Southern Nuclear

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Joseph M. Farley Nuclear Plant - Units 1 & 2
Revision 29 to the Updated Final Safety Analysis Report, Updated NFPA 805 Fire Protection Program Design Basis Document, Technical Specification Bases Changes, Technical Requirements Manual 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 29 to the Joseph M. Farley Nuclear Plant (FNP) Updated Final Safety Analysis Report (UFSAR). The revised FNP UFSAR pages, indicated as Revision 29, reflect changes through March 31, 2020.

The FNP Technical Specifications, Section 5.5.14, "Technical Specifications (TS) Bases Control Program," provides for changes to the Bases without prior Nuclear Regulatory Commission (NRC) approval. In addition, TS Section 5.5.14 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 5.5.14, SNC hereby submits a complete copy of the FNP TS Bases. The revised FNP TS Bases pages, indicated as Revision 98, reflect changes to the TS Bases through March 31, 2020.

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 FNP UFSAR, the FNP UFSAR reference drawings, the TS Bases, the Technical Requirements Manual (TRM), and the National Fire Protection Association (NFPA) 805 Fire Protection Program Design Basis Document are hereby submitted on CD-ROM in portable document format (PDF). The revised FNP TRM pages, indicated as Version 46.0, reflect changes to the TRM through March 31, 2020. The updated NFPA 805 Fire Protection Program Design Basis Document, Version 6.0, also reflects changes through March 31, 2020.

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.

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In accordance with NEI 99-04, "Guidelines for Managing NRC Commitment Changes," Revision 0, SNC hereby submits a Revised NRC Commitments Report containing the original commitment, the revised commitment, and the justification for the change.

SNC conducted a review of FNP 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-ROM (Enclosure 2). Enclosure 3 provides the 10 CFR 50.59 Summary Report. Enclosure 4 provides the Revised NRC Commitments 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 28th day of April 2020.

Respectfully submitted,

Cheryl Gayheart
Regulatory Affairs Director

CAG/TLE/scm

Enclosures:
1. CD-ROM Table of Contents
2. CD-ROM
3. 10 CFR 50.59 Summary Report
4. Revised NRC Commitments Report

cc: Regional Administrator, Region II (w/o enclosures)
Senior NRR Project Manager – Farley (w/o enclosures)
Senior Resident Inspector – Farley (w/o enclosures)
INPO Emergency Management Manager (Enclosure 2, CD-ROM, only)
RType: CFA04.054

NL-20-0442

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Enclosure 1
CD-ROM Table of Contents

SEQ	FILENAME CONTENT	EXTENSION
001	FARLEY FSAR_EF PG LST, TOC, CH1, CH2-PRT 1 Effective Page List Table of Contents Chapter 1 Chapter 2 (Part 1) § 2.1 – 2.5 Appendix 2A & 2B	.pdf
002	FARLEY FSAR_CH2-PRT 2 Chapter 2 (Part 2) Appendix 2B Figures (thru 2B5A-4)	.pdf
003	FARLEY FSAR_CH2-PRT 3 Chapter 2 (Part 3) Appendix 2B Figures Continued (2B5B-1 thru end of chapter)	.pdf
004	FARLEY FSAR_CH3-PRT 1 Chapter 3 § 3.1 – 3.11 Appendix 3A – 3J (up to 3J figures)	.pdf
005	FARLEY FSAR_CH3-PRT 2 Chapter 3 (Part 2) Appendix 3J (Figures) – Appendix 3M	.pdf
006	FARLEY FSAR_CH4, CH5 Chapter 4 Chapter 5	.pdf
007	FARLEY FSAR_CH6, CH7 Chapter 6 Chapter 7	.pdf
008	FARLEY FSAR_CH8, CH9, CH10 Chapter 8 Chapter 9 Chapter 10	.pdf
009	FARLEY FSAR_CH11, CH12, CH13, CH14, CH15, CH16, CH17, CH18 Chapter 11 Chapter 12 Chapter 13 Chapter 14 Chapter 15 Chapter 16 Chapter 17 Chapter 18	.pdf

SEQ	FILENAME CONTENT	EXTENSION
010	FARLEY FSAR_TECH SPECS BASES Technical Specifications Bases	.pdf
011	FARLEY FSAR_TECHNICAL REQUIREMENTS MANUAL	.pdf
012	FARLEY FSAR_NFPA 805 FIRE PROTECTION PROGRAM	.pdf
013	FARLEY FSAR REF DWGS A177040 sh 360 thru A177048 sh 325	.pdf
014	FARLEY FSAR REF DWGS A177048 sh 326 thru A177048 sh 568	.pdf
015	FARLEY FSAR REF DWGS A207048 sh 1 thru A207048 sh 300	.pdf
016	FARLEY FSAR REF DWGS A207048 sh 301 thru A207048 sh 568	.pdf
017	FARLEY FSAR REF DWGS A508650 sh 1 thru D175012 sh 1	.pdf
018	FARLEY FSAR REF DWGS D175014 sh 1 thru D177944 sh 1	.pdf
019	FARLEY FSAR REF DWGS D181620 sh 1 thru U611138	.pdf
020	REVISION 29 NOMENCLATURE	.doc

NL-20-0442

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Revision 29 to the Updated Final Safety Analysis Report, Updated NFPA 805 Fire Protection Program Design Basis Document, Technical Specification Bases Changes, Technical Requirements Manual Changes, 10 CFR 50.59 Summary Report, and Revised NRC Commitments Report

Enclosure 2
CD-ROM

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Enclosure 3
10 CFR 50.59 Summary Report

10 CFR 50.59 Summary Report

Activity: Design Change SNC706480

Title: Radiation Monitor Replacement Group (Unit 1)

10 CFR 50.59 Evaluation Summary:

This activity replaces skid mounted containment exhaust flow gas radiation monitors R24A/B, spent fuel pool (SFP) exhaust flow gas monitors R25A/B, and control room makeup air inlet gas monitors R35A/B with in-vent monitors and digital controllers. The new monitors and supporting equipment will perform the same basic functions as the existing monitors.

The R25A/B monitors are evaluated as a result of an increase in the time to isolate the SFP exhaust damper after a fuel handling accident (FHA). R24A/B, R25A/B, and R35A/B are evaluated as a result of a digital upgrade.

The replacement radiation monitors and associated equipment act to limit a release after an FHA and are not accident initiators. Therefore, they do not result in more than a minimal increase the frequency of occurrence of an accident previously evaluated in Updated Final Safety Analysis Report (UFSAR).

The monitors and associated equipment are compatible with the installed environment and they do not have an adverse impact on the installed environment. The software of the remote display units (RDUs) and local processing display units (LPDUs) has been developed in accordance with industry standards and regulatory guidance. An RDU or LPDU software failure does not cause a fuel handling accident. Common cause failures have been evaluated and determined that a failure of the digital controllers would have the same impact as a failure of the existing equipment. Therefore, this activity does not result in a more than minimal increase in the likelihood of occurrence of a malfunction of the structure, system, or component (SSC) important to safety previously evaluated in the UFSAR.

The monitors and associated equipment do not increase the burdens or constraints on the operators' ability to adequately respond to an accident and a system failure has the same results as a system failure of the existing equipment. Therefore, this activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in the UFSAR.

The monitors and associated equipment provide local and main control room (MCR) alarms and provide input to the plant computer. Plant annunciator and the plant computer are non-safety related. Failure of the monitoring equipment to provide inputs to the plant annunciators or plant computer do not result in an increase in the consequence of a malfunction important to safety. Therefore, this activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The UFSAR does not discuss failure modes for the radiation monitoring system. A failure of the replacement monitors has the same results as the existing monitors and will not cause an accident of a different type evaluated in the UFSAR or change the basis for the most limiting scenario previously considered in the UFSAR. Therefore, this activity does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

The monitors and associated equipment do not combine previously separate functions into one digital device. The result of a power loss and restoration of power will not create the possibility of a malfunction with a different result than the malfunctions considered in the UFSAR. The failure modes introduced by the human-system interface (HMI) are loss of user interface and incorrect data displayed on the user interface. Both failure modes will result in degraded monitor functionalities without loss of monitor principal functions. The UFSAR does not directly address the failure modes for the radiation monitoring system. Therefore, none of the failure modes for the upgraded system will result in effects not bounded by the results previously considered in the UFSAR. Therefore, this activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

The R24A/B monitors, located in the Unit 1 auxiliary building, act to limit the radioactive releases associated with an FHA in the refueling canal, inside containment, by automatically isolating the containment purge and exhaust lines. The new R24A/B monitors perform the same functions as the existing monitors and do not adversely affect the integrity of containment. These monitors are not accident initiators since they are not actively involved in the movement of fission product barriers (fuel bundles) and therefore, have no impact on numerical values used to determine the integrity of the fission product barriers identified in the UFSAR. They have no direct or indirect impact on the reactor cooling pressure boundary. The R25A/B & R35A/B monitors, located in the Unit 1 auxiliary building, have no direct or indirect functions related to the movement of fission product barriers (fuel bundles), the reactor cooling pressure boundary, or containment isolation. Therefore, this activity does not have any impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment.

Activity: Design Change SNC706481

Title: Radiation Monitor Replacement Group (Unit 2)

10 CFR 50.59 Evaluation Summary:

This activity replaces skid mounted containment exhaust flow gas radiation monitors R24A/B and spent fuel pool (SFP) exhaust flow gas monitors R25A/B with in-vent monitors and digital controllers. The new monitors and supporting equipment will perform the same basic functions as the existing monitors.

The R25A/B monitors are evaluated as a result of an increase in the time to isolate the SFP exhaust damper after a fuel handling accident (FHA). R24A/B and R25A/B are evaluated as a result of a digital upgrade.

The replacement radiation monitors and associated equipment act to limit a release after an FHA and are not accident initiators. Therefore, they do not result in more than a minimal increase the frequency of occurrence of an accident previously evaluated in Updated Final Safety Analysis Report (UFSAR).

The monitors and associated equipment are compatible with the installed environment and they do not have an adverse impact on the installed environment. The software of the remote display units (RDUs) and local processing display units (LPDUs) has been developed in accordance with industry standards and regulatory guidance. An RDU or LPDU software failure does not cause a fuel handling accident. Common cause failures have been evaluated and determined that a failure of the digital controllers would have the same impact as a failure of the existing equipment. Therefore, this activity does not result in a more than minimal increase in the likelihood of occurrence of a malfunction of the structure, system, or component (SSC) important to safety previously evaluated in the UFSAR.

The monitors and associated equipment do not increase the burdens or constraints on the operators' ability to adequately respond to an accident and a system failure has the same results as a system failure of the existing equipment. Therefore, this activity does not result in more than a minimal increase in the consequences of an accident previously evaluated in UFSAR.

The monitors and associated equipment provide local and main control room (MCR) alarms and provide input to the plant computer. Plant annunciator and the plant computer are non-safety related. Failure of the monitoring equipment to provide inputs to the plant annunciators or plant computer do not result in an increase in the consequence of a malfunction important to safety. Therefore, this activity does not result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

The UFSAR does not discuss failure modes for the radiation monitoring system. A failure of the replacement monitors has the same results as the existing monitors and will not cause an accident of a different type evaluated in the UFSAR or change the basis for the most limiting scenario previously considered in the UFSAR. Therefore, this activity does not create the possibility for an accident of a different type than any previously evaluated in the UFSAR.

The monitors and associated equipment do not combine previously separate functions into one digital device. The result of a power loss and restoration of power will not create the possibility of a malfunction with a different result than the malfunctions considered in the UFSAR. The failure modes introduced by the human-system interface (HMI) are loss of user interface and incorrect data displayed on the user interface. Both failure modes will result in degraded monitor functionalities without loss of monitor principal functions. The UFSAR does not directly address the failure modes for the radiation monitoring system. Therefore, none of the failure modes for the upgraded system will result in effects not bounded by the results previously considered in the UFSAR. Therefore, this activity does not create the possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the UFSAR.

The R24A/B monitors, located in the Unit 2 auxiliary building, act to limit the radioactive releases associated with an FHA in the refueling canal, inside containment, by automatically isolating the containment purge and exhaust lines. The new R24A/B monitors perform the same functions as the existing monitors and do not adversely affect the integrity of containment. These monitors are not accident initiators since they are not actively involved in the movement of fission product barriers (fuel bundles) and therefore, have no impact on numerical values used to determine the integrity of the fission product barriers identified in the UFSAR. They have no direct or indirect impact on the reactor cooling pressure boundary. The R25A/B monitors, located in the Unit 2 auxiliary building, have no direct or indirect functions related to the movement of fission product barriers (fuel bundles), the reactor cooling pressure boundary, or containment isolation. Therefore, this activity does not have any impact on the integrity of the fuel cladding, reactor coolant pressure boundary, or containment.

Activity: Design Change SNC900664

Title: Unit 1 Open Phase Trip Mode Enable

10 CFR 50.59 Evaluation Summary:

The activity enables the trip mode of the open phase protection (OPP) systems installed by DCPs SNC679759 and SNC679760. The activity adds a new function to the OPP panel and existing protective relaying for each SAT 1A and SAT 1B. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of each transformer. Modification of the OPP system at the Farley 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. 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: Design Change SNC900665

Title: Unit 2 Open Phase Trip Mode Enable

10 CFR 50.59 Evaluation Summary:

The activity enables the trip mode of the open phase protection (OPP) systems installed by DCPs SNC679761 and SNC679762. The activity adds a new function to the OPP panel and existing protective relaying for each SAT 2A and SAT 2B. The individual OPP panel isolates the monitored transformer on detection of a loss of phase on the high side (upstream) of each transformer. Modification of the OPP system at the Farley 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. 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: Technical Evaluation 1040425

Title: Incorporate AFW Analysis Contained in Westinghouse Letter ALA-14-12 into FSAR

10 CFR 50.59 Evaluation Summary:

The following sections of the final safety analysis report (FSAR) are updated per Westinghouse Letter ALA-14-12, "Transmittal of LTR-TA-13-159, "Farley Units 1 and 2 Feedline Break Analysis Results and FSAR Markups for a Reduced AFW Flow Rate (105 gpm)"":

- Revise FSAR Section 15.4.2.2.2 Case B
- Revise FSAR Table 15.4-5 (sheet 2 of 4)
- Revise FSAR Figure 15.4-32 (sheets 1 through 4)

Incorporate the results of Westinghouse Letter ALA-14-12 into Farley FSAR Section 15.4.2.2. This will replace the current Auxiliary Feedwater (AFW) flow value of 150 gal/min with the current Analysis of Record value of 105 gal/min as well as the affected figures and tables as provided in LTR-TA-13-159 FSAR markups.

This change involves an update to the FSAR to reflect the current analysis of record as it pertains to an AFW flow rate of 105 gpm before faulted steam generator isolation with 30-minute operator action time in order to maximize the AFW flow rate margin available to the plant. No physical changes will be required to any equipment in the plant. The updated flow parameters represent the minimum required AFW flow rate while still satisfying all MFLB requirements as described in Farley FSAR Section 15.4.2.2.

NL-20-0442

Joseph M. Farley Nuclear Plant - Units 1 & 2
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Enclosure 4
Revised NRC Commitments Report

Original Commitment: SNC500323 - Changes to the Tech Specs Related to Turbine Valve Testing and LCO for Unit 1

APCO believes their turbine valve maintenance, calibration, testing and inspection interval (greater than one month) is sufficient to provide assurance of valve operation on demand. Their program encompasses an intensive and effective turbine valve maintenance program to preempt valve failures coupled with a periodic testing, calibration, and a thorough inspection of valve internals by valve disassembly on alternate refueling outages. The internal inspection would cover at least one of each type valve installed on the Farley Unit 1 and 2 Turbine each outage or at least two of each type valve installed on the Farley Unit 1 and 2 Turbine every other outage on each unit. In the event a valve problem is discovered, all turbine valves of that type would be disassembled and the problem corrected.

Revised Commitment: APCO believes their turbine valve maintenance, calibration, testing and inspection interval is sufficient to provide assurance of valve operation on demand. Their program is described and tracked in the Turbine Overspeed Reliability Assurance Program (TORAP) and encompasses an intensive and effective turbine valve maintenance program to preempt valve failures coupled with a periodic testing, calibration, and a thorough inspection of valve internals by valve disassembly. In the event a valve problem is discovered, all turbine valves of that type would be disassembled and the problem corrected.

Justification for Change:

A review of the Main Turbine Intercept and Reheat Stop Valve performance history, refurbishment reports and industry OE for Farley Unit 1 and Farley Unit 2 warrant a frequency extension.

Due Date: January 27, 1984 (implemented)

Original Commitment: SNC500326 - Changes to the Tech Specs Related to Turbine Valve Testing and LCO for Unit 1

The testing program includes testing of the turbine valves and the turbine overspeed protection system. Testing is performed during each turbine startup, unless tested within the previous seven (7) days, including start-up after each refueling outage. The testing program includes a complete test of all turbine valves on an approximate interval of four (4) months.

Revised Commitment: The testing program includes testing of the turbine valves and the turbine overspeed protection system. Testing is performed during each turbine startup, unless tested within the previous seven (7) days, including start-up after each refueling outage. The testing program includes a complete test of all turbine throttle and governor valves on an approximate interval of six (6) months and a complete test of all turbine reheat stop and intercept valves on an approximate interval of eighteen (18) months.

Justification for Change:

A review of the Main Turbine Intercept and Reheat Stop Valve performance history, refurbishment reports and industry OE for Farley Unit 1 and Farley Unit 2 warrant a frequency extension.

Due Date: February 12, 2019 (closed)

Original Commitment: SNC500332 - Turbine Valve Technical Specification Change Request

The testing program addresses the turbine valves and the turbine overspeed protection system. Testing is performed during each turbine startup, unless tested within the previous seven (7) days, including start-up after each refueling outage. The testing program includes a complete test of all turbine valves on an approximate interval of four (4) months. Also, the turbine is subjected to an actual overspeed trip test every refueling outage or following major maintenance on the turbine.

Revised Commitment: The testing program addresses the turbine valves and the turbine overspeed protection system. Testing is performed during each turbine startup, unless tested within the previous seven (7) days, including start-up after each refueling outage. The testing program includes a complete test of all turbine throttle and governor valves on an approximate interval of six (6) months and a complete test of all reheat stop and intercept valves on an approximate interval of eighteen (18) months. Also, the turbine is subjected to an actual overspeed trip test every refueling outage or following major maintenance on the turbine.

Justification for Change:

A review of the Main Turbine Intercept and Reheat Stop Valve performance history, refurbishment reports and industry OE for Farley Unit 1 and Farley Unit 2 warrant a frequency extension.

Due Date: February 12, 2019 (closed)

Original Commitment: SNC500334 - Turbine Valve Technical Specification Change Request

The turbine valve maintenance program includes inspection and maintenance of all throttle, governor, intercept, and reheat stop valves at least every 72 months.

Revised Commitment: The turbine valve maintenance program includes inspection and maintenance of all throttle and governor valves at least every 72 months and the intercept and reheat stop valves every 126 months.

Justification for Change:

A review of the Main Turbine Intercept and Reheat Stop Valve performance history, refurbishment reports and industry OE for Farley Unit 1 and Farley Unit 2 warrant a frequency extension.

Due Date: October 6, 1983 (implemented)

Original Commitment: SNC1001846 – Commitments form FNP Alternative Source Term

1. Administrative controls will be established to ensure appropriate personnel are aware of the open status of the penetration flow path(s) during core alterations or movement of irradiated fuel assemblies within the containment.
2. Existing administrative controls for open containment airlock doors will be expanded to ensure specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.
3. With the Personnel Airlock open during fuel handling operations or core alterations, the Containment Purge System will be in operation.
4. In the event of an FHA, the containment will be evacuated and the Personnel Airlock will be closed within 30 minutes of detection of the accident.
5. In the event of an FHA, Control Room occupants will use the secondary door to the Control Room for ingress and egress.

Revised Commitment:

1. Administrative controls will be established to ensure appropriate personnel are aware of the open status of the penetration flow path(s) during core alterations or movement of <u>recently</u> irradiated fuel assemblies within the containment.
2. Existing administrative controls for open containment airlock doors will be expanded to ensure specified individuals are designated and readily available to isolate any open penetration flow path(s) in the event of an FHA inside containment.
3. In the event of an FHA, the containment will be evacuated and the Personnel Airlock will be closed within 30 minutes of detection of the accident.
4. In the event of an FHA, Control Room occupants will use the secondary door to the Control Room for ingress and egress.

Justification for Change:

Modification of Item 1:

The NRC approval of TSTF-51 revised the Applicability of TS 3.9.3, Containment Penetrations, such that this TS is only applicable during movement of recently irradiated fuel assemblies within containment. The previous TS applicability was during Core Alterations and during movement irradiated fuel assemblies within the containment. The SNC LAR requesting to adopt

TSTF-51 contained the revised Commitment description to make the applicability consistent with the provisions of TSTF-51.

Deletion of Previous Item 3:

The original commitment was made to alleviate the potential (during a fuel handling accident in containment cf. FSAR section 15.4.5) for air leakage from the containment to the auxiliary building through the personnel airlock. However, evaluations included in the analysis of record have eliminated the need for this precaution to assure control room doses meet GDC 19 limits.

NRC required SNC to evaluate an accident condition where the Equipment Hatch was closed and the personnel Airlock was open in their RAIs on the AST Submittal (NL-16-0388). SNC did this and provided a response to the RAI. SNC performed sensitivity studies and determined a reasonable, but worst case, flow rate through the PAL, given the auxiliary building HVAC system was operating and the PAL was open. The result of these studies created a maximized dose to the control room operators during the accident from a potentially contaminated auxiliary building. GDC 19 requirements are shown in the calculation to be met.

Given that the fuel handling accident analysis uses conservative methods to calculate the dose in the control room from the leakage out of the PAL, and given that that part of the calculation does not assume that the containment purge system is on, the analysis of record bounds any reasonable condition for which we would have required the Containment Purge System to be operating while performing fuel movement or core alterations.

Therefore, the commitment is no longer necessary. In fact having the purge system on during fuel movement seems a bit counter-intuitive given that such an operation would assure that activity released in the fuel handling accident inside containment would be discharged to the environment. This would lead to unnecessary doses to the public and the control room operators.

Due Date: March 17, 2019 (implemented)