17

Title: Technical Requirements Manual (TRM) Burden Reduction

10 CFR 50.59 Evaluation Summary:

This activity replaces Technical Requirements Manual (TRM) TLCO 3.3.7, "MCREC System Instrumentation," Required Actions to trip the channel or place the system in operation with a Required Action for initiation of a Condition Report. The changes associated with the instrumentation addressed in this TRM Specification are administrative changes to procedural actions to be taken when the instrumentation is nonfunctional. The MCREC instrumentation Functions are not assumed in the evaluation of MCREC initiation in any design basis accident or transient. They are anticipatory initiation signals only which are not considered in the design basis evaluations. The Control Room Air Inlet Radiation-High Function is the only Function assumed to initiate the MCREC System in the design basis evaluations.

This activity removes Technical Requirements Manual (TRM) TLCO 3.3.8, "Offgas Post-Treatment Instrumentation," Required Actions to trip the channel. The changes associated with the instrumentation addressed in this TRM Specification are administrative changes to procedural actions to be taken when the instrumentation is nonfunctional. The instrumentation Functions are not assumed in the design basis evaluations.

This activity removes Technical Requirements Manual (TRM) TLCO 3.4.1, "Reactor Coolant System Chemistry" requirements. The changes associated with the TRM Specification are administrative changes to procedural actions to be taken which are less stringent than other procedural actions which implement the EPRI BWRVIP-190 chemistry guidelines.

This activity replaces Technical Requirements Manual (TRM) TLCO 3.4.2, "Structural Integrity," Required Actions with a Required Action for initiation of a Condition Report. The changes associated with the structural integrity addressed in this TRM Specification are administrative changes to procedural actions to be taken when the Class 1, 2, 3 or MC components are nonconforming or a required inspection has been missed. The Actions for a nonconforming component have not changed in that FUNCTIONALITY or OPERABILITY must still be addressed. Further, the Actions for a missed inspection, while perhaps not on the same time frames, will continue to be addressed in a timely manner in accordance with the program and the regulations, with due consideration of the potential impact on safety.

This activity removes Technical Requirements Manual (TRM) TLCO 3.7.1, "Snubbers" requirements. The changes associated with the TRM Specification are administrative changes to procedural actions to be taken which are duplicative of other procedural actions which implement the TS definition of Operability and TS 3.0.8 for supported systems.

The activities were not identified to more than minimally increase the frequency of occurrence or consequences of an accident previously evaluated in the Updated FSAR, more than minimally increase the likelihood of occurrence or consequences of a malfunction of a structure, system, or component important to safety previously evaluated in the Updated FSAR, create the possibility for an accident of a different type or for a malfunction of an SSC important to safety with a different result than any previously evaluated in the Updated FSAR, have any impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment, or result in a departure from a method of evaluation described in the Updated FSAR used in establishing the design bases or in the safety analyses.

Activity: SNC1100336

Title: Unit 1 Main Steam Line Leak Detection Temperature Switch Replacement

10 CFR 50.59 Evaluation Summary:

Main Steam Line (MSL) Leak Detection for Hatch Unit 1 includes the Turbine Building Leak Detection System. There is a total of 64 temperature switches for the Unit 1 MSLs located in the Turbine Building condenser bay. The temperature switches provide an alarm when their setpoint is exceeded. Operators respond to the alarm, and initiate appropriate corrective actions, as necessary, in accordance with Technical Specification 3.7.10 and the associated annunciator response procedures.

The existing temperature switches are safety-related Fenwall thermo-switches, arranged in four sixteen-input channels or strings. Each string's thermo-switches are connected in series. Due to the unacceptable performance of the switches, this design change package replaces the switches with Conax Type-T thermocouples (T/Cs), like those deployed on Unit 2. In addition, four Yokogawa DX2020 recorders are connected to the T/Cs for continuous monitoring and indication of temperature excursions. (These correspond to the four strings for equivalent coverage of all temperature element signals.) The temperature switch inputs to the existing annunciator circuits are replaced with outputs from the new recorders. The recorder's high temperature ALARM output, FAIL output, and STATUS (measurement error) output contacts are wired in series to the annunciator circuit. With these outputs wired in series, any credible failure of the new equipment will initiate a TEMP HIGH annunciation. The new recorders are located in the main control room. Yokogawa DX2000 series recorders have been installed in the main control room numerous times and operators are familiar with them.

This modification is considered to be a digital upgrade. The existing temperature switches are difficult to validate and do not allow operators to view temperature trends. The new recorders provide operators a greater ability to validate temperature high annunciations produced by the T/Cs. T/Cs are individual inputs to their channel's recorder versus the existing series connections. Therefore, individual failures are evident via the recorder versus the current inability to detect a failure of a specific thermo-switch.

The design function of the MSL leak detection instrumentation (i.e., the turbine building area temperature monitoring instrumentation) is to provide for temperature monitoring in the condenser bay area of the turbine building, near the MSLs. Credit for the MSL leak detection instrumentation in the condenser bay area is not taken in any transient or accident analysis in the Updated FSAR, since bounding analyses are performed for large breaks, such as main steam line breaks (MSLBs). Instead, the area is monitored for MSL leakage and steps are taken to correct the condition if a small MSL leak (i.e., 1 to 10% of rated steam flow) is detected. This preserves the initial conditions assumed in the design basis accident and transient analyses. Should a large MSL break occur, the main steam isolation valves will automatically close based on a Main Steam Line Flow - High signal. The redundancy of channels provided for all essential variables provides a high probability that whenever an essential variable exceeds the isolation setting, the Primary Containment Isolation System (PCIS) initiates isolation. In the unlikely event that all channels for one essential variable in one trip system fail in such a way that a system trip does not occur, the system could still respond properly as other monitored variables exceed their isolation settings. Thus, the proposed activity does not impact the existing calculated dose for any accident or transient analysis and the analyses for MSLBs remain bounding for small MSL leaks. Therefore, neither the consequences of an accident nor the consequences of a malfunction of a structure, system, or component (SSC) important to safety, as previously evaluated in the Updated FSAR, have a more than minimal increase as a result of this change.

The PCIS cannot initiate an automatic trip due to alarm signals generated by the recorders. No failure of a recorder will affect other plant safety systems. No failure of a recorder, or common cause failure of multiple recorders, will initiate a plant event, reactor trip, or turbine trip. In accordance with station procedures, operators will confirm there is a MSL leak prior to manually initiating a plant shutdown.

In accordance with NRC RIS 2002-22, Supplement 1, and NEI 96-07, Appendix D, Rev. 1, a qualitative assessment, S&L Report No. SL-016320, was included in the engineering/technical information supporting the proposed activity. The qualitative assessment considered system design attributes, quality of the design processes employed, and operating experience of the proposed equipment and concluded that the failure likelihood introduced by the modified SSCs is sufficiently low. Therefore, there is a no more than minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the Updated FSAR. Additionally, the proposed activity does not increase the frequency of occurrence of an accident, create the possibility for an accident of a different type, nor create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the Updated FSAR.

Finally, the proposed activity does not have any impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment. No methods of evaluation, as described in the Updated FSAR and used in establishing the design bases or in the safety analyses, are affected by the proposed activity.

Activity: SNC1183620

Title: U2 Generator, Main Transformer, UAT Protective Relay Upgrades

10 CFR 50.59 Evaluation Summary:

The protective relays (for conditions such as differential protection, overcurrent, ground, overexcitation, negative sequence, & loss of excitation) associated with the main generator, main transformer, and unit auxiliary transformers (UATs) 2A and 2B are electromechanical relays without redundancy. These relays receive inputs from current transformers (CTs) & potential transformers (PTs) that also lack redundancy. A failure of these single point vulnerabilities (SPVs) would result in a generator trip and reactor scram. The new replacement relays are multifunctional devices; therefore, one digital relay will replace multiple existing electromechanical relays.

The proposed activity replaces the electromechanical relays, except the lockout and fault pressure relays, with digital, microprocessor-based Schweitzer Engineering Laboratories (SEL) relays in the main control room (MCR). The relays will be configured to provide tripping in a two-out-of-three voting logic scheme such that for a trip actuation to occur, two relays will need to actuate on the same fault condition. A trip actuation is also possible if a loss of power, software failure, or hardware failure on a relay occurs, and a fault occurs on the relay in series with that relay. A contact in parallel with one of that relay's fault trip contacts will close on any of these conditions and is in series with the trip contact of one of the alternate relays. Alarm functions are maintained and configured in a one-out-of-three logic.

A qualitative assessment (performed in accordance with Regulatory Issue Summary (RIS) 2002-22 and RIS 2002-22, Supplement 1) of factors involving the design attributes of the digital upgrade, the quality of the design process used in developing the upgrade, and operating experience of the equipment with similar upgrades concluded that the digital upgrade will exhibit a sufficiently low likelihood of failure.

A turbine trip with bypass and a turbine trip without bypass are the two Updated FSAR (UFSAR) described accidents that are associated with the control system affected by this modification. A turbine trip is an evaluated anticipated operational occurrence (AOO), which is defined as a condition of normal operation expected to occur one or more times during the life of the plant per UFSAR Chapter 15.

On the basis of the qualitative assessment (SNC1183620QUAL) and the review of the changes to the main generator, main transformer, UAT 2A, and UAT 2B protective relaying, it was concluded that the proposed activity does NOT result in more than a minimal increase in frequency of occurrence of an accident previously evaluated in the UFSAR, or in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the UFSAR.

Although a failure of the SEL relays can cause the previously evaluated accidents above, the effectiveness of actions described or assumed in a turbine trip event are unchanged and the existing failure effects remain bounding (Ref. 4). Therefore, it is concluded that existing radiological dose consequences remain bounding, such that the proposed activity does NOT result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR, or in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The results of the qualitative assessment indicated that the likelihood of a SCC failure was sufficiently low – that is, comparable to the likelihood of failures that are not considered in the UFSAR. On that basis, it was concluded that the proposed activity does NOT create the possibility for an accident of a different type or a malfunction with a different result than previously evaluated in the UFSAR.

Since failures in the protective relay systems can cause the previously evaluated accidents identified above, and since the failure modes and effects analysis (FMEA) performed in support of the design modification determined that the existing failure effects remain bounding, the existing safety analysis, which demonstrates that design basis limits for a fission product barrier are not exceeded, remains bounding. Therefore, it was concluded that the proposed activity does NOT result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.

Activity: SNC1348498

Title: U1 RFP Min. Flow Valve Upgrade

10 CFR 50.59 Evaluation Summary:

The existing operation of the minimum (min.) flow valves includes the normally open, manually controlled min. flow isolation motor operated valves (MOVs) (1N21F040A/B), and the fail open, flow controlled min. flow hydraulically operated valves (1N21F200A/B). During normal operation, these valves ensure there is an acceptable minimum flow through the reactor feed pump by creating a bypass pathway to the condenser if the feedwater or main steam is isolated. With the min. flow isolation valve open, inadvertent failures of the min. flow control valve that cause it to open could lead to a recirculation flow runback.

Due to recent reliability issues with the operation of the min. flow control valve, the min. flow isolation valves are procedurally closed during operation to prevent a recirculation flow runback if the control valve opens inadvertently at power.

This modification replaces the hydraulically operated min. flow control valves with air operated DRAG valves with improved reliability which allow the isolation valves to be controlled based on flow conditions without increasing the likelihood of a recirculation flow runback at power. In both the existing and modified cases, the control valve is automatically controlled.

The min. flow isolation valve is currently manually controlled, and is modified to be automatically controlled to open based on low feedwater flow. If the automatic function fails to open the min flow valve, it can also be manually actuated in the same manner as the original configuration.

If the isolation valve fails to open, the min. flow line will not allow the required flow through the pump and will allow it to deadhead and the pump may be damaged due to high pressure and low flow. The only credible scenario for potential damage to the reactor feed pump (RFP) due to high system pressure requiring the operation of the min. flow line is if the system is being isolated via the feedwater or the steam lines. In this scenario, the RFP and associated systems (including min. flow) are no longer required to support the design function of providing feedwater to the reactor, and the min. flow line is only needed for the commercial longevity of the RFP itself.

The RFP min. flow isolation valves and the change in controls from manual to automatic do not introduce any new failure modes that are initiators of an accident, and the min. flow system is not credited for mitigation of an accident, therefore the proposed activity does not increase the frequency of occurrence or consequences of an accident previously evaluated in the Updated FSAR. The proposed activity is demonstrated to be more reliable than the existing system, and therefore does not increase the likelihood of occurrence or consequences of a malfunction of an SSC important to safety previously evaluated in the Updated FSAR. The proposed activity does not introduce any new failure modes or system interactions that could be an accident initiator of a different type than any previously evaluated in the Updated FSAR. The effects from the new failure modes of the automatic controls for the min. flow isolation valve are bounded by existing effects, and therefore does not create the possibility for a malfunction with a different result than previously evaluated. The min. flow isolation valve does not impact the ability of the reactor feed pump and feedwater system to provide water to the reactor when required and therefore, the proposed activity does not impact the integrity of the fuel cladding, reactor coolant pressure boundary, or containment. The proposed activity does not involve a method of evaluation; therefore, this activity 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: SNC1462366

Title: MET Tower Equipment Upgrade

10 CFR 50.59 Evaluation Summary:

The activity upgrades the Hatch Meteorological Data Collection System (MDCS). Equipment being upgraded includes the instrumentation on both the primary and backup meteorological towers, the data collection and processing equipment, the communications equipment between the meteorological towers and the simulator building, and the equipment that interfaces to the plant Meteorological Information and Dose Assessment System (MIDAS) and Safety Parameter Display System (SPDS).

This MDCS upgrade will install new modern equipment on both the primary and backup meteorological towers. The wind speed, wind direction, temperature, dew point, and precipitation sensors are replaced.

The MDCS does not control any plant SSC. The MDCS is not able to initiate any of the analyzed anticipated operational occurrences (AOOs) or Accidents evaluated in the UFSAR. Therefore, the proposed activity does not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the Updated FSAR.

The MDCS is not safety-related, and is not considered important to safety. No safety-related system or SSC is dependent upon or supported by the MDCS. The upgraded MDCS design, meteorological instrumentation, and data processing equipment, is at least as reliable as the existing design. Therefore, the proposed activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the Updated FSAR.

The MDCS does not affect any SSC designed to prevent or mitigate radiological releases during accidents evaluated in the UFSAR. Therefore, this design change does not result in more than a minimal increase in the consequences of an accident previously evaluated in the Updated FSAR.

The MDCS is not safety-related, and is not considered important to safety. No safety-related system or SSC is dependent upon or supported by the MDCS. The MDCS does not control any plant SSC. Therefore, this design change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety previously evaluated in the Updated FSAR.

The MDCS does not control any plant SSC. The MDCS cannot initiate any plant transient or event. Therefore, this design change does not create the possibility for an accident of a different type than any previously evaluated in the Updated FSAR.

The MDCS is not safety-related, and is not considered important to safety. No safety-related system or SSC is dependent upon or supported by the MDCS. The MDCS does not control any plant SSC. Therefore, this design change does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the Updated FSAR.

This activity upgrades the MDCS. Existing meteorological instrumentation and data processing equipment is replaced with new equipment and interfaced to the plant. No aspect of this activity has any effect on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment.

Lastly, no UFSAR described method of evaluation is affected by any aspect of this design activity.

Activity: SNC1480654

Title: Hatch AST Dose Calculation Conversion

10 CFR 50.59 Evaluation Summary:

The activity consists of converting all the dose consequence analyses of record from LocaDose to RADTRAD and revising some input parameters in the analyses to be consistent with the current plant configuration. The activity does not result in a more than minimal increase in consequences.

The input parameter changes are made as part of deterministic, prescribed radiological accidents to conservatively evaluate the dose consequences. Therefore, the activity is not an accident or malfunction initiator nor does it have an impact on the fission product barriers.

The change from LocaDose to RADTRAD does not represent a departure from a method of evaluation described in the FSAR because RADTRAD has been demonstrated to produce results that are essentially the same as LocaDose for the same set of inputs, and RADTRAD has been approved by the NRC for the intended application at numerous other plants, including Farley and Vogtle.

Activity: TE1109250

Title: Update of FSAR Section 15.3.3.4.2.2 for Revised Doses Documented in BH2-M-V999-0063

10 CFR 50.59 Evaluation Summary:

The activity involves an update to the FSAR described Loss-of-Coolant Accident (LOCA) dose consequence analysis to correct an input parameter. The input parameter is an assumed bypass leakage flow rate that is postulated coincident with the accident initiation to conservatively evaluate the dose consequences.

The activity to update the LOCA radiological consequence analysis to correct a bypass leakage flow rate in the offsite dose models results in a less than minimal increase in total FSAR reported dose at the EAB (i.e., NRC notification not required prior to implementing change). The input parameter changed is an assumed bypass leakage flow rate that is postulated coincident with the accident initiation to conservatively evaluate the dose consequences. Therefore, it is not an accident or malfunction initiator, nor does it represent a change in the evaluation methodology.