March 30, 2004

LICENSEE: Southern Nuclear Operating Company

FACILITY: Joseph M. Farley Nuclear Plant, Units 1 and 2

SUBJECT: SUMMARY OF A MEETING AND A TELEPHONE CONFERENCE ON MARCH 10, 2004, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND THE SOUTHERN NUCLEAR OPERATING COMPANY CONCERNING DRAFT REQUESTS FOR ADDITIONAL INFORMATION ON JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (TAC NOS. MC0774 AND MC0775)

The U.S. Nuclear Regulatory Commission staff and representatives of Southern Nuclear Operating Company (SNC or the applicant) held a meeting and a telephone conference on March 10, 2004, to discuss draft requests for additional information (D-RAIs) concerning the Joseph M. Farley Nuclear Plant (FNP) license renewal application.

The meeting and the conference call were useful in clarifying the intent of the staff’s questions. On the basis of the discussions, the applicant was able to better understand the staff's questions. No staff decisions were made during the meeting or telephone conference. In some cases, the applicant agreed to provide information for clarification.

Enclosure 1 provides a list of the meeting and telephone conference participants. Enclosure 2 contains a listing of the D-RAIs discussed with the applicant, including a brief description on the status of the items. The applicant has had an opportunity to review and comment on this summary.

/***RA***/

Tilda Y. 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

Enclosures: As stated

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LIST OF PARTICIPANTS FOR MEETING AND TELEPHONE CONFERENCE ON DRAFT REQUESTS FOR ADDITIONAL INFORMATION

March 10, 2004

Telephone Conference (10 am)

Participants	Affiliation
Tilda Liu	NRC
James Medoff	NRC
J. Hornbuckle	NRC
Jan Fridrichsen	SNC
Mike Macfarlane	SNC
Chuck Pierce	SNC
Bill Evans	SNC
Louis Bohn	SNC
Rick Mullins	SNC
Willie Jennings	SNC
W. Lundeford	SNC
Tom Laubeham	Westinghouse
K. Knight	Westinghouse
L. Stern	Westinghouse

Meeting (1 pm)

Participants	Affiliation
Tilda Liu	NRC
David Chen	NRC
Raman Pichumani	NRC
Goutam Bagchi	NRC
Rex Wescott	NRC
Jacquan Walker	NRC
Jan Fridrichsen	SNC
Bill Evans	SNC

REVIEW OF LICENSE RENEWAL APPLICATION (LRA) FOR FARLEY UNITS 1 AND 2
DRAFT REQUESTS FOR ADDITIONAL INFORMATION (D-RAIs)

March 10, 2004
Via Telephone Conference

B.5.1: Reactor Vessel Internals Program

On February 13, 2004, the staff discussed an alternate option for defining the component locations and examinations that will be performed as part of implementation of the applicant's Reactor Vessel (RV) Internals Program. During this phone call, the staff informed the applicant that this alternative viable option would use a commitment for industry participation (including implementation of industry recommendations) and submittal of an inspection plan for the RV internals as the basis for defining the component locations and inspections that will be implemented as part of the aging management program (AMP). The applicant indicated that it would discuss the applicability of this option to the Farley LRA with its management, and informed the staff of its proposed action at a future conference call.

During the conference call on March 10, 2004, the applicant informed the staff that it agreed to apply this option to the Farley LRA.

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

D-RAI 3.1.3.1.1-1

a. The staff requests additional information on the applicant's aging management reviews (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 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 hold down 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 nominal pipe size (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 man way 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 hold down 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 reactor coolant system (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.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

D-RAI 3.1.3.1.2-1

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 Units. 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 man way 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 units.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

D-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.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

Section 4.2: Reactor Vessel Neutron Embrittlement

Section 4.2.1: Neutron Fluence

D-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.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

Section 4.2.3: Pressurized Thermal Shock

D-RAI 4.2.3.2-1

In License Renewal Application Table 4.2.3-1, Southern Nuclear Operating Company (SNC or the applicant) did not appear to apply the "ratio procedure" of 10 CFR 50.61(c)(2)(ii)(B) when it performed its calculation of the 54 EFPY RT_{PTS} value for Reactor Vessel (RV) Intermediate Shell Axial Weld Seams 19-894 A & B (Heat No. 33A277). In this case, SNC calculated the 54 EFPY RT_{PTS} value for Intermediate Shell Axial Weld Seams 19-894 A & B to be 129°F, which was in exact agreement with the 54 EFPY RT_{PTS} value calculated by the staff for the welds when the "ratio procedure" was not included as part of the staff's basis for the RT_{PTS} calculation. When the "ratio procedure" of 10 CFR 50.61(c)(2)(ii)(B) was applied by the staff to its independent calculation of the 54 EFPY RT_{PTS} value for the welds, the staff calculated the 54 EFPY RT_{PTS} values to be 217°F, which causes the welds to become the limiting 54 EFPY beltline materials for PTS at Farley Unit 1. SNC's omission to apply the "ratio procedure" to the calculation of the 54 EFPY RT_{PTS} value for Intermediate Shell Axial Weld Seams 19-894 A & B is not in compliance with the requirements of 10 CFR 50.61(c)(2)(ii)(B) and must be corrected for the application to ensure that the 54 EFPY RT_{PTS} value for the RV beltline materials will continue to be calculated correctly in accordance with the requirements of 10 CFR 50.61.

Response: Based on the discussion with the applicant, the staff indicated that it will need to conduct additional research and that this question will be deferred for clarification in a future conference call or a meeting. The applicant agreed to the staff's proposal.

D-RAI 4.2.3.3-1

SNC's FSAR Supplement summary description for the time-limited aging analysis of pressurized thermal shock (i.e., TLAA for PTS, as described in Section A.4.1.2 of the application) summarizes the applicable PTS requirements that must be met to ensure continued compliance with 10 CFR 50.61 and discusses why the RV beltline materials at Farley Units 1 and 2 will be in compliance with the applicable requirements in 10 CFR 50.61, as projected through the expiration of the extended periods of operation for the units. However, the limiting RT_{PTS} value cited by the applicant for Farley Unit 1 in the FSAR Supplement summary description (i.e., 191°F for Lower Shell Plate B6919-1) was determined using a process that was not in compliance with the applicable requirements of 10 CFR 50.61 for calculating the RT_{PTS} values operating reactors and is therefore not conservative (Please refer to the compliance issue raised in RAI 4.2.3.2-1). The limiting RT_{PTS} value cited by the applicant for Farley Unit 1 in the FSAR Supplement summary description must be corrected appropriately pending acceptable resolution of RAI 4.2.3.2-1. In addition, 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 by the applicant in Table 4.2.3-2 of the application (i.e., 208°F, based on Intermediate Shell Plant B7212-1) and must be corrected appropriately.

Response: Based on the discussion with the applicant, the staff indicated that it will need to conduct additional research and that this question will be deferred for clarification in a future conference call or a meeting. The applicant agreed to the staff's proposal.

Section 4.2.4: Adjusted Reference Temperatures

D-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. In addition, according to independent calculations performed by the staff, it appears that SNC also did not apply the "ratio procedure" in Position 2.1 of RG 1.99, Revision 2, as part of its process for calculating the 1/4T RT_{NDT} value for Intermediate Shell Axial Welds 19-894 A & B (Heat No. 33A277), which are represented in the Farley Unit 1 RV Surveillance Program.

There is a regulatory basis for requiring that the "ratio procedure" be applied to the 1/4T and 3/4T RT_{NDT} calculations for the beltline welds represented in the Farley Unit 1 and 2 RV Surveillance Programs. The omission to apply the "ratio procedure" to the RT_{NDT} calculations is a non-compliance with the provisions of Technical Specification 5.6.6 for the Farley Pressure Temperature Limits Report (PTLR), and may effect both the current set (34 EFPY) and extended set (54 EFPY) of pressure-temperature (P-T) limit curves administratively controlled in the PTLR. Technical Specification 5.6.6 invokes the methodology of WCAP-14040-NP-A for calculating the Farley P-T limits that are administratively controlled in the Farley PTLR. The methods of WCAP-14040-NP-A require that the 1/4T and 3/4T RT_{NDT} calculations be performed in accordance with the methods of Regulatory Guide (RG) 1.99, Revision 2 [May 1988]. For welds represented in the RV Materials Surveillance Program, the methods of RG 1.99, Revision 2, dictate that the “ratio procedure” be applied to the surveillance data calculations when there is “clear evidence that the copper or nickel content for the surveillance welds differs from that of the vessel weld.”

According to Table 1 of WCAP-14689, Revision 6, “Farley Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation,” this is the case for the surveillance weld representing Intermediate Shell Axial Welds 19-894 A and B in the Farley Unit 1 RV Material Surveillance Program. The copper and nickel values listed for Intermediate Shell Axial Welds 19-894 A and B (Heat No. 33A277) are 0.258% Copper and 0.165% Nickel. The copper and nickel values listed for corresponding surveillance welds fabricated from this heat are 0.140% Copper and 0.190% Nickel. According to the staff’s independent calculations, when these copper and nickel values, the “credible” surveillance data, and the “ratio procedure” are applied to the 1/4T and 3/4T RT_{NDT} calculations for Farley Unit 1 Intermediate Shell Axial Welds 19-894 A and B, the 1/4T and 3/4T RT_{NDT} values for the welds at 54 EFPY will be 192.6°F and 141.9°F, respectively. This includes application of a “Margin Term” of 44°F to the calculations. Application of the “ratio procedure” to the 1/4T and 3/4T RT_{NDT} values calculations causes the beltline weld materials to be the limiting materials at the 1/4T location of the Farley Unit 1 RV at 54 EFPY, while Lower Shell Plate 6919-1 would remain as the limiting RV beltline materials for the 3/4T location. Since the “ratio procedure” was not applied to the RT_{NDT} calculations and since 3/4T RT_{NDT} calculations were not included in the application for the Farley Units 1 and 2 beltline materials, the staff requests that the applicant provide its calculations of the 1/4T and 3/4T RT_{NDT} values for the RV beltline materials at Farley Units 1 and 2 through 54 EFPY. This includes proper application of the “ratio procedure” in Regulatory Position 2.1 of RG 1.99, Revision 2, if any of the 1/4T and 3/4T RT_{NDT} values for the RV weld materials are based on available surveillance data for the weld materials, as obtained through implementation of the applicant’s RV Surveillance Program (i.e., 10 CFR Part 50, Appendix H, program). The staff also requests that SNC revise the description in Section 4.2.4 of the application to identify what the limiting 1/4T and 3/4T RV beltline materials are for Farley Unit 1 and RVs and to list what the RT_{NDT} values are for these limiting RV beltline materials through 54 EFPY, as calculated in conformance with the guidelines of RG 1.99, Revision 2. The impact of this omission needs to be assessed for the set of 54 EFPY P-T limit curves in the Farley PTLR.

NOTE: In addition, this omission also impacts the calculation of the 1/4T and 3/4T RT_{NDT} values for these welds at 34 EFPY (which have been calculated by the staff to be 168.5°F and 118.8°F, respectively). The impact of the omission to apply the ratio procedure to the calculations of the 1/4T and 3/4T RT_{NDT} values for Unit 1 Intermediate Shell Axial Welds 19-894 A and B (Heat No. 33A277) should be evaluated for the set of 34 EFPY P-T limit curves in the Farley PTLR.

Response: Based on the discussion with the applicant, the staff indicated that it will need to conduct additional research and that this question will be deferred for clarification in a future conference call or a meeting. The applicant agreed to the staff’s proposal.

<u>D-RAI 4.2.4.3-1</u>

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.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

Section 4.3.2: Fatigue of the Reactor Coolant Pump Flywheel

<u>D-RAI 4.3.2.2-1</u>

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.

Response: The applicant indicated that the question is clear. The staff informed the applicant and the applicant agreed that this D-RAI will be combined with other D-RAIs that are of CLARIFYING and/or CONFIRMATORY in nature, and will be sent under one umbrella RAI.

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

<u>D-RAI B.5.2-1</u>

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.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

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 will 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.

Response: The applicant indicated that the question is clear. The staff informed the applicant and the applicant agreed that this D-RAI will be combined with other D-RAIs that are of CLARIFYING and/or CONFIRMATORY in nature, and will be sent under one umbrella RAI.

D-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.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

D-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 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).

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

D-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.

Response: The applicant indicated that the question is clear. This D-RAI will be sent as a RAI.

REVIEW OF LICENSE RENEWAL APPLICATION (LRA) FOR FARLEY UNITS 1 AND 2
DRAFT REQUESTS FOR ADDITIONAL INFORMATION (D-RAIs)

March 10, 2004
Via Meeting

Section 4.5.1: Ultimate Heat Sink Silting Calculations

D-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) silting problem. 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 details of the UHS surveillance program including the actual (periodic) measurements of silting in the past and the projected rate of silting in the future at FNP.

b. Provide the updated calculations to show that the UHS will perform its intended function for the extended period of 20 years of operation of the Farley Nuclear Plant.

Response: The applicant brought in the following documents and was provided to staff for review during the meeting:

- FNP-0-ETP-4385: “Service Water Storage Pond Volume Survey Evaluation”
- SM-ES-1500-001: “Ultimate Heat Sink: Depth vs. Volume and Depth vs. Surface Area Curves”
- FNP-0-ETP-4338: “Service Water Storage Pond Sounding Survey”

Based on the discussion, the staff and the applicant agreed that this question will be revised to read as follows:

> 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.

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cc:

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