April 2, 2004

LICENSEE: Southern Nuclear Operating Company

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

SUBJECT: SUMMARY OF TELEPHONE CONFERENCES ON MARCH 22, 24 and 25, 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 telephone conferences on March 22, 24 and 25, 2004, to discuss requests for additional information (RAIs), draft requests for additional information (D-RAIs), and questions concerning the Joseph M. Farley Nuclear Plant (FNP) license renewal application.

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

Enclosure 1 provides a list of the telephone conference participants. Enclosure 2 contains a listing of the RAIs, D-RAIs, questions 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

cc w/enclosures: See next page

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March 22, 2004

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David Chen	NRC
Duc Nguyen	NRC
Amar Pal	NRC
Paul Gill	NRC
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Jan Fridrichsen	SNC
Cary Martin	SNC
Mark Crisler	SNC

March 24, 2004

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March 25, 2004

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Tilda Liu	NRC
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Tim Steingass	NRC
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REVIEW OF LICENSE RENEWAL APPLICATION (LRA) FOR FARLEY UNITS 1 AND 2 REQUESTS FOR ADDITIONAL INFORMATION (RAI)

March 22, 2004

Section 2.5: Scoping and Screening Results: Electrical and Instrumentation and Controls Systems

The staff asked the applicant to clarify the Station Blackout (SBO) recovery path as indicated in the red line in the drawings D-166970L, Sheets 1, 2, and 3, and Drawing D-173096L, Sheet 1, provided in LRA compact disc.

Discussion: The applicant clarified this question during the discussion. The staff and the applicant agreed that this question should be sent as a formal RAI and it will be stated as follows:

<u>RAI 2.5-1</u>

Interim Staff Guidance (ISG) -2, “NRC Staff Position on the License Renewal Rule (10 CFR 54.4) as it relates to the Station Blackout Rule (10 CFR 50.63)”, states, in part, that “The offsite power systems consist of a transmission system (grid) component that provides a source of power and a plant system component that connects that power source to a plant’s onsite electrical distribution system which power safety equipment. For the purpose of the license renewal rule, the staff has determined that the plant system portion of the offsite power system that is used to connect the plant to the offsite power source should be included within the scope of the rule.” Provide a detail description of the Farley Nuclear Plant (FNP) recovery path and discuss how the recovery path is in compliance with ISG-2. The discussion should also include restoration of power to each 4.16 kV safety bus. Clarify how startup transformers 2A, 1A, and 1B are fed from the offsite power source without using breakers 830, 820, and 800.

REVIEW OF LICENSE RENEWAL APPLICATION (LRA) FOR FARLEY UNITS 1 AND 2 REQUESTS FOR ADDITIONAL INFORMATION (RAI)

March 24, 2004

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

D-RAI 3.1.3.3-1

Neither Tables 3.1.1 nor 3.1.2-3 list the cast austenitic stainless steel (CASS) pressurizer spray head assembly as being susceptible to cracking due to thermal fatigue or that a time-limited aging analysis (TLAA) exists to address aging management for this component. For this component and commodity group, (IVC.2.5.4) GALL recommends a TLAA to address cumulative fatigue damage. Provide further information as to whether this plant specific component is susceptible to the aging effect requiring management.

Discussion: During the discussion, the staff indicated this D-RAI will be renumbered from 3.1.3.3-1 to 3.1-4. The applicant indicated the question is clear. This D-RAI will be sent as a formal RAI.

Section 3.5: Aging Management of Containments, Structures, and Component Supports

D-RAI 3.5-13

For ASME Class 1, 2 and 3 piping and components support members, NUREG-1801, GALL Report, calls for ASME Section XI, Subsection IWF Program to manage aging effects due to loss of material, pitting and general corrosion of carbon steel support members, welds, bolted connections and support anchorages (refer to GALL Report III B1.1.1-a and III B1.2.1-a). However, in Table 3.5-9 (page 3.5-62) of the LRA, the applicant credited Structures Monitoring Program instead of the Inservice Inspection Program for managing aging of the same support members/elements. The applicant is requested to discuss the basis for taking such exceptions to the GALL Report.

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

D-RAI 3.5-14

For constant and variable load spring hangers, guides, stops, sliding surfaces and vibration isolators listed in Table 3.5.2-9 of the LRA, GALL Report calls for ASME Section XI, Subsection IWF for aging management of these components; whereas FNP opted to credit Structures Monitoring Program for managing aging of these components. Additionally, item 3.5.1-32 in Table 3.5.1 of the LRA states a position, under its discussion column, that FNP does not consider loss of material function to be an aging effect requiring management based on the plant operating experience, contrary to that of the GALL Report (refer to GALL Report Sections III B1.1.3-a, III B1.2.2-a and III B1.3.2-a). The applicant is requested to justify these deviations from the GALL Report.

Discussion: Based on the discussion with the applicant, the staff indicated the phrase "...loss of material function..." will be reworded as "...loss of mechanical function...". The applicant indicated the question is clear. This D-RAI will be sent as a formal RAI.

Appendix B 5.8: NiCrFe Component Assessment Program

D-RAI B.5.8-1

Under Appendix A2.18 of the LRA, the applicant stated that it will implement the new NiCrFe Component Assessment Program (NCAP) prior to the period of extended operation. In its commitment, the applicant stated that the NiCrFe Component Assessment Program will be developed to address industry concerns regarding the potential for primary water stress corrosion cracking (PWSCC) in nickel alloy components exposed to the reactor coolant environment.

The applicant's commitment needs to reflect that the lessons learned from industry initiatives and research will become part of the NCAP. The applicant is requested to modify commitment A2.18 to assure that interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and it's final version will be used as part of the basis for the NCAP. The commitment should state that the NCAP will be submitted with sufficient time for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging per 10 CFR 54.21(a)(3).

Discussion: See pg. 4

D-RAI B.5.8-2

The applicant stated that the NCAP program scope includes nickel base alloy reactor coolant pressure boundary components with known or potential susceptibility to PWSCC, excluding steam generator tubes, which are specifically addressed by the Steam Generator Program, and Reactor Internals which are addressed by the Reactor Vessel Internals Program. The Program Scope submitted in the application did not include NRC Bulletins 2002-01, 2002-02, 2003-02 and the first Revised Order EA-03-009 as part of the current licensing basis (CLB) for the NCAP. Therefore, the staff is requesting the following actions from the applicant:

a. Update the Program Scope to include the responses to NRC Bulletin 2002-01, 2002-02, 2003-02 and Order EA-03-009 (including First Revised Order EA-03-009).

b. Either summarize the scope and results of inservice inspections and additional augmented (if applicable) examinations that have been performed on FNP reactor vessel heads to date and comment on the impact the inspection results will have on the program attributes for the NCAP; or if the responses to the generic communications provide this type of information, reference the responses to these generic communications and that the lessons learned will be incorporated into the NCAP.

Discussion: See pg. 4

D-RAI B.5.8-3

The applicant stated that the NCAP does not contain any direct preventative or mitigating attributes and that the Water Chemistry Control Program provides prevention attributes. Material replacement was also stated as an available option to prevent or mitigate the potential for PWSCC. The Preventative Actions did not clearly indicate what actions and materials for replacement that would be used, and therefore, demonstrate acceptable management of age-related degradation mechanisms (ARDMs). The applicant is requested to update the Preventative Actions section to include examples of actions taken or to be taken to prevent ARDMs and types of materials used for replacement.

Discussion: See pg. 4

D-RAI B.5.8-4

The applicant stated that the NCAP will not directly inspect or monitor cracking within NiCrFe alloy components and that the program assessment will utilize the most current industry susceptibility models to develop a set of plant specific inspection requirements to address potential PWSCC in FNP NiCrFe components. Since the NCAP is a plant specific program that ranks components susceptible to PWSCC rather than a condition monitoring or performance programs as defined by Branch Technical Position RLSB-1, the applicant needs to demonstrate that the ranking of the components for susceptibility to PWSCC is appropriate and consistent with industry experience and regulatory requirements. The applicant is requested to submit any ranking of components performed to date to assist the staff in determining if the NCAP demonstrates the ability to rank nickel based alloys for testing to find ARDMs consistent with current Regulatory guidelines and industry experience.

Discussion: See pg. 4

D-RAI B.5.8-5

The applicant stated that NCAP will not directly detect or size PWSCC cracks within the NCAP components, but will be used to recommend augmented inspection locations, schedules and techniques based upon the capability of detecting tight PWSCC type cracks prior to any loss of component intended function. These techniques may include visual, surface, or volumetric methods. The staff expects that aging effects can be detected before there is a loss of intended function. This expectation is based on operating experience to date, and through a combination of engineering evaluation to predict PWSCC, periodic visual inspection, and nondestructive testing to validate predictions. This program element describes the “when, where, and how” aspects of program data are collected. The applicant stated that the intent of this element was to detect the effects of aging prior to loss of component intended function without the justification to support the program’s ability to accomplish this.

The applicant is requested to provide justification including codes and standards referenced that the technique and frequency used in the NCAP are adequate to detect the aging effects before a loss of system or component function occurs.

Discussion: See pg. 4

D-RAI B.5.8-6

The applicant stated that the cracking susceptibility assessment, subsequent identification of enhanced inspection requirements, and any initial inspections will be performed for both Units 1 and 2 prior to entering the period of extended operation. The applicant stated that program inspections will be integrated with FNP ISI Program inspections and results will be tracked within the FNP ISI Plan. The staff finds that this element does not provide adequate detail to assure that the effects of PWSCC are managed. For instance, the element does not discuss how the data/results are evaluated against the acceptance criteria and a prediction regarding the rate of degradation is made. This prediction is made to confirm that the timing of the next scheduled inspection will occur before a loss of the system/component intended function.

The applicant is requested to discuss in detail how the results of the inspections performed in the inservice inspection (ISI) Program will be used in trending to assure the frequency of inspections in the NCAP are adequate to detect the ADRM prior to loss of component/system function.

Discussion: See pg. 4

D-RAI B.5.8-7

The applicant stated that the acceptance criteria for any flaws identified will be based upon ASME Section XI requirements or other acceptable fracture mechanic methods. If the flaw is to remain in service, the acceptance evaluation will consider component stresses, updated crack growth rate models, and material toughness. As a minimum, the applicant is required by 10 CFR 50.55a to comply with the flaw acceptance criteria specified for ASME Class 1 components in the ASME Code, Section XI, Articles IWA-3000 and IWB-3000, regardless of whether the material is fabricated from Alloy 600. The applicant may use alternative acceptance criteria either by the applicant or the industry if the alternative criteria have been submitted to and accepted by the staff pursuant to 10 CFR 50.55a(a)(3). The acceptance criteria stated was not definitive enough to determine if the applicant would allow pressure boundary leakage even if the fracture mechanics analysis proved that the component could perform its intended function.

The applicant is requested to discuss the process for calculating specific numerical values of conditional acceptance criteria to ensure that the structure and component intended functions will be maintained under all CLB design conditions. The discussion should include how pressure boundary leakage due to PWSCC will be handled.

Discussion: The applicant indicated these questions are clear. The staff indicated that D-RAIs B.5.8-1 to B.5.8-7 will be combined into one RAI and will be stated as follows and sent to the applicant as a formal RAI.

> RAI B.5.8-1
>
> Under Appendix A2.18 of the LRA, the applicant stated that it will implement the new NiCrFe Component Assessment Program (NCAP) prior to the period of extended

operation. In its commitment, the applicant stated that the NiCrFe Component Assessment Program will be developed to address industry concerns regarding the potential for primary water stress corrosion cracking (PWSCC) in nickel alloy components exposed to the reactor coolant environment.

The applicant's commitment needs to reflect that the lessons learned from industry initiatives and research will become part of the NCAP. The applicant is requested to modify commitment A2.18 to state that the NCAP will be submitted with sufficient time prior to the period of extended operation in order for staff review and approval to determine if the program demonstrates the ability to manage the effects of aging in Alloy 600 components per 10 CFR 50.54.21(a)(3). Also add a commitment that interim report "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds," and/or its final version, will be used as part of the basis for the NCAP when the ranking of components' susceptibility to PWSCC is performed.

**REVIEW OF LICENSE RENEWAL APPLICATION (LRA) FOR FARLEY UNITS 1 AND 2
REQUESTS FOR ADDITIONAL INFORMATION (RAI)**

March 25, 2004

Section 4.5.2: Leak-Before-Break Analysis

<u>D-RAI 4.5.2-1</u>

Since the V. C. Summer main coolant loop weld cracking event involving Alloy 82/182 weld material, the staff has been addressing the effect of primary water stress corrosion cracking (PWSCC) on Alloy 82/182 piping welds on a generic basis for all currently operating PWR plants. To resolve this current operating issue, the industry is taking the initiative to (1) develop overall inspection and evaluation guidance, (2) assess the current inspection technology, and (3) assess the current repair and mitigation technology. An interim industry report, “PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 82/182 Pipe Butt Welds,” was published in April 2001 to justify the continue operation of PWR plants while the industry completes the development of the final report. The staff documented its acceptance of this interim report in a safety evaluation issued on June 14, 2001. The final industry report on this issue has not yet been published. Pending its receipt of the final report and additional UT inspection data from piping involving Alloy 82/182 weld material from the industry, the staff is pursuing resolution of this current operating issue pursuant to 10 CFR Part 50.

The applicant is requested to (1) identify the locations in the FNP RCS piping that contain Alloy 82/182 welds, and (2) describe actions it has taken to address this operating experience.

Discussion: Based on the discussion with the applicant, the applicant indicated that it will need to involve its vendor, Westinghouse, and that this question will be deferred for clarification in a future conference call or a meeting. The staff agreed to the applicant’s proposal.

cc:

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