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NL-18-1299

U.S. Nuclear Regulatory Commission
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Joseph M. Farley Nuclear Plant - Units 1 & 2
Revision 28 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 28 to the Joseph M. Farley Nuclear Plant (FNP) Updated Final Safety Analysis Report (UFSAR). The revised FNP UFSAR pages, indicated as Revision 28, reflect changes through October 1, 2018.

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 92, reflect changes to the TS Bases through October 1, 2018.

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 41.0, reflect changes to the TRM through October 1, 2018. The updated NFPA 805 Fire Protection Program Design Basis Document, Version 5.0, also reflects changes through October 1, 2018.

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

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

A006
A053
NRR

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 30 day of October 2018.

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 ROMs, only)
RType: CFA04.054

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

| SEQ | FILENAME<br>CONTENT | EXTENSION |
|---|---|---|
| 010 | FARLEY FSAR_TECH SPECS BASES<br>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 PART 1 (A177040 sh 360 thru A177048 sh 325) | .pdf |
| 014 | FARLEY FSAR REF DWGS PART 2 (A177048 sh 326 thru A177048 sh 568) | .pdf |
| 015 | FARLEY FSAR REF DWGS PART 3 (A207048 sh 1 thru A207048 sh 300) | .pdf |
| 016 | FARLEY FSAR REF DWGS PART 4 (A207048 sh 301 thru A207048 sh 568) | .pdf |
| 017 | FARLEY FSAR REF DWGS PART 5 (A508650 sh 1 thru D175012 sh 1) | .pdf |
| 018 | FARLEY FSAR REF DWGS PART 6 (D175014 sh 1 thru D177944 sh 1) | .pdf |
| 019 | FARLEY FSAR REF DWGS PART 7 (D181620 sh 1 thru U611138) | .pdf |
| 020 | REVISION 28 NOMENCLATURE | .doc |

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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 SNC368758

**Title:** Open Permissive Interlock (OPI) Setpoint Change (Unit 1)

**10 CFR 50.59 Evaluation Summary:**

This activity reduces the OPI setpoint from 402.5 psig to 383 psig. The design of the Residual Heat Removal (RHR) system includes two isolation valves in series on each inlet from the high-pressure Reactor Coolant System (RCS) to the lower pressure RHR. Each isolation valve on each inlet line is interlocked with one of the two independent RCS pressure signals to provide an OPI to these valves. The OPI prevents the valves from being opened when the RCS pressure is approximately greater than 402 psig and the pressurizer vapor space temperature is greater 475° F (pressurizer vapor space temperature interlock is applicable only to the valves closest to the RHR).

The OPI changes from 402.5 psig to 383 psig based on the current Farley RHR OPI setpoint of 402.5 psig not incorporating instrument uncertainty. The revised setpoint of 383 psig incorporates the instrument uncertainty documented in LTR-SEE-16-48, "RHRS Autoclosure Deletion Alarm Setpoint and Revised RHRS Open Permissive Interlock Setpoint for Farley Units 1 & 2"; therefore, it makes the new setpoint more accurate.

The OPI will continue to perform its intended functions of preventing the valves from being opened when RCS pressure is greater than 383 psig and the pressurizer vapor space temperature is greater than 475° F. The upper and lower limits for Farley's RHR open permissive setpoint are 432 psig and 234 psig, respectively per ALA-98-250, "Upper and Lower Limits for Open Permissive Setpoints".

The upper limit protects against RHR relief valve lift and overpressurization of the RHR pump discharge; therefore, this represents a beneficial improvement to the RHR system based on adding more margin to the upper OPI operational limit and RHR design pressure (600 psig). Lowering the OPI setpoint does not represent an adverse change to the RHR system as described in the Updated FSAR; design functions of the RHR system will continue to be met.

The current method of evaluation does not incorporate instrument uncertainty; the revised setpoint incorporates instrument uncertainty resulting in a more accurate method of evaluation.

The square root sum of the squares method is "approved by the NRC for the intended application" as discussed in NEI 96-07, Section 3.4. This method is consistent with ISA Standard S67.04, Part I, 1994 which is endorsed by the NRC. Therefore, the proposed 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:** Design Change SNC368759

**Title:** Open Permissive Interlock (OPI) Setpoint Change (Unit 2)

**10 CFR 50.59 Evaluation Summary:**

This activity reduces the OPI setpoint from 402.5 psig to 383 psig. The design of the Residual Heat Removal (RHR) system includes two isolation valves in series on each inlet from the high-pressure Reactor Coolant System (RCS) to the lower pressure RHR. Each isolation valve on each inlet line is interlocked with one of the two independent RCS pressure signals to provide an OPI to these valves. The OPI prevents the valves from being opened when the RCS pressure is approximately greater than 402 psig and the pressurizer vapor space temperature is greater 475° F (pressurizer vapor space temperature interlock is applicable only to the valves closest to the RHR).

The OPI changes from 402.5 psig to 383 psig based on the current Farley RHR OPI setpoint of 402.5 psig not incorporating instrument uncertainty. The revised setpoint of 383 psig incorporates the instrument uncertainty documented in LTR-SEE-16-48, "RHRS Autoclosure Deletion Alarm Setpoint and Revised RHRS Open Permissive Interlock Setpoint for Farley Units 1 & 2"; therefore, it makes the new setpoint more accurate.

The OPI will continue to perform its intended functions of preventing the valves from being opened when RCS pressure is greater than 383 psig and the pressurizer vapor space temperature is greater than 475° F. The upper and lower limits for Farley's RHR open permissive setpoint are 432 psig and 234 psig, respectively per ALA-98-250, "Upper and Lower Limits for Open Permissive Setpoints".

The upper limit protects against RHR relief valve lift and overpressurization of the RHR pump discharge; therefore, this represents a beneficial improvement to the RHR system based on adding more margin to the upper OPI operational limit and RHR design pressure (600 psig). Lowering the OPI setpoint does not represent an adverse change to the RHR system as described in the Updated FSAR; design functions of the RHR system will continue to be met.

The current method of evaluation does not incorporate instrument uncertainty; the revised setpoint incorporates instrument uncertainty resulting in a more accurate method of evaluation.

The square root sum of the squares method is "approved by the NRC for the intended application" as discussed in NEI 96-07, Section 3.4. This method is consistent with ISA Standard S67.04, Part I, 1994 which is endorsed by the NRC. Therefore, the proposed 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:** Design Change SNC728701

**Title:** Unit 2 Main and Bypass Feedwater Reg Valve Positioner Replacement

**10 CFR 50.59 Evaluation Summary:**

The proposed activity installs new digital valve positioners and dual solenoid assemblies on the main feedwater regulating valves (MFRVs) and main feedwater bypass valves (MFBVs) for Farley Unit 2. Using the guidance obtained from supporting documents "Guideline on Licensing Digital Upgrades", TR-102348 Revision 1 by the Nuclear Energy Institute (NEI) and "Guidelines for 10CFR50.59 Implementation", NEI 96-07 Revision 1, it was determined that the new digital valve positioners do not increase the frequency, likelihood, or severity of any accident or malfunction previously evaluated in the UFSAR, nor do the new digital valve positioners introduce any new accident as documented in the 50.59 Screening/Evaluation. Therefore, the installation of the MFRV and MFBV digital positioners and dual solenoid assemblies into the Farley Unit 2 plant does not constitute a need to notify or seek permission from the Nuclear Regulatory Commission prior to implementation, in accordance with 10CFR50.59.

**Activity:** Design Change SNC730101

**Title:** Unit 1 Main and Bypass Feedwater Reg Valve Positioner Replacement

**10 CFR 50.59 Evaluation Summary:**

The proposed activity installs new digital valve positioners and dual solenoid assemblies on the main feedwater regulating valves (MFRVs) and main feedwater bypass valves (MFBVs) for Farley Unit 1. Using the guidance obtained from supporting documents "Guideline on Licensing Digital Upgrades", TR-102348 Revision 1 by the Nuclear Energy Institute (NEI) and "Guidelines for 10CFR50.59 Implementation", NEI 96-07 Revision 1, it was determined that the new digital valve positioners do not increase the frequency, likelihood, or severity of any accident or malfunction previously evaluated in the UFSAR, nor do the new digital valve positioners introduce any new accident as documented in the 50.59 Screening/Evaluation. Therefore, the installation of the MFRV and MFBV digital positioners and dual solenoid assemblies into the Farley Unit 1 plant does not constitute a need to notify or seek permission from the Nuclear Regulatory Commission prior to implementation, in accordance with 10CFR50.59.

**Activity:** Design Change SNC793926

**Title:** Integrated Plant Computer Cyber Security Remediation (Unit 1 and Common IPC)

**10 CFR 50.59 Evaluation Summary:**

The proposed activity involves a design change that will upgrade the Unit 1 and common integrated plant computer (IPC) to make it compliant with the cyber security requirements of Regulatory Guide 5.71 and NEI 08-09 while maintaining all existing functions of the IPC.

The IPC provides calculations, alarms, and trending to the operators, reducing their workload; however, it is not a technical specification instrument. For all values displayed by the IPC, there exists alternate indications or means of calculating values. The functions provided to the operators and to other plant systems, and the way in which the operators interact with the upgraded IPC system is the same as in the existing system. Operators will not be adversely impacted by the upgraded IPC.

This is a digital-for-digital upgrade, and the guidance of NEI 01-01 has been utilized to evaluate this upgrade. The proposed activity cannot result in increased frequency, nor in increased consequences, of an accident evaluated in the UFSAR. The IPC is not an accident initiator and does not have any mitigating control functions. The upgraded IPC will not adversely impact the environment in which it is installed. The UFSAR described functions of the IPC remain unchanged by this modification. The proposed activity cannot result in increased occurrences or consequences of an SSC malfunction.

The possibility of adverse impacts from software changes are mitigated by validation of the system software by the factory acceptance test and the site acceptance test, consistent with plant procedure NMP-GM-007-002. Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.

**Activity:** Design Change SNC793928

**Title:** Integrated Plant Computer Cyber Security Remediation (Unit 2)

**10 CFR 50.59 Evaluation Summary:**

The proposed activity involves a design change that will upgrade the Unit 2 integrated plant computer (IPC) to make it compliant with the cyber security requirements of Regulatory Guide 5.71 and NEI 08-09 while maintaining all existing functions of the IPC.

The IPC provides calculations, alarms, and trending to the operators, reducing their workload; however, it is not a technical specification instrument. For all values displayed by the IPC, there exists alternate indications or means of calculating values. The functions provided to the operators and to other plant systems, and the way in which the operators interact with the upgraded IPC system is the same as in the existing system. Operators will not be adversely impacted by the upgraded IPC.

This is a digital-for-digital upgrade, and the guidance of NEI 01-01 has been utilized to evaluate this upgrade. The proposed activity cannot result in increased frequency, nor in increased consequences, of an accident evaluated in the UFSAR. The IPC is not an accident initiator and does not have any mitigating control functions. The upgraded IPC will not adversely impact the environment in which it is installed. The UFSAR described functions of the IPC remain unchanged by this modification. The proposed activity cannot result in increased occurrences or consequences of an SSC malfunction.

The possibility of adverse impacts from software changes are mitigated by validation of the system software by the factory acceptance test and the site acceptance test, consistent with plant procedure NMP-GM-007-002. Based on this 10 CFR 50.59 evaluation, the proposed activity may be implemented without prior NRC approval.

**Activity:** Design Change SNC749222

**Title:** Replace Control Room Air Conditioning System Condenser Motor Controller

**10 CFR 50.59 Evaluation Summary:**

The proposed activity replaces each of the obsolete Reliance SP500s associated with the Control Room Air Conditioning System (CRACS) condensing units with Danfoss FC-102s, a power filter, and three ferrite chokes. Each of the four, 100% capacity CRACS condensing units contains six direct drive, propeller type cooling fans to reject the control room heat to the atmosphere. Two of the six fan motors utilize VFDs to control fan speed based on refrigerant pressure. The Danfoss FC-102s are designed to control fan speed based on refrigerant line pressure, the same manner the existing Reliance SP500s use to control fan speed.

The supplemental questions provided in NEI 01-01, "Guideline on Licensing Digital Upgrades," Appendix A have been reviewed and considered in the basis for the answers provided to the 10 CFR 50.59 Evaluation questions.

The answers to the applicable questions in this evaluation are all no. Therefore, the proposed activity does not does not constitute a need to notify or seek permission from the Nuclear Regulatory Commission prior to implementation, in accordance with 10 CFR 50.59.

**Activity:** Temporary Modification SNC881186

**Title:** Unit 2 Containment Pressure Transmitter PT0952 and PT0980 Signal Swap

**10 CFR 50.59 Evaluation Summary:**

This proposed activity removes the output of PT0952 from service, removes the input of PT0950 as the input to the PT0950 instrument loop, and replaces the output of PT0950 as the input to the functions for the PT0952 instrument loop. The input for containment spray initiation and containment isolation phase B, MSIV isolation logic, and SI initiation logic are maintained for the PT0952 instrument loop. The PT0950 input to containment spray initiation logic and containment isolation phase B is bypassed. The PT0950 input to recorder Q2E13PR0950Z (PT0950Z) and the PT0950 input to normal containment pressure indication is inoperable.

Use of the PT0950 transmitter output as the input to the PT0952 instrument loop maintains the functions of the PT0952 instrument loop. Removal of the input to the PT0950 instrument loop requires entry into required action statements for the PT0950 instrument loop (containment pressure HI-3). The containment spray and containment isolation phase B actuation logic is essentially two out of three, achieved by placing the PT0950 loop in bypass. The technical specification requirements for post-accident monitoring instrumentation per Table 3.3.3-1 of the technical specification requires two channels of containment pressure (narrow range), therefore removal of one of four does not require action.

Technical specification requirements for SI logic initiation on containment pressure (HI-1) and MSIV closure on containment pressure (HI-2) are not impacted.

Placing the PT0950 loop in bypass results in three of four channels available for containment spray actuation and containment isolation phase B actuation on containment pressure (HI-3), with the fourth channel in bypass. With the PT0950 instrument loop in bypass, wiring changes to the PT0952 loop effectively removes the PT0952 loop from service during that work and therefore the proposed activity and work to restore to normal configuration requires entry into technical specification LCO 3.0.3.

The narrow range input to recorder PT0950Z is not used in accident analysis. In accordance with references 6 and 7, the containment pressure trends and transients required by the EOPs can be identified by observations of the direct indicating displays. The remaining channels are also recorded on the computer. Containment pressure extended range transmitter PT0950Z also provides an input to this recorder and is not impacted by the proposed activity.

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Enclosure 4
Revised NRC Commitments Report

**Original Commitment**: Scope of the Service Water Program
SNC28376

The scope of the Service Water Program will be expanded prior to the period of extended operation to include inspection of piping from the main service water header to the air compressor credited for Appendix R safe shutdown and inspection of the service water pump columns.

**Revised Commitment**: The scope of the Service Water Program will be expanded prior to the period of extended operation to include inspection of piping from the main service water header to the air compressor credited for Appendix R safe shutdown and inspection of the service water pump columns. FIVP-O-M-82.0 and the Service Water Pump Repair Plan have been revised to show requirement to inspect the Service Water pump columns.

**Justification for Change**

A new last sentence is added to address the part of the commitment which requires the Service Water pump columns to be inspected. In addition, the status of this commitment is changed to "IMPL" (implemented).

**Original Commitment**: Visual Inspection of Fire Barriers
SNC28379

Visual inspection by qualified fire protection personnel of fire barrier walls/floors/ceilings, as well as doors and penetration seals.

**Revised Commitment**: Visual inspection of fire barrier walls/floors/ceilings, as well as doors and penetration seals.

**Justification for Change**

Fire barrier walls/floors/ceilings are being inspected under NMP-ES-021, "Structural Monitoring Program (SMP)". Comparison of this commitment with the SMP guidelines show that the age-related degradation to fire barriers will be noted as part of the SMP inspections. The personnel conducting the SMP inspections are trained and qualified to evaluate structural components and they specifically look for concrete spalling and cracking as called for in the fire barrier inspection process. This is considered an editorial change to the commitment to clarify that the fire barrier inspections are being conducted by qualified personnel who are not part of the fire protection team.

**Original Commitment:** External Surfaces Monitoring Program
SNC28398 (Legacy Number 10684)

License Renewal Commitment (33): The External Surfaces Monitoring Program will inspect accessible, in-scope polymers and elastomers for age related degradation.

**Revised Commitment:** N/A – Incorporated into FSAR Section 18.2.16, "External Surfaces Monitoring Program".

**Justification for Change**

This commitment has been incorporated into FSAR Section 18.2.16, thereby allowing its status to change to "Closed".

**Original Commitment:** One-Time Inspection Program
SNC28404 (Legacy Number 10691)

License Renewal Commitment (41): The One-Time Inspection Program sample population will include the Pressurizer cast austenitic stainless-steel spray heads and associated coupling/lock bar.

**Revised Commitment:** The One-Time Inspection Program sample population will include the Pressurizer cast austenitic stainless-steel spray heads and associated coupling/lock bar. The inspection will consist of a visual inspection of the Pressurizer Spray Head and associated coupling/lock bar. Additionally, if an indication is identified, a flaw tolerance evaluation will be performed. This will satisfy the inspection requirement for loss of fracture toughness identified in Table 3.1.2-3 of the FNP License Renewal Application.

**Justification for Change**

The purpose for revising the commitment is to supplement the commitment to state that the inspection of the spray head and associated coupling/lock bar for loss of fracture toughness, as specified in Table 3.1.2-3 of the License Renewal Application, is met by performance of a VT inspection and a flaw tolerance evaluation of any identified indications.

**Original Commitment:** FNP Operations QA Program for Aging Management Activities
SNC28416 (Legacy Number 10703)

License Renewal Commitment (81): The FNP Operations QA Program will apply the quality assurance criteria of 10 CFR 50, Appendix B to the elements of corrective actions, confirmation process, and administrative controls for the aging management program activities and implementing documents during the period of extended operation. These criteria will be applied to all safety-related and nonsafety-related structures and components that perform an intended function for license renewal.

**Revised Commitment:** N/A – Incorporated into FSAR Section 18.2.1, "Quality Assurance Requirements".

**Justification for Change**

This commitment has been incorporated into FSAR Section 18.2.1, thereby allowing its status to change to "Closed".

**Original Commitment:** The Application of Environmental Factors for Charging Nozzles and Alternate Charging Nozzles
SNC28430

The application of environmental factors results in adjusted cumulative usage factors exceeding 1.0 for the charging nozzles and alternate charging nozzles, as well as the RHR 6 NPS RHR/SI nozzles to the RCS cold leg. For these locations, SNC will take corrective actions prior to the period of extended operation. These actions may include a more refined analysis, repair, replacement, or an NRC approved inspection program. If the option to pursue an inspection program is selected, SNC will submit a license amendment request to obtain the required NRC approval.

**Revised Commitment:** N/A – Incorporated into FSAR Section 18.4.2.1, "ASME Section III, Class 1 Component Fatigue Analysis".

**Justification for Change**

This commitment has been incorporated into FSAR Section 18.4.2.1, thereby allowing its status to change to "Closed".

**Original Commitment:** Periodic Evaluation of Severe Accident Mitigation Alternatives (SAMAs) 7 and 11
SNC35955 (Legacy Number 10749)

License Renewal Commitment (127): SNC will periodically evaluate SAMAs 7 and 11.

SAMA 7: Increase Charging Pump Lube Oil Capacity

SAMA 11: Use existing hydro test pump for RCP seal injection

**Revised Commitment:** No change in commitment description; status change only. See discussion in Justification for Change.

**Justification for Change**

A review of the basis for SAMAs 7 and 11, which were originally identified from the risk profiles of the Farley Internal Events PRA models in the 2002-2003 timeframe, was conducted. Based on this review, it is concluded that as a result of plant design and PRA model changes since 2004, these SAMAs would no longer address the dominant contributors to risk in the current Farley PRA models. Therefore SAMAs 7 and 11 are no longer applicable and re-evaluation and update of the cost-benefit analysis does not need to be performed.

This commitment status is changed to "Closed".

**Original Commitment:** Technical Specification 3/4.6.1.7
SNC499872

Alabama Power Company currently maintains the purge valves (48-inch) in a closed position. These valves are air-operated with air supply required to open the valves. The electrical breakers to the air supply solenoid valves are tagged out in Modes 1 through 4. The Plant Manager's permission is required to remove the tag and provide air to the operator. Only after this action can the valves be operated. The closed valve position is verified every 31 days.

**Revised Commitment:** No change in commitment description; status change only. See discussion in Justification for Change.

**Justification for Change**

Commitment became outdated with the transition to Improved Tech Specs. Tech Spec Surveillance requirement 3.6.3.1 covers meeting the intent of this commitment, therefore a separate commitment is not required. The valves are deactivated by opening links in the associated air solenoid valve control circuitry as directed by plant procedure FNP-1/2-STP-14.0. These links are verified open during performance of FNP-1/2-STP-14.0. The use of a tagging order with the Plant Manager holding clearance is no longer desired.

This commitment status is changed to "Closed".

**Original Commitment**: FSAR Chapter 13.0 – Conduct of Operations
SNC501101 (Legacy Number 8880)

The Manager – Safety Audit and Engineering Review will make or cause to be made an annual audit of the plant security program.

**Revised Commitment**: N/A – Incorporated into FSAR Section 18.4.2.1, "ASME Section III, Class 1 Component Fatigue Analysis"..

**Justification for Change**

FSAR Section 13.7.2.6 currently states, "The nuclear oversight vice president will make or cause to be made audits of the plant security program as specified in the Security Plan." The frequency for a Security Audit at least every 24 months is contained in the QATR and the Security Plan, which both are licensing documents. The Security Plan, Section 17 wording agrees with 10 CFR 73.55(m).

This commitment status is changed to "Closed".

**Original Commitment**: July 27, 1987 Inspection Report Inaccuracies – The Lack of a Formal Process to Contact Key Vendors
SNC34379 (Legacy Number 198736993)

The lack of a formal process to contact key vendors is still an open item with Alabama Power Company for Section 2.2.2 dated December 15, 1986. Alabama Power Company has noted its disagreement with the NRC position and on March 17, 1987 formally notified the NRC of such. This is one of several instances in the report (we note this as an example) in which the NRC appears to have applied the provisions of Section 2.2.2. Alabama Power Company intends to continue working with the NRC staff to resolve this open issue.

**Revised Commitment**: No change in commitment description; status change only. See discussion in Justification for Change.

**Justification for Change**

This commitment documented resolution of differences between Alabama Power Company and the NRC regarding implementation of GL 90-03(b). Benchmarking has shown that the industry has weighed the cost and the time required to complete the vendor contact effort balanced against the risks to safe operation and the goals of equipment reliability. Many operators have eliminated or are in the process of eliminating the non-NSSS portion of their programs. Elimination of undue administrative burden aligns with the goals of Delivering the Nuclear Promise and reducing regulatory risk. Furthermore, there are multiple other programs currently in place in the SNC fleet that achieve the same goals as the original intent of this aspect of the generic letter.

The technological advancements in equipment performance information exchanges and other OE sharing meet the intent of GL 83-28 and GL 90-03 and have obviated the need for direct periodic contact with vendors. This proposed change will not affect SNC contact with fleet NSSS vendors as defined in GL 90-03(a). The requirement to have direct contact with key safety-related vendors was made in a time when computers, email and other regulatory and industry processes were not well integrated into every-day business. Therefore, sunsetting (status of CT-CLOSED) of this commitment is justified.

**Original Commitment**: Generic Letter 90-03, Alabama Power Company Will Expand the Vendor Contact Program
SNC35642 (Legacy Number 199038610)

To comply with the requirements of Generic Letter 90-03, Alabama Power Company will expand the Vendor Contact Program to include certain vendors of other safety-related equipment. Included in this expanded program will be vendors of auxiliary feedwater pumps, safety-related batteries and battery chargers, safety-related inverters, key safety-related switchgear, service water pumps, component cooling water pumps, and key safety-related non-manual valve operators. The program will contain provisions for contacting these vendors on an annual basis to request pertinent technical information documentation issued by the vendor during the preceding year. The means for contacting vendors will normally be via telephone. Results of these telephone conversations will be documented.

Alabama Power Company will revise existing procedures to incorporate the expanded program described above by January 1, 1991. All vendors will be contacted by June 1, 1991 and on an annual basis thereafter.

**Revised Commitment**: No change in commitment description; status change only. See discussion in Justification for Change.

**Justification for Change**: This commitment documented resolution of differences between Alabama Power Company and the NRC regarding implementation of GL 90-03(b). Benchmarking has shown that the industry has weighed the cost and the time required to complete the vendor contact effort balanced against the risks to safe operation and the goals of equipment reliability. Many operators have eliminated or are in the process of eliminating the non-NSSS portion of their programs. Elimination of undue administrative burden aligns with the goals of Delivering the Nuclear Promise and reducing regulatory risk. Furthermore, there are multiple other programs currently in place in the SNC fleet that achieve the same goals as the original intent of this aspect of the generic letter.

The technological advancements in equipment performance information exchanges and other OE sharing meet the intent of GL 83-28 and GL 90-03 and have obviated the need for direct periodic contact with vendors. This proposed change will not affect SNC contact with fleet NSSS vendors as defined in GL 90-03(a). The requirement to have direct contact with key safety-related vendors was made in a time when computers, email and other regulatory and industry processes were not well integrated into every-day business. Therefore, sunsetting (status of CT-CLOSED) of this commitment is justified.