March 17, 2004

Mr. L. M. Stinson
Vice President
Southern Nuclear Operating Company
Post Office Box 1295
Birmingham, Alabama 35201

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION

Dear Mr. Stinson:

By letter dated September 12, 2003, Southern Nuclear Operating Company, Inc. (SNC or the applicant) submitted an application pursuant to 10 CFR Part 54, to renew the operating licenses for Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, for review by the U.S. Nuclear Regulatory Commission's (NRC). The NRC staff is reviewing the information contained in the license renewal application (LRA) and has identified, in the enclosure, areas where additional information is needed to complete the review. Specifically, the enclosed requests for additional information (RAIs) are from Section 3.0, Aging Management Review Results [Clarification and/or Confirmatory in nature]; Section 3.1, Aging Management of Reactor Vessel, Internals, and Reactor Coolant System; Section 4.0, Time Limited Aging Analysis; and Appendix B, Aging Management Programs and Activities.

These RAIs, in a draft format, have been provided to Mr. Jan Fridrichsen of your staff on November 25, 2003; February 4 and 9, 2004. The NRC staff has discussed draft versions of these RAIs, via a conference call and a meeting, to provide clarifications to the SNC staff on March 10 and 15, 2004. Your responses to these RAIs are requested within 30 days from the date of this letter. Mr. Fridrichsen has agreed to this request. If needed, the NRC staff is willing to meet or discuss with SNC again prior to the submittal of the applicant's responses to provide clarifications to the staff's RAIs.

If you have any questions, please contact me at 301-415-1315 or e-mail tyl1@nrc.gov.

Sincerely,

/***RA***/

Tilda Liu, Project Manager
License Renewal Section A
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: As stated

cc w/encl: See next page

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JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION (RAI)

Section 3.0: Aging Management Review Results

RAI 3.0-1

The following questions are CONFIRMATORY and CLARIFICATION (C/C) in nature. The corresponding draft RAI number associated with each question is indicated in parenthesis.

A. (D-RAI 3.1-3)

In LRA Table 3.1.2-3, the ISI program is not credited to manage cracking of non-Class 1 piping and valve components. However, LRA summary Table 3.1.1, item 3.1.1-36 (linked to the non-Class 1 piping and valve bodies) states:

> "The FNP AMR results are consistent with this summary item. Consistent with NUREG-1801, the Water Chemistry Program and Inservice Inspection Program will manage cracking of these components."

There is an apparent inconsistency between the two Tables of the LRA. Please explain whether the ISI program is credited for the non-Class 1 piping and valve bodies and, if necessary, correct the apparent inconsistency.

B. (D-RAI 3.4-4)

LRA Table 3.4.2-1 identifies no aging effects for alloy steel steam/fluid traps in an outside environment. The LRA defines an outside environment as: "An environment where components are exposed to direct sunlight, precipitation, and freezing conditions. The outside environment also conservatively includes components located in sheltered areas where the component is beneath some type of roof structure or outdoor enclosure (such as a valve box) but is otherwise open to the ambient environment." The GALL report recommends aging management for the loss of material due to general corrosion on the external surfaces of carbon (alloy) steel components exposed to operating temperatures less than 212°F, such corrosion may be due to air, moisture, or humidity. The applicant is requested to provide a program to manage corrosion on the external surface of alloy steel steam/fluid traps in an outside environment or to provide justification for not managing this aging effect.

C. (D-RAI 3.5-6)

Regarding the AMR summary covering FNP's sump trash rack listed on Table 3.5.2-1 (page 3.5-38) of the LRA, the applicant identified no applicable aging effect as well as AMP for the stainless steel component. Since sumps tend to be exposed to high moisture, acidic or accumulated water environment, discuss FNP's past operating/inspection experience covering sump trash racks to support its AMR finding that no AMP is needed for the component.

D. (D-RAI 3.5-11)

In Item 3.3.1-11 (Table 3.3.1), the applicant states that the FNP new fuel storage racks are fabricated from both carbon steel (CS) and stainless steel (SS). Chapter VII of NUREG 1801 does not address such hybrid rack configurations. Depending on the CS-SS interface between the racks, stress corrosion cracking of the SS portion of the racks cannot be ruled out. The applicant is requested to provide justification for not requiring aging management of the SS portion of the new fuel storage racks.

Section 4.0: Time Limited Aging Analysis

RAI 4.0-1

The following question is CONFIRMATORY and CLARIFICATION (C/C) in nature. The corresponding draft RAI number associated with each question is indicated in parenthesis.

A. (D-RAI 4.3.2.2-1)

The staff needs further clarification as to the number of reactor coolant pump (RCP) start/stop cycles that are assumed in the 60-year RCP flywheel fatigue crack growth assessment for the Farley units. In Section 4.3.2 of the Farley license renewal application (LRA), SNC states that 4000 RCP start/stop cycles are assumed in the analysis. However, in its letter dated December 5, 2003, SNC states that 6000 RCP start/stop cycles are assumed for the bounding 60-year RCP flywheel fatigue crack growth assessment. Clarify the number of start/stop cycles assumed for the bounding 60-year RCP flywheel fatigue crack growth assessment and which reference (WCAP Topical Report) contains the 60-year RCP flywheel fatigue crack growth analysis for the Farley units.

Appendix B: Aging Management Programs

RAI B-1

The following question is CONFIRMATORY and CLARIFICATION (C/C) in nature. The corresponding draft RAI number associated with each question is indicated in parenthesis.

A. (D-RAI B.5.2-2)

SNC's AMP description for the Flux Detector Thimble Inspection Program implies that SNC may use alternative inspection methods for the thimble tubes examinations in lieu of ECT but did not define which inspection methods might be used as an alternative to ECT. The staff therefore requests that, if alternative inspection methods are used in lieu of ECT, the applicant provide further clarification regarding the inspection methods that will be used for the flux thimble examinations and how the alternative inspection methods, if used, will be qualified as being capable of detecting loss of material/wear in the flux thimble tubes.

Section 3.1: Aging Management of Reactor Vessel, Internals, and Reactor Coolant System

RAI 3.1.3.1.1-1

a. The staff requires additional information on the applicant's AMRs for managing loss of material in the NiCrFe components and stainless steel components that are exposed to borated water environment, particularly since aging management strategies for license renewal are more dependent of the specific types of aging mechanisms that can induce age-related degradation and to a lesser degree on the general classification of the aging effect (in this case loss of material). For the components listed within the scope of this RAI, confirm that loss of material is an applicable aging effect requiring aging management. Specifically, for each NiCrFe or stainless steel component or commodity group that is identified below as being within the scope of this RAI and for which loss of material has been confirmed to be an applicable aging effect, define which aging mechanism or mechanisms are known to induce loss of material in the specific components or commodity group of components. This RAI is applicable to the following commodity group components in LRA Tables 3.1.2-1, 3.1.2-2, and 3.1.2-3 that have corresponding AMRs for evaluating loss of material under internal exposure to the borated water environment:

Table 3.1.2-1, Reactor Coolant Systems, Reactor Vessel – Summary of Aging Management Review

- bottom head torus and dome (low alloy steel with stainless steel cladding)
- bottom mounted instrumentation guide tubes (stainless steel)
- bottom mounted instrumentation penetrations (Alloy 600, a NiCrFe alloy)
- core exit thermocouple (CET) and heated junction thermocouple closure (HJTC) assemblies (stainless steel)
- closure head dome and flange (low alloy steel with stainless steel cladding)
- RV core support lugs (Alloy 600, a NiCrFe alloy)
- control rod drive mechanism (CRDM) and instrumentation housing penetration nozzles (thermally treated Alloy 690, a NiCrFe alloy)
- CRDM housing flange adapters (stainless steel)
- CRDM latch housings and rod travel housings (stainless steel)
- RV head vent penetration (thermally treated Alloy 690, a FeNiCr alloy)
- intermediate and lower shell courses (low alloy steel with stainless steel cladding)
- RV leakage monitoring tube assembly (Alloy 600, a NiCrFe alloy)
- RV primary nozzle safe end (stainless steel with Alloy 82/182 welds and buttering, NiCrFe weld filler metals)
- vessel flange (alloy steel with stainless steel cladding)
- seal table and fittings (stainless steel)
- upper (nozzle) shell course (low alloy steel with stainless steel cladding)
- stainless steel cladding for the alloy steel or carbon steel RV components that are clad with austenitic stainless steel

Table 3.1.2-2, Reactor Coolant Systems, Reactor Vessel Internals – Summary of Aging Management Review

- baffle and former plates (stainless steel)
- baffle bolts (stainless steel)
- Bottom mounted instrumentation (BMI) column cruciforms (cast austenitic stainless steel or CASS)
- BMI columns with fasteners (stainless steel)
- clevis inserts and fasteners (FeNiCr Alloy - Alloy 600 inserts and Alloy X-750 fasteners)
- control rod drive guide tube assemblies with associated fasteners (stainless steel) core barrel and core barrel flange (stainless steel)
- core barrel outlet nozzles (stainless steel)
- control rod drive guide tube (CRGT) support pins (stainless steel)
- flux thimble tubes (stainless steel)
- reactor pressure vessel/head alignment pins with associated fasteners (stainless steel)
- head cooling spray nozzles (stainless steel)
- HJTC probe holder, probe holder extension, and probe holder shroud assemblies with associated fasteners (stainless steel)
- internals holddown spring (stainless steel)
- lower core plate and fuel alignment pins (stainless steel)
- lower support columns with associated fasteners (stainless steel)
- lower support forging (stainless steel)
- neutron panels (stainless steel)
- radial keys and fasteners (stainless steel)
- secondary core support assembly with associated fasteners (stainless steel)
- upper core alignment pins with associated fasteners (stainless steel)
- upper core plate and fuel alignment pins with associated fasteners (stainless steel)
- upper instrumentation conduit and supports with associated fasteners (stainless steel)
- upper support assembly with associated fasteners (stainless steel)
- upper support column bases (stainless steel)
- upper support column with associated fasteners (stainless steel)

Table 3.1.2-3, Reactor Coolant Systems, Reactor Coolant System and Connected Lines – Summary of Aging Management Review

- Class 1 piping - reactor coolant loop (cast austenitic stainless steel)
- small bore Class 1 piping less than 4 NPS (stainless steel)
- Class 1 piping greater than or equal to 4 NPS (stainless steel)
- Class 1 valve bodies (stainless steel)
- Class 1 flow orifices or elements (stainless steel)
- reactor coolant pump (RCP) casing (CASS)
- RCP main closure flange (CASS)
- pressurizer heater sheaths (austenitic stainless steel)
- pressurizer instrumentation nozzle and heater well nozzles (stainless steel)
- pressurizer manway and cover (alloy steel with stainless steel insert)

- pressurizer nozzle safe ends (stainless steel with alloy 82/182 welds and buttering, NiCrFe weld filler metals)
- pressurizer surge, spray, safety, and relief nozzles (alloy steel with stainless steel cladding)
- pressurizer shell, upper head, and lower head (alloy steel with stainless steel cladding)
- pressurizer spray head assembly (CASS)
- pressurizer surge and spray nozzle thermal sleeves (stainless steel with alloy 82/182 welds, NiCrFe weld filler metals)
- Non-Class 1 RCS piping (stainless steel)
- Non-Class 1 valve bodies (stainless steel)

b. With the exception of AMPs credited for management of loss of material in the RV flange, incore flux thimble tubes, reactor vessel (RV) internals holddown spring, RV internals radial keys and fasteners, and pressurizer spray head assembly, the applicant credits only the Water Chemistry Program as the aging management program for management of loss of material in the RV, RV internals, pressurizer, and RCS piping and connected system components listed within the scope of RAI 3.1.3.1.1-1a. Justify why SNC considers that the Water Chemistry Program alone is sufficient to manage loss of material in these components without the need to credit an inspection-based AMP to verify that the Water Chemistry Program is accomplishing its mitigative aging management function. The applicant is requested to discuss how the implementation of the Water Chemistry Program relates to management of the specific aging mechanisms that are identified as being capable of inducing loss of material in the components. If the technical assessments (justifications) conclude that the Water Chemistry Program alone is insufficient to manage all of the aging mechanisms leading to loss of material in any of these components, propose acceptable inspection-based AMP for management of loss of material that is applicable to the specific RV, RV internal, RCS piping or pressurizer component.

<u>RAI 3.1.3.1.2-1</u>

a. In Tables 3.1.2-1 and 3.1.2-3 of the Farley LRA, SNC did not identify the aging mechanisms that it determined to be capable of inducing loss of material in reactor vessel (RV), RCS piping, and pressurizer components fabricated from alloy steel or carbon steel materials. Therefore, the staff requests that SNC identify the aging mechanisms that SNC has determined are capable of inducing loss of material in alloy steel or carbon steel RV, RCS piping, or pressurizer components that are exposed externally to the inside environments. In addition, SNC's description of the inside environment in Table 3.0.4-2 of the Farley LRA does not indicate that the applicant is managing the water vapor content in the inside environment to low humidity levels. Provide clarification as to whether the applicant considers loss of material due to general corrosion is an applicable aging effect for external surfaces of alloy steel or carbon steel RV, RCS piping, and pressurizer components that are exposed to the inside environment, and if not, provide the technical basis as to why SNC does not consider general corrosion to be an aging mechanism that needs management in the external surfaces of alloy steel or carbon steel RV, RCS piping, and pressurizer components during the extended periods of operation for Farley Nuclear Plant, Units 1 and 2. This D-RAI is applicable to the following commodity group components in LRA

Tables 3.1.2-1 and 3.1.2-3 that have corresponding AMRs for evaluating loss of material under external exposure to the inside environment:

Table 3.1.2-1, Reactor Coolant Systems, Reactor Vessel – Summary of Aging Management Review

- bottom head torus and dome (alloy steel with stainless steel cladding)
- closure head dome and flange (alloy steel with stainless steel cladding)
- RV closure studs, nuts, and washers (alloy steel)
- intermediate and lower shell courses (alloy steel with stainless steel cladding)
- primary inlet and outlet nozzles and nozzle support pads (alloy steel with stainless steel cladding)
- refueling seal ledge (carbon steel)
- vessel flange (alloy steel with stainless steel cladding)
- upper (nozzle) shell course (alloy steel with stainless steel cladding)
- ventilation shroud support ring (carbon steel)

Table 3.1.2-3, Reactor Coolant Systems, Reactor Coolant System and Connected Lines – Summary of Aging Management Review

- Class 1 closure bolting (alloy steel)
- reactor coolant pump (RCP) main flange bolting (alloy steel)
- pressurizer closure bolting (alloy steel)
- pressurizer manway cover (alloy steel with a stainless steel insert)
- pressurizer nozzles (surge, spray, safety and relief nozzles - low alloy steel with stainless steel cladding)
- pressurizer shell, upper head, and lower head (alloy steel with stainless steel cladding)

b. In the Farley LRA, the SNC credited only the Borated Water Leakage Assessment and Evaluation Program with the management of loss of material from the external surfaces of the alloy steel or carbon steel RV components that are exposed to the inside environment. In RAI 3.1.3.1.2-1a, the staff requested additional information regarding the aging mechanisms that could induce loss of material from the external surfaces of alloy steel and carbon steel RV, RCS piping, and pressurizer components under exposure to inside environments. The staff therefore requests additional information (a technical basis) why SNC considers that the Borated Water Leakage Assessment and Evaluation Program alone is sufficient to manage loss of material in external surfaces of the alloy steel and carbon steel RV, RCS piping, and pressurizer components within the scope of RAI 3.1.3.1.2-1a, and particularly if the “loss of material” aging effect is known to be induced by aging mechanisms other than “boric acid-leakage and boric acid-induced wastage.” If the “loss of material” aging effect is known to be induced by aging mechanisms other than “boric acid-leakage and boric acid-induced wastage,” the staff requests that SNC credit additional aging management programs or activities with management of the “loss of material” aging effect if the “Borated Water Leakage Assessment and Evaluation Program” is determined to be insufficient to assure adequate aging management of the “loss of material” aging effect during the extended periods of operation for the Farley Nuclear Plant, Units 1 and 2.

RAI 3.1.3.2.1-1

In the staff's aging management review (AMR) for Commodity Group IV.B2.6-c of the GALL Report, Volume 2, the staff recommends that both a "plant-specific" aging management program (AMP) and the ASME Section Inservice Inspection, Subsections IWB, IWC, and IWD Program be credited with management of loss of material due to wear in flux detector thimble tubes. The applicant has credited both the Flux Detector Thimble Inspection Program (i.e., a "plant-specific" AMP) and the Water Chemistry Program with aging management of wear in the Farley flux detector thimble tubes. SNC did not credit the ISI Program for management of loss of material due to wear in the flux detector thimble tubes at Farley. Although SNC has credited the Flux Detector Thimble Inspection Program with management of loss of material due to wear in the flux detector thimble tubes, the applicant is requested to provide the technical basis for not crediting the ISI Program as an additional AMP for management of this aging effect in the thimble tubes, as would otherwise be consistent with the staff recommendations in GALL Commodity Group IV.B2.6-c.

Section 4.2: Reactor Vessel Neutron Embrittlement

Section 4.2.1: Neutron Fluence

RAI 4.2.1.3-1

Pursuant to 10 CFR Part 54.21(d), the FSAR Supplement for a facility license renewal application (LRA) must contain a summary description for each aging management program and time-limited aging analysis proposed for management of the effects of aging. The staff has determined that Appendix A of the LRA (FSAR Supplement) did not include a corresponding FSAR Supplement summary description for the TLAA in Section 4.2.1, "Neutron Fluence," of the LRA. The staff recognizes that the licensee calculated fluence values to 54 EFPYs (i.e., the end of the requested license extension). However, the operating assumptions in these calculations could change as for example with the introduction of new fuel, new material properties, etc. In such an instance 10 CFR 50.61 and other regulations requires recalculation of the fluence and reevaluation of the material properties. Therefore it is necessary to capture this information in the FSAR Supplement. Pursuant to 10 CFR 54.21(d), the staff requires that a corresponding FSAR Supplement summary description for LRA Section 4.2.1 be included in the FSAR Supplement.

Section 4.2.3: Pressurized Thermal Shock

RAI 4.2.3.3-1

The applicant's FSAR Supplement summary description for the time-limited aging analysis of pressurized thermal shock (i.e., TLAA for PTS), is described in Section A.4.1.2 of the application. The limiting RT_{PTS} value cited by the applicant for Farley Unit 2 in the FSAR Supplement summary description (i.e., 193°F) is not consistent with the limiting RT_{PTS} value cited in Table 4.2.3-2 of the LRA (i.e., 208°F, based on Intermediate Shell Plant B7212-1). The staff requests that SNC revise the limiting RT_{PTS} value cited in the FSAR Supplement A.4.1.2 for Farley Unit 2 to be consistent with the limiting RT_{PTS} value reported in Table 4.2.3-2 of the LRA (i.e., 208°F).

Section 4.2.4: Adjusted Reference Temperatures

RAI 4.2.4.2-1

The limiting 1/4T and 3/4T adjusted reference temperature values (i.e, RT_{NDT} values) for the reactor vessel (RV) beltline materials in operating reactors are used in the calculations of pressure-temperature (P-T) limits, which are calculated under the scope of the requirements of Section IV.A.2 to 10 CFR Part 50, Appendix G. The applicant did not provide the 3/4T RT_{NDT} values for the limiting 3/4T beltline materials in the RVs of Farley, Units 1 and 2. The staff requests that the applicant supplement the discussion in Section 4.2.4 of the LRA to provide the 3/4T RT_{NDT} values for the limiting 3/4T beltline materials in the reactor vessels of Farley, Units 1 and 2, through 54 EFPY of operation.

RAI 4.2.4.3-1

The applicant did not include an FSAR Supplement summary description for SNC's TLAA on the calculation of the adjusted reference temperature values (RT_{NDT} values) for the RV beltline materials at the 1/4T and 3/4T locations of the Farley RVs. Since SNC has defined these adjusted reference temperature calculations as TLAAs in Section 4.2.4 of the LRA, the applicant is required by 10 CFR 54.21(d) to include an FSAR supplement summary description for the applicant's calculation of the adjusted reference temperature values (RT_{NDT} values) for the RV beltline materials at the 1/4T and 3/4T locations of the Farley RVs. The staff requests that SNC amend the Farley license renewal application to include an FSAR Supplement summary description for the TLAA (Section 4.2.4 of the application) on the calculation of the adjusted reference temperature values (RT_{NDT} values) for the RV beltline materials at the 1/4T and 3/4T locations of the Farley RVs.

Section 4.5.1: Ultimate Heat Sink Silting Calculations

RAI 4.5.1-1

Section 4.5.1 "Ultimate Heat Sink Silting" of the FNP LRA states that the applicant has updated the design calculations pertaining to the surveillance of the Ultimate Heat Sink (UHS) to address silting induced aging. It is further stated that this update addresses the UHS silting issue for the additional 20 years of operations in the extended term in accordance with 10 CFR 54.21(c)(1)(ii). In order to complete the review of the UHS silting issue at FNP site, the staff needs the following additional information:

a. Provide the UHS pond volume surveillance data from all the available sounding measurement records to date. (Raw sounding measurements data are not required)

b. Provide the rate of siltation of the UHS pond that was observed in the past based on the periodic surveillance measurements made thus far. Also address the applicability of this measured rate to the remaining years of the current license period and the extended period of operation (i.e., are there any known future changes in the hydrology of the river likely to increase significantly sediment intake?)

c. Explain briefly the procedure that was used to determine the observed and projected rates of siltation mentioned in item b above, and summarize the significant results indicating the safety margin achieved in volume of water (acre-feet) in UHS.

Appendix B.3.5: Borated Water Leakage Assessment and Evaluation Program

RAI B.3.5-1

NUREG/CR-5576, Survey of Boric Acid Corrosion of Carbon Steel Components in Nuclear Plants [January 1990], summarizes boric acid leakage and corrosion events that occurred in the industry prior to 1990. More recently, industry experience (refer to the operating events summarized in NRC Bulletins 2002-01 and 2003-02 and in NRC Executive Order EA-03-009) has demonstrated that the bi-metallic partial penetration welds (Alloy 82/182 welds) used in the fabrication of upper and lower RV head penetration nozzles may be susceptible to primary water stress corrosion cracking (PWSCC) that could induce leakage of the borated reactor coolant over time. However, the corresponding partial-penetration welds at FNP, Units 1 and 2, were not listed in the SNC's (Alabama Power Company's) GL 88-05 response, dated May 31, 1988, with locations that could be potential sources of borated water leaks.

The staff seeks additional clarification regarding the list of components that are within the scope of the Borated Leakage Assessment and Evaluation Program and the process the applicant uses to augment the list of components that were originally specified within the scope of the applicant's GL 88-05 response, dated May 31, 1988.

a. Provide the list of component locations that are currently within the scope of the Borated Water Leakage Assessment and Evaluation Program, and discuss the process that is used to augment of ASME Code Class 1 and 2 components locations within the scope of the aging management program (AMP) based on industry experience that is relevant to the scope and implementation of the AMP.

b. Discuss how SNC's responses to the following NRC documents have been used to update the list of component locations and types of visual inspections credited within the scope of the Borated Water Leakage Assessment and Evaluation Program or within the scope of other aging management programs (AMPs) that provide for implementation of similar or more conservative types of inspections: NRC Bulletin 2002-01, dated March 29, 2002, and May 16, 2002; NRC's RAIs on the bulletin, dated January 17, 2003; NRC Bulletin 2003-02, dated September 19, 2003; and NRC Order EA-03-009, dated March 3, 2003; April 11, 2003, and April 18, 2003. If the responses have been used to supplement the scope of the Borated Water Leakage Assessment and Evaluation Program or other AMPs, identify which component locations have been added to the scope of the program and clarify what type of visual examinations (i.e., specify whether VT-1, VT-2 or VT-3, and whether the visual examinations are enhanced, bare-surface, qualified, etc.) will be performed on the components within the current scope of the program.

D-RAI B.3.5-2

The applicant's FSAR Supplement summary description for the Borated Water Leakage Assessment and Evaluation Program provides a general reference to commitments made in the applicant's response to Generic Letter (GL) 88-05. However, the staff's requests that the applicant amend the FSAR Supplement summary description to provide a more specific reference to the applicant's response (i.e., Alabama Power Company's response) to GL 88-05, dated May 31, 1988, and to any additional responses to NRC generic communications (i.e., Generic Letters, Bulletins, Orders, or Circular Letters) that are germane to the scope or other program attributes for the AMP or have been used to amend the program attributes for the AMP, including those responses to NRC Bulletins 2002-01 and 2003-02, and to NRC Order EA-03-009, as appropriate.

Appendix B: B.5.2, Flux Detector Thimble Inspection Program

RAI B.5.2-1

In SNC's response to Bulletin 88-09, the SNC indicated that it had performed inspections of 100% of the flux detector thimble tubes at Farley Unit 1 during refueling outages Nos. 7 and 8 and at Farley Unit 2 during refueling outage No. 5. SNC's bulletin response did not indicate whether the applicant would continue to perform 100% inspections of the thimble tubes during subsequent refueling outages. The staff seeks clarification whether the scope of the Flux Detector Thimble Inspection Program will continue to perform eddy current testing (ECT) inspections of 100% of the flux detector thimble tubes. If the percentage of the flux detector thimble tubes inspected during subsequent ECT examinations will be less than 100%, the staff requests that SNC provide its technical basis for reducing the percentage of tubes inspected during implementation of the program.

RAI B.5.2-3

In SNC's response to NRC Bulletin No. 88-09, dated November 2, 1988, SNC stated that the program included ECT at each refueling outage until adequate confidence is established in wear rate projections. In an audit trip report issued on January 12, 1990, the staff stated that SNC's inspection frequency of every refueling outage is acceptable. However, during the audit of November 3-7, 2003, the staff determined that the applicant is basing its implementation of Flux Detector Thimble Inspection Program on the analysis in Westinghouse Proprietary Class 2 Topical Report WCAP-12866, "Bottom Mounted Instrument Flux Thimble Wear" dated January 11, 1991. The staff seeks confirmation that the analysis in WCAP-12866 has not changed SNC's inspection frequency for the Flux Detector Thimble Inspection Program from that approved in the audit trip report of January 12, 1990. If analysis described WCAP-12866 has revised the inspection frequency for the Flux Detector Thimble Inspection Program from a frequency of once every refueling outage, state what the new inspection frequency is and provide the technical basis (i.e., using wear rate projections to support a less frequent basis) for supporting the conclusion that the new inspection frequency will be capable of monitoring for the integrity of the thimble tubes prior to a loss of thimble tube function.

RAI B.5.2-4

The NRC previously approved the Flux Detector Thimble Inspection Program in an NRC Audit Trip Report dated January 12, 1990. In the audit trip report, the staff determined that the applicant was basing its evaluations of wear on an acceptance criterion of 65% through-wall wear in the thimble tubes. However, during the staff's audit of November 3-7, 2003, the staff verified that SNC is currently using Westinghouse Proprietary Class 2 Topical Report WCAP-12866, "Bottom Mounted Instrument Flux Thimble Wear [January 11, 1991] as its current design basis document for evaluating wear that may be detected in the Farley flux thimble tubes as a result of SNC's implementation of the Flux Detector Thimble Inspection Program. This WCAP uses an acceptance criteria of 80% through-wall wear-induced degradation as its basis for performing the evaluations of wear in the Farley flux detector thimble tubes. The staff requests that SNC provide further technical justification and a technical basis for changing the acceptance criterion for the Flux Detector Thimble Inspection Program from 65% through-wall wear and for concluding that 80% through-wall wear is considered to be acceptable for maintaining the component intended functions of the flux detector thimble tubes. The applicant is requested to Include in the technical justification, as appropriate, an assessment of whether or not the establishment of an 80% through-wall acceptance criterion is in conformance with the minimum acceptable wall thickness criterion for the thimble tubes (including allowances to take into account wear that is projected to occur in the thimble tubes during the interval that occurs between examinations and NDE uncertainties).

RAI B.5.2-5

In Section B.5.2 of the LRA, SNC indicated that the original flux detector thimble tubes at Farley Unit 1 were replaced during Unit 1 refueling outage No. 15 with thimble tubes fabricated from chrome-coated, strain-hardened stainless steel. Clarify whether the wear experience for the thimbles tubes at Farley Unit 1 or the change in the material of fabrication for the flux detector thimble tubes at Farley Unit 1 have been used as a basis for revising the [Scope of Program], the [Monitoring and Trending], and the [Acceptance Criteria] program attributes for the Flux Detector Thimble Inspection Program, as implemented for the Farley Unit 1. If the wear experience or the change in the material of fabrication for the flux detector thimble tubes at Farley Unit 1 have been used to revise the [Scope of Program], [Monitoring and Trending], and [Acceptance Criteria] program attributes, as implemented for Farley Unit 1, clarify further and discuss the responses to D-RAIs B.5.2-1, B.5.2-2, and B.5.2-3 whether and how the [Scope of Program], [Monitoring and Trending], and [Acceptance Criteria] program attributes for the Flux Detector Thimble Inspection program differ from Unit 1 to Unit 2, if at all.

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