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

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Docket Nos.: 50-348
50-364

NL-21-0912

U.S. Nuclear Regulatory Commission
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Washington, DC 20555-0001

Joseph M. Farley Nuclear Plant - Units 1 & 2
Revision 30 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 30 to the Joseph M. Farley Nuclear Plant (FNP) Updated Final Safety Analysis Report (UFSAR). The revised FNP UFSAR, indicated as Revision 30, reflects changes through October 15, 2021.

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, indicated as Revision 106, reflects changes to the TS Bases through October 15, 2021.

The revised FNP TRM, indicated as Version 54.0, reflects changes to the TRM through October 15, 2021. The updated National Fire Protection Association (NFPA) 805 Fire Protection Program Design Basis Document, Version 6.0, also reflects changes through October 15, 2021.

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 NFPA 805 Fire Protection Program Design Basis Document are hereby submitted on Optical Disc in portable document format (PDF).

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.

AD53
NRR

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 (April 1, 2020 to October 15, 2021).

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 Optical Discs (Enclosure 2, public version with security-related information redacted; Enclosure 3, non-public version). Enclosure 3 contains security-related information within the Farley UFSAR. 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 NRC commitments. If you have any questions, please contact Ryan Joyce at (205) 992-6468.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 28th day of October 2021.

Respectfully submitted,

Cheryl Gayheart
Regulatory Affairs Director

CAG/tle

Enclosures:
1. Optical Discs Table of Contents
2. Optical Disc – Public
3. Optical Disc – Nonpublic (Withhold from public disclosure in accordance with 10 CFR 2.390(d)(1))
4. 10 CFR 50.59 Summary Report

cc: Regional Administrator, Region II (w/o enclosures)
Senior NRR Project Manager – Farley (w/o enclosures)
Senior Resident Inspector – Farley (w/o enclosures)
RType: CFA04.054

NL-21-0912

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Enclosure 1
Optical Discs Table of Contents

SEQ	FILENAME CONTENT FOR **PUBLIC ENCLOSURE 2***	EXTENSION
001	EPL-TOC_P Effective Page List and Table of Contents	.pdf
002	Chapter 1_P	.pdf
003	Chapter 2_P	.pdf
004	Chapter 3_P	.pdf
005	Chapter 4_P	.pdf
006	Chapter 5_P	.pdf
007	Chapter 6_P	.pdf
008	Chapter 7_P	.pdf
009	Chapter 8_P	.pdf
010	Chapter 9_P	.pdf
011	Chapter 10_P	.pdf
012	Chapter 11_P	.pdf
013	Chapter 12_P	.pdf
014	Chapter 13_P	.pdf
015	Chapter 14_P	.pdf
016	Chapter 15_P	.pdf
017	Chapter 16_P	.pdf
018	Chapter 17_P	.pdf
019	Chapter 18_P	.pdf
027*	TECHNICAL SPECIFICATIONS BASES_P	.pdf
028*	TECHNICAL REQUIREMENTS MANUAL_P	.pdf
029*	NFPA 805 FIRE PROTECTION PROGRAM_P	.pdf

* Files 020 through 026 are completely non-public and thus not provided on the Enclosure 2 public optical disc. Files 027 through 029 are completely public and thus not provided on the Enclosure 3 non-public optical disc.

SEQ	FILENAME CONTENT FOR **NON-PUBLIC ENCLOSURE 3***	EXTENSION
001	EPL-TOC_NP Effective Page List and Table of Contents	.pdf
002	Chapter 1_NP	.pdf
003	Chapter 2_NP	.pdf
004	Chapter 3_NP	.pdf
005	Chapter 4_NP	.pdf
006	Chapter 5_NP	.pdf
007	Chapter 6_NP	.pdf
008	Chapter 7_NP	.pdf
009	Chapter 8_NP	.pdf
010	Chapter 9_NP	.pdf
011	Chapter 10_NP	.pdf
012	Chapter 11_NP	.pdf
013	Chapter 12_NP	.pdf
014	Chapter 13_NP	.pdf
015	Chapter 14_NP	.pdf
016	Chapter 15_NP	.pdf
017	Chapter 16_NP	.pdf
018	Chapter 17_NP	.pdf
019	Chapter 18_NP	.pdf
020*	FARLEY FSAR REF DWGS_NP A177040 sh 360 thru A177048 sh 325	.pdf
021*	FARLEY FSAR REF DWGS_NP A177048 sh 326 thru A177048 sh 568	.pdf
022*	FARLEY FSAR REF DWGS_NP A207048 sh 1 thru A207048 sh 300	.pdf

SEQ	FILENAME CONTENT FOR **NON-PUBLIC ENCLOSURE 3***	EXTENSION
023*	FARLEY FSAR REF DWGS_NP A207048 sh 301 thru A207048 sh 568	.pdf
024*	FARLEY FSAR REF DWGS_NP A508650 sh 1 thru D175012 sh 1	.pdf
025*	FARLEY FSAR REF DWGS_NP D175014 sh 1 thru D177944 sh 1	.pdf
026*	FARLEY FSAR REF DWGS_NP D181620 sh 1 thru U611138	.pdf

* Files 020 through 026 are completely non-public and thus not provided on the Enclosure 2 public optical disc. Files 027 through 029 are completely public and thus not provided on the Enclosure 3 non-public optical disc.

NL-21-0912

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Enclosure 2
Optical Disc – Public

NL-21-0912

Joseph M. Farley Nuclear Plant - Units 1 & 2
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Enclosure 3
Optical Disc – Nonpublic
(Withhold from public disclosure in accordance with 10 CFR 2.390(d)(1))

NL-21-0912

Joseph M. Farley Nuclear Plant - Units 1 & 2
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Enclosure 4
10 CFR 50.59 Summary Report

10 CFR 50.59 Summary Report

Activity: CR 10077925

Title: Down Classification of Spent Fuel Bridge Crane to Non-Nuclear Safety

10 CFR 50.59 Evaluation Summary:

The spent fuel pool (SFP) Bridge Crane safety classification and seismic classification, as given in Farley FSAR Table 3.2-1 are overly conservative. Consequently, this activity changes the Safety Classification of the SFP Bridge Crane to "NNS" (Non-Nuclear Safety) related and the Seismic Classification from Category I to "II with special considerations" in Table 3.2-1.

The evaluation questions were each addressed for the activity being performed and it was determined each of the questions can be answered "no". Therefore, this activity can be implemented without NRC prior approval.

Activity: RER SNC994691

Title: Evaluation of Impact on Doses Due to Increasing Peak Rod Burnup to 62 GWD/MTU

10 CFR 50.59 Evaluation Summary:

In 2006 the NRC staff concluded that the peak rod burnup limit may be increased to 62 GWD/MTU, provided the fuel design performance is analyzed with PAD 4.0 (ADAMS No. ML061420458). The Farley 1 & 2 fuel design performance is evaluated with PAD 4.0 per section 4.2.1.1.1 of the UFSAR.

The effect on the Farley 1 &2 UFSAR Chapter 15 dose consequences of increasing the peak rod burnup limit to 62 GWD/MTU has been evaluated as resulting in less than minimal increases in those doses.

Thus, the Farley 1 &2 UFSAR Chapter 15 dose consequences may be revised without prior NRC review and approval.

Activity: TE 1064469

Title: Evaluations Performed to Address Long-Term Cooling with Containment Sump Debris (GSI-191)

10 CFR 50.59 Evaluation Summary:

This activity is associated only with additional evaluations that were performed to support the safety analysis with respect to 1 0 CFR 50.46(b)(5), "Long-term cooling." These additional evaluations consider the possibility that debris generated during a loss of coolant accident (LOCA) inside the containment of pressurized water reactors (PWRs) could block the emergency core cooling system (ECCS) and containment spray system (CSS) flow paths that

are required for containment sump recirculation. To demonstrate that long-term cooling will be met following accidents that require recirculation of coolant from the containment sump, evaluations are performed to ensure that the containment sump can perform its design basis function, which is to provide a source of borated water to the ECCS and CSS during long-term recirculation. These additional evaluations demonstrate that meeting the strainer, downstream ex-vessel, and in-vessel debris limits ensures adequate long-term cooling as required by 10 CFR 50.46(b)(5).

The addition of a sump debris limits is being considered as a conservative change to an element of the current method of evaluation (MOE). As such, this activity is considered to be a change to an element of the method described in the UFSAR that produces results that are conservative. The revised analysis demonstrates that the ECCS, CSS and containment sump can perform their intended safety function to assure adequate long-term cooling in accordance with 10 CFR 50.46(b)(5). Therefore, this activity does not 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: TE 1066707

Title: Revise FSAR Polar Crane Main Hoist Load Block Safe Path Restrictions

10 CFR 50.59 Evaluation Summary:

The activity being implemented changes the way the polar crane main hoist load block is controlled when the core is exposed. Instead of prohibiting load block travel over the exposed core, the main hoist will be made inoperable when the load block is traveling over the exposed core, except during removal or installation of the vessel head or upper internals. This action will also be applied if the polar crane main hoist load block is traversed over open RCP hatches in modes 5 or 6.

Additionally, the FSAR is also being revised to eliminate the polar crane main hoist load block from being considered a load or heavy load in section 9.1.7.2 in accordance with current industry guidance.

The questions posed were each addressed for the activity being performed and it was determined each of the questions can be answered "NO". Therefore, this activity can be implemented without prior NRC approval.

Activity: DCP SNC876765, Version 2.0

Title: U1 Main Generator Exciter Switchgear Replacement

10 CFR 50.59 Evaluation Summary:

The 10 CFR 50.59 evaluation concludes that a license amendment is not required before the activity may be implemented. The proposed activity implements a digital modification. The

modification introduces new failure modes associated with implementing the digital control system (digital devices) that start, stop, and regulate main generator excitation. The introduction of failure modes due to digital modification are evaluated and all applicable 50.59 Evaluation questions are answered "No."

The nonsafety-related main generator excitation system has no accident mitigation design function; however, the excitation system is inherently relied upon for preventing an unnecessary trip of the main generator and turbine, and a resultant reactor trip. The excitation system does not impact any safety-related SSC.

The replacement ABB 6080 excitation control system has built-in redundancies and defense-in-depth for high availability. The ABB Reliability and Availability Report concludes high availability and low mean-time-between-failure for a fatal failure of the excitation system such that the frequency of occurrence of the main generator or turbine trip event is not more than minimally increased. The ABB excitation control systems have been used extensively in the nuclear power industry and have exhibited a low incidence of failure. Malfunctions of the excitation control system are limited to the nonsafety-related excitation system. New failure modes and malfunctions associated with the digital system (e.g., digital device or software failures, including common cause software errors) are bounded by the main generator or turbine trip event analyzed in the UFSAR and would still be classified as a Condition II event (Incident of Moderate Frequency). There is no impact on the radiological consequences evaluated in the UFSAR. No new malfunctions are identified with SSCs outside the excitation system. The proposed activity does not impact the integrity of any fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, or containment).

Activity: DCP SNC892053, Version 2.0

Title: U2 Main Generator Exciter Switchgear Replacement

10 CFR 50.59 Evaluation Summary:

The 10 CFR 50.59 evaluation concludes that a license amendment is not required before the activity may be implemented. The proposed activity implements a digital modification. The modification introduces new failure modes associated with implementing the digital control system (digital devices) that start, stop, and regulate main generator excitation. The introduction of failure modes due to digital modification are evaluated and all applicable 50.59 Evaluation questions are answered "No."

The nonsafety-related main generator excitation system has no accident mitigation design function; however, the excitation system is inherently relied upon for preventing an unnecessary trip of the main generator and turbine, and a resultant reactor trip. The excitation system does not impact any safety-related SSC.

The replacement ABB 6080 excitation control system has built-in redundancies and defense-in-depth for high availability. The ABB Reliability and Availability Report concludes high availability and low mean-time-between-failure for a fatal failure of the excitation system such that the

frequency of occurrence of the main generator or turbine trip event is not more than minimally increased. The ABB excitation control systems have been used extensively in the nuclear power industry and have exhibited a low incidence of failure. Malfunctions of the excitation control system are limited to the nonsafety-related excitation system. New failure modes and malfunctions associated with the digital system (e.g., digital device or software failures, including common cause software errors) are bounded by the main generator or turbine trip event analyzed in the UFSAR and would still be classified as a Condition II event (Incident of Moderate Frequency). There is no impact on the radiological consequences evaluated in the UFSAR. No new malfunctions are identified with SSCs outside the excitation system. The proposed activity does not impact the integrity of any fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, or containment).

Activity: DCP SNC1030354

Title: Radiation Monitoring System (RMS) – U2 Group 8B (RE15s)

10 CFR 50.59 Evaluation Summary:

This activity replaces existing monitors R15, R15B and R15C with new monitors R15A and R15B, and installs digital controllers. The replacement radiation monitors and associated equipment 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 effect 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. 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 new R15A and R15B 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 coolant pressure boundary.

Activity: DCP SNC1086636, Version 2.0

Title: Bypass Thermal Overloads (TOLs) for Unit 1 LOCA Motor Operated Valves (MOVs)

10 CFR 50.59 Evaluation Summary:

The proposed activity involves a design change to bypass thermal overloads for 17 MOVs. Thermal overload relays are installed in the motor control center (MCC) compartments of the respective MOVs and are used to protect the motor from excessive temperature by cutting off power to the motor. The MOVs listed in the scope of this design are required to perform a safety function during, and/or after, a loss of coolant accident (LOCA). A loss of offsite power (LOSP) followed by a LOCA will require these MOVs to be fully transferred to the emergency diesel generators (EDGs) and the (TOLs) are required to not trip prior to these loads being transferred to the EDGs.

By-passing the TOLs will be accomplished by adding a permanent jumper across the MCC terminal block for the overload contacts. This activity will ensure that these valves perform their safety-related function. Failure of the TOL to operate as designed would result in the valve continuing to stroke until the overcurrent/overtemperature condition prevents the motor from continuing to run. Likewise, bypassing the TOLs would result in the valve continuing to stroke

until the overcurrent/overtemperature condition prevents the motor from continuing to run. In both cases where the TOL is not bypassed but fails to operate as intended and the case where the TOL is bypassed, the effect of the failure on the system is not different to what has been previously evaluated.

Activity: DCP SNC1097052

Title: Revision to Farley Containment P/T Analysis to Incorporate Outstanding NSALs

10 CFR 50.59 Evaluation Summary:

The LOCA containment pressure/temperature analysis of record (AOR) for Farley Units 1 and 2 has been revised to incorporate three outstanding Nuclear Safety Advisory Letters (NSALs), NSAL-06-6, NSAL-11-5, and NSAL-14-2. The revised LOCA mass and energy releases lead to a slight increase in calculated peak containment pressure and temperature . The peak containment pressure, P_a, is included in the Technical Specifications (TS Section 5.5.17). Therefore, the increase in peak pressure calculated in the Farley Units 1 and 2 NSAL incorporation containment response analysis will be submitted to the NRC in an upcoming LAR.

Otherwise, design input changes made to the LOCA containment pressure/temperature analysis do not require prior NRC approval to implement. The input changes being evaluated here will require accompanying changes to the FSAR. The results of these changes are that the design function of the containment system remains valid with no change to those functions due to the updates . The frequency and occurrence of an accident or malfunction of an SSC are not increased due to these changes . These changes do not increase the consequences of an accident or of a malfunction of an SSC. No accidents of a different type or malfunction of an SSC with a different result are created due to these changes. The changes do not result in an exceedance or alteration of a fission product barrier. Lastly, the changes do not depart from a method of evaluation described in the UFSAR to establish the design basis or safety analysis.

Activity: LDCR 2020-008

Title: Revise the FNP Licensing Basis for Tornado Missile Protection

10 CFR 50.59 Evaluation Summary:

In response to NRC Regulatory Issue Summary (RIS) 2015-06, SNC performed walkdowns at FNP to identify potential discrepancies with the current licensing basis related to tornado missile protection. Those walkdowns identified conditions where the plant configuration did not conform to the design and licensing bases. The non-conforming conditions were entered into the corrective action program. Conditions that rendered the affected SSCs inoperable were processed in accordance with Enforcement Guidance Memorandum (EGM) 15-002, with short term compensatory and long-term compensatory actions taken. These compensatory actions resulted in those SSCs being restored to Operable, but non-conforming status. The compensatory actions will remain in effect until the SSCs have been restored to full compliance

with the licensing basis. The NRC-approved Tornado Missile Risk Evaluator (TMRE) methodology (NEI 17-02, Revision 1A approved by the NRC for Vogtle Electric Generating Plant (VEGP)) will be used to resolve these non-conformances.

FNP previously used the NRC-approved TORMIS methodology to analyze the probability of tornado missile damage to non-protected, non-conforming SSCs. The TMRE analysis addresses the previously analyzed non-conforming conditions in addition to the newly identified non-conforming conditions.

This activity uses the TMRE evaluation methodology to evaluate and accept as-is nonconformances regarding tornado missile protection (TMP) for certain aspects of the affected components. As part of this activity, the licensing basis (Updated Final Safety Analysis Report) is being updated to reflect that tornado missile protection is not required for these components.

The responses to the evaluation questions (1-8) were no and as a result, NRC prior approval is not required for this activity.