November 3, 2003

Mr. J. B. Beasley, Jr.
Vice President - Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 — REQUEST FOR ADDITIONAL INFORMATION RE: RISK INFORMED INSERVICE INSPECTION PROGRAM (TAC NOS. MC0178 AND MC0179)

Dear Mr. Beasley:

On October 6, 2003, during a conference call between Southern Nuclear Operating Company (SNC) and the U.S. Nuclear Regulatory Commission (NRC) staff, we discussed NRC staff's concerns related to our review of your July 17, 2003, submittal related to Farley Nuclear plant, Units 1 and 2 (FNP), Inservice Inspection Program. Your submittal requested our approval of a Risk-Informed Inservice Inspection Program (RI-ISI) as an alternative to the FNP ISI Program requirements of the American Society of Mechanical Engineers (ASME) Code Category B-F, B-J, C-F-1, and C-F-2 piping. The NRC staff requests the following information in order to continue with its review:

1. SNC has reported that Resistance Temperature Detectors were installed on the top and bottom of unisolable reactor coolant system (RCS) branch piping identified in SNC's response to NRC Bulletin 88-08 for piping believed to be susceptible to thermal stratification and cycling. If the top-to-bottom temperature differential exceeds a predetermined value, the licensee determines the cause and the potential damage to the piping. For these monitored lines, SNC did not consider the potential for thermal stratification and cycling in the structural reliability and risk assessment (SRRA) failure probability assessments. Therefore, the potential for thermal stratification and cycling were not reflected in the calculated risk ranking values. The Interim Thermal Fatigue Management Guideline (MRP [Materials Reliability Program] - 24) was used to screen those lines that were not monitored for potential thermal fatigue cracking.

   The NRC staff agrees that implementation of continuous temperature monitoring can help reduce the probability of pipe failure caused by thermal stratification and cycling. However, it will not eliminate the potential occurrence of this degradation mechanism nor can it be expected to eliminate any degradation to the piping if the loading occurs. The extent of damage to the pipe will depend on a number of factors including: the cause of the stratification condition, the cyclic nature of the mechanism (e.g., high or low cycle behavior caused by turbulent penetration effects, convection flow, etc.), the time before corrective actions can be implemented to eliminate the stratification load, and the ability to characterize the extent of cracking that might result during exposure to these loading conditions. In addition, synergistic effects resulting from thermal stratification degradation can affect failure probabilities for other cycle fatigue loadings. In light of these concerns, the NRC staff believes that the pipe segment failure probability assessments for the monitored piping and resulting segment risk ranking for these lines should include a contribution for thermal stratification and cycling. Please, discuss whether the SRRA models for these lines need to be revised to include consideration for cyclic thermal stratification. If so, report the revised failure probability estimates and identify any changes to the risk ranking of these segments.

2. SNC has stated that for FNP, each unit has a total of 18 dissimilar metal welds. All the dissimilar metal welds are located in the RCS piping and are in contact with primary coolant. Six reactor pressure vessel (RPV) nozzle safe-end welds and six pressurizer nozzle safe-end welds contain Inconel 82 weld material. The remaining six dissimilar metal welds consist of Inconel 52 buttered hot and cold leg nozzles located on the steam generators (SG). Because of primary water stress corrosion cracking (PWSCC) issues associated with Inconel weld material in contact with reactor coolant, SNC has selected all Inconel 82 welds and the three cold leg SG nozzle Inconel 52 welds for examination. SNC has stated that since Inconel 52 weld material is generally considered to be less susceptible to PWSCC, the remaining three SG Inconel 52 welds located in the same hot leg as the three Inconel 82 RPV outlet nozzle welds were not selected for examination.

   Limited laboratory data suggests that Alloy 52 weld material offers improved resistance to PWSCC over Alloy 82 material. However, current understanding is based on a very limited amount of data on laboratory-prepared specimens. Also, very little service experience has been accumulated for these weld materials in thick section pressurized water reactor (PWR) reactor coolant piping. Recent investigations have found that many weldability issues associated with Alloy 52/152 thick welds are just beginning to be recognized. Significant amounts of ductility dip cracking, lack of fusion and porosity have been observed. Weldability issues like these have resulted in significant numbers of repairs and higher localized residual stresses at the inside surface of the weld. In a recent application, the NRC inspection team concluded that the PWSCC phenomenon for Alloy 52/152 welding material is not fully understood and further studies developing quantitative data should be performed before the new Alloy 52/152 weld can be considered immune to PWSCC. In light of the above discussion and in keeping with fundamental defense-in-depth principals, the NRC staff believes that PWSCC should be treated as an "active" degradation mechanism for all 18 RCS dissimilar metal welds in each Farley unit and a "high failure importance" should be assigned to each of these welds. This is consistent with the definition for high failure importance in Westinghouse Report WCAP-14572, Revision 1, Section 3.7.1, as interpreted in Section 3.4.1 of the NRC Safety Evaluation Report, dated December 15, 1998. Provide further discussion on this issue as it relates to Category B-F welds at FNP, and show justification for not inspecting all dissimilar metal welds in high safety significant segments.

3. SNC has committed to perform the examinations listed in Table 4.1-1 of the WCAP-14572, A-version (WACAP), with the exception of the examinations required for PWSCC. The WCAP lists a visual VT[Visual Testing]-2, performed during system or component pressure tests to detect PWSCC. SNC has noted that VT-2 tests are not volumetric, and as such, will implement VT-2 per ASME Section XI, Table IWB-2500-1.

Also, SNC has committed to performing volumetric or “other appropriate examinations” each interval to detect PWSCC originating from the inside diameter of susceptible piping.

The NRC staff contends, and SNC recognizes, that the visual VT-2 specified in Table 4.1-1 of the WCAP is not appropriate for detecting PWSCC prior to failure of the component having occurred. Given recent industry experience, it is expected that internally-initiated PWSCC will occur at Inconel-bearing dissimilar metal welds (ASME Category B-F) exposed to primary coolant. Therefore, the guidelines in Table 4.1-1 of the WCAP are not acceptable for piping elements susceptible to PWSCC. SNC should confirm that all Category B-F welds susceptible to PWSCC will be volumetrically examined each interval as part of the RI-ISI program. In addition, the licensee should describe what is intended by “other appropriate examinations” that may be applied to these welds.

4. Table 3.4-1 of the WCAP indicates failure probabilities (cumulative for 40 years) as high as 1.04E-01 (small leak) and 4.23E-02 (disabling leak) for the main steam system, and as high as 8.56E-02 (small leak) and 5.26E-02 (disabling leak) for the chemical and volume control system. Failure probabilities at such high levels would suggest that observable damage (small leaks, cracking, or wall thinning) has been observed at FNP or at other plants with similar designs and operating conditions.

    a) Describe the degradation mechanisms and locations in the main steam (MS) and chemical and volume control system (CVCS) that correspond to these values of failure probabilities.

    b) Describe applicable operating experience that would support the high values of failure probabilities listed in Table 3.4-1 of the WCAP for the MS and CVCS.

    c) To what extent are the inspections for the MS and CVCS as listed in Tables 5-1a and 5-1b of the WCAP, directed to locations associated with the high values of failure probabilities listed in Table 3.4-1 of the WCAP?

5. The licensee argues that, in several instances, the RI-ISI program will require examinations that are not currently required by the ASME Section XI program. Examples cited include: (1) Class 1 piping between 2- and 4-inch nominal pipe size (NPS), (2) Class 2 piping less than 4-inch NPS and (3) Class 2 piping greater than 4-inch NPS, but less than 3/8-inch in wall thickness. For the first example, SNC states that the RI-ISI program will now require volumetric examination. However, for examples 2 and 3, SNC simply states that the RI-ISI program will now require examination. Please, clarify the type of examination (volumetric, surface, or visual) that will be applied to these new inspection elements as a result of the RI-ISI process. It should be noted that, if the new examinations are simply visual VT-2, the current ASME Code program contains this requirement; therefore, no new examinations are being implemented.

6. It is noted that Tables 5-1a and 5-1b of the WCAP are intended to summarize and compare new RI-ISI with existing ASME Code examinations, list the relevant degradation mechanisms for elements (examination locations) by plant system, and

include other relevant information. There are several questions related to the information contained in this Table, as follows:

a) In order to determine if appropriate examination methods are being correctly applied to target specific degradation, further clarification is necessary. Please, "break-out" the planned methods for examination (i.e., show how many volumetric or surface examinations will be applied as a result of the RI-ISI process, instead of listing these only as “NDE [non-destructive examination]).”

b) Similarly, describe the type of visual examination that will be applied for those components where “VT” is listed. Since Footnote (a), for Tables 5-1a and 5-1b of the WCAP, specify that VT-2 examinations during system pressure tests will continue to be performed per ASME Code requirements, differentiate between any VT-2 examinations performed as a result of the RI-ISI process and how any new visual VT-2 examinations provide an adequate margin of safety, since they may already be required by the ASME Code. Also, identify if any VT-1 or VT-3 examinations are being applied to the inside surfaces of the subject piping.

c) Identify the type and extent, if any, of the listed “NDE” (volumetric or surface) examinations that are being performed to satisfy existing augmented inspection programs versus being the result of RI-ISI process evaluations.

d) Under the Table column “Degradation Mechanism(s),” it is unclear which mechanisms go with which ASME Code Category welds. Several mechanism designations (MF [Mechanical Fatigue], TF [Thermal Fatigue], SCC [Stress Corrosion Cracking], VF [Vibrational Fatigue]) are multiply listed for several plant systems. Please clarify how to interpret the information in this column.

7. Observations SY-02 - Point 5, and SY-07 from the peer review of the Farley Probabilistic Risk Assessment (PRA) indicate that there appeared to be no common cause failures (CCF’s) modeled between redundant trains with some pumps in standby and others operating (service water and component cooling pumps respectively). SNC states that these CCFs will be modeled in a future revision. Please, explain why the lack of these CCFs models in Revision 5 are not expected to effect the RI-ISI conclusion or otherwise evaluate the potential impact. For example, Observation DA-05 also discusses apparent CCF modeling weakness but SNC provides a reasonable argument that the diesels’ CCF values will have little to no impact on the pipe rupture events that dominate the RI-ISI evaluation.

8. Observation DA-02 notes that there are significant differences between the CCF values used in the Farley PRA (based on CCF estimates developed in the 1990s) and current generic values. Performance of the WCAP uncertainty analysis will not correct for large and potentially inappropriate deviations in mean values. Please identify SNC’s current CCF estimates that vary significantly from the current generic estimates and verify your estimates using the current methodologies and generic estimates or explain why your values are not expected to affect the conclusions of the RI-ISI submittal.

9. Observation HR-04 and HR-05 relate to the lack of calibration error modeling and identify other questionable details (i.e., the use of an 0.1 multiplier) in the human error analyses used in the PRA. The NRC staff's safety evaluation report on the individual plant examination also noted that calibration errors were not included in the models and questioned the limited and probably optimistic treatment of diagnosis and the "blanket" application of selected performance shaping factors (PFS) without case-by-case assessment. Part of your response on calibration errors is that the miscalibration errors are included in the reliability and common cause failures. However, as noted in the previous two questions, the NRC staff has had some reservations with SNC's common cause analyses. Although your submittal indicates that SNC has individually reviewed human errors such that blanked application of PFS may no longer be a concern, the continued used of the multiplicative factors indicate that the human error analyses may not yet be complete and your response to Observation HR-05 indicates that you continue to review the human reliability analysis and will update the models and values as appropriate. Performance of the WCAP uncertainty analysis will not correct for large and potentially inappropriate deviations in mean values. Please explain why these difficulties associated with the human error analysis are not expected to effect the RI-ISI conclusion, or otherwise evaluate the potential weakness.

This request was discussed with B. D. McKinney of your staff on October 14, 2003, and it was agreed that a response would be provided within 30 days of receipt of this letter.

If you have any questions, please contact me at 301-415-1447.

Sincerely,

/RA/

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

cc: See next page

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Sincerely,
/RA/

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Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: As stated

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