April 21, 2004

LICENSEE: Southern Nuclear Operating Company

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

SUBJECT: SUMMARY OF TELEPHONE CONFERENCES ON MARCH 29, 30 AND APRIL 2, 2004, BETWEEN THE U.S. NUCLEAR REGULATORY COMMISSION AND THE SOUTHERN NUCLEAR OPERATING COMPANY CONCERNING DRAFT REQUESTS FOR ADDITIONAL INFORMATION ON THE 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 29, 30 and April 2, 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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LIST OF PARTICIPANTS FOR TELEPHONE CONFERENCES ON DRAFT REQUESTS FOR ADDITIONAL INFORMATION

March 29, 2004

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Tilda Liu	U.S. Nuclear Regulatory Commission (NRC)
David Chen	NRC
James Medoff	NRC
Jan Fridrichsen	Southern Nuclear Operating Company (SNC)
Wayne Lunceford	SNC
Mike Macfarlane	SNC

March 30, 2004 (Morning Call)

Participants	**Affiliation**
Tilda Liu	NRC
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Mike Macfarlane	SNC
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Mike Belford	SNC
Wayne Lunceford	SNC
Lee Stern	Westinghouse
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Dulal Bhowmick	Westinghouse

March 30, 2004 (Afternoon Call)

Participants	**Affiliation**
Tilda Liu	NRC
David Chen	NRC
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Yueh-Li (Renee) Li	NRC
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Harvey Abelson	Information Systems Laboratories (ISL)
Shazia Faridi	ISL
Farideh Saba	ISL
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LIST OF PARTICIPANTS FOR TELEPHONE CONFERENCES ON DRAFT REQUESTS FOR ADDITIONAL INFORMATION (CONT.)

March 30, 2004 (Afternoon Call) cont'd

Participants	**Affiliation**
Charles Pierce	SNC
Bill Evans	SNC
Jon Hornbuckle	SNC
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April 2, 2004

Participants	**Affiliation**
Tilda Liu	NRC
David Chen	NRC
Joe Terrell	NRC
Jan Fridrichsen	SNC
Mike Macfarlane	SNC

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

March 29, 2004

D-RAI B.5.1-1

In RAI 3.1.3.1.2-1, Parts a and b, the staff requested information about the different aging mechanisms that could induce loss of material in the NiCrFe and stainless steel reactor vessel (RV) internal components and justification why the water chemistry program was considered to be capable of managing loss of material in these components, particularly if the "loss of material" aging effect was determined to be induced by a mechanical-type of aging mechanism, such as wear, erosion, or fretting. Pending the applicant's responses to RAI 3.1.3.1.2-1, Parts a and b, the staff seeks confirmation that loss of material will be added to the list of aging effects credited for aging management by the Reactor Vessel Internal Program if, in response to RAI 3.1.3.1.2-1, Parts a and b, the Reactor Vessel Internal Program is credited as an additional program for managing "loss of material" in RV internal components fabricated from stainless steel (including cast austenitic stainless steel) or NiCrFe materials.

Discussion: The applicant indicated and the staff agreed that the correct reference to this question is RAI 3.1.3.1.1-1 and not RAI 3.1.3.1.2-1. The applicant stated that the Reactor Vessel Internals Program is not credited in the LRA for managing loss of material. The applicant further stated that the inservice inspection (ISI) Program is credited to manage wear for reactor vessel internals components (where wear is applicable), with the exception of the flux thimble tubes. The Flux Detector Thimble Inspection Program is credited for wear of the thimble tubes. Therefore, this question is WITHDRAWN and will not be sent as a formal RAI.

D-RAI B.5.1-2

In Section B.5.1 of Appendix B to the Farley LRA, SNC states that the following components are within the scope of the Reactor Vessel Internals Program: (1) baffle and former assemblies, (2) bottom mounted instrumentation cruciforms, (3) core barrel, (4) lower core plate and fuel alignment pins, (5) lower support forging, and (6) lower support column bases. However, in the aging management reviews (AMRs) of Table 3.1.2-2 of the Farley LRA, SNC indicates that the Reactor Vessel Internals Program is credited for aging management of the following RV internal components:

- baffle and former plates
- baffle bolts
- bottom mounted instrumentation (BMI) column cruciforms
- BMI columns with fasteners
- clevis inserts and fasteners
- control rod drive guide tube assemblies with associated fasteners
- core barrel and core barrel flange
- core barrel outlet nozzles
- control rod drive guide tube (CRGT) support pins
- flux thimble tubes
- reactor pressure vessel / head alignment pins with associated fasteners
- head cooling spray nozzles

- HJTC probe holder, probe holder extension, and probe holder shroud assemblies with associated fasteners
- lower core plate and fuel alignment pins
- lower support columns with associated fasteners
- lower support forging
- neutron panels
- radial keys and fasteners
- secondary core support assembly with associated fasteners
- upper core alignment pins with associated fasteners
- upper core plate and fuel alignment pins with associated fasteners
- upper instrumentation conduit and supports with associated fasteners
- upper support assembly with associated fasteners
- upper support column bases
- upper support column with associated fasteners

The components that are within the scope of the Reactor Vessel Internals Program, as described in Section B.5.1 of Appendix B to the Farley LRA, need to be consistent with the list of RV internal components in LRA Table 3.1.2-2 that the aging management program (AMP) is credited for. The staff requests that the scope of Reactor Vessel Internal Program be supplemented to make the list of components within the scope of the AMP consistent with those listed in Table 3.1.2-2 for which the AMP is credited.

Discussion: The applicant indicated that this question was clear. This D-RAI will be sent as a formal RAI.

D-RAI B.5.1-3

In a teleconference on March 10, 2004, (documented in a teleconference summary dated March 30, 2004) SNC stated that it would amend its program description for the Reactor Vessel Internal Program to indicate that the applicant would use its participation in the industry initiatives on RV internals (i.e., industry research studies and activities) as a basis for implementing the Reactor Vessel Internals Program and that the AMP would include a commitment that incorporates the following elements:

a) A commitment to participate in the industry's initiatives of aging of pressurized-water reactor (PWR) RV internal components.

b) A commitment to implement the recommendations for component locations inspected, aging effects monitored for, inspection methods, inspection qualifications, frequency of examinations, number of components inspected, acceptance criteria, and corrective actions, that result from the industry's initiatives on aging degradation of PWR RV internal components.

c) A commitment to submit the inspection plan for the PWR Vessel Internals to the staff for review and approval two years prior to entering the periods of extended operation for the Farley units.

Part 1: The staff seeks confirmation that the commitment made on the Reactor Vessel Internals Program will incorporate the three elements discussed above and that the commitment will be docketed for the Farley units prior to staff's issuance of the Safety Evaluation Report with Open Items for the Farley LRA. This RAI includes a request for confirmation that the FSAR supplement summary description for the Reactor Vessel Internals Program (Chapter A.2.13 of Appendix A to the LRA) will be amended to incorporate the changes to the program that the applicant stated will be made.

Part 2: If SNC decides not to use this alternative approach for aging management of the RV internals, the staff requests that the applicant provided further discussion and justification on the following action items and exceptions taken against GALL Volume 2:

a) Exception to GALL AMP XI.M16 on inspection methods proposed for detection of cracking in baffle/former bolts - Provide further discussion and justification as to why VT-1 inspection methods are considered to be capable of detecting cracking in the bolt shafts when the staff has taken the position that volumetric inspections would be needed for the detection of cracking in the bolt shafts.

b) Exception to GALL AMP XI.M16 on limiting the VT-1 and enhanced VT-1 inspections of the RV internals only to those locations identified as being limiting by industry research and/or operating experience - Provide further identification of the scope of RV internals components and number of components that will be inspected as part of the Reactor Vessel Internals Program and the basis for their inclusion in the scope, and clarify which inspection methods, frequency of inspections, methods for qualifying the inspections will be used for the examinations, as well as the acceptance criteria that will be used to assess the examination results.

c) Action item on the need for a technical analysis to assess the synergistic effects of thermal aging and neutron irradiation embrittlement on the fracture toughness properties for Cast Austenitic Stainless Steel (CASS) RV internals (i.e., in the BMI cruciforms and upper support plate castings) - Assess the synergistic impacts of thermal aging and neutron irradiation embrittlement on the size of flaws that can be tolerated in the CASS RV internals (i.e., assess the flaw tolerance [critical crack size] of the materials based on the synergistic impacts of thermal aging and neutron irradiation embrittlement) and justify how the inspection methods proposed for these components will be capable of detecting a flaw in the components prior to a loss of component intended function.

d) Exception to GALL AMP XI.M13 on the neutron fluence threshold value used for managing the synergistic effects of thermal aging and neutron irradiation embrittlement in the CASS RV internals at Farley Units 1 and 2 - Justify the difference in neutron irradiation embrittlement threshold value assumed for the CASS RV internals from that identified in GALL AMP XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)."

Discussion: Because SNC stated, in a teleconference with the staff held on March 10, 2004 (documented in a teleconference summary dated March 30, 2004), that it would amend its program description for the Reactor Vessel Internal Program to indicate that it would use its participation in the industry initiatives on RV internals (i.e., industry research studies and activities) as a basis for implementing the Reactor Vessel Internals Program, the staff and applicant agreed that only Part 1 of this question is applicable for the purpose of this RAI, and that Part 2 of this question may be removed. Therefore, this D-RAI will be revised as follows and sent as a formal RAI.

> <u>D-RAI B.5.1-3</u>
>
> In a teleconference held on March 10, 2004, (documented in a teleconference summary dated March 30, 2004) SNC stated that it would amend its program description for the Reactor Vessel Internal Program to indicate that the applicant would use its participation in the industry initiatives on RV internals (i.e., industry research studies and activities) as a basis for implementing the Reactor Vessel Internals Program and that the AMP would include a commitment that incorporates the following elements:
>
> a) A commitment to participate in the industry's initiatives of aging of PWR RV internal components.
>
> b) A commitment to implement the recommendations for component locations inspected, aging effects monitored for, inspection methods, inspection qualifications, frequency of examinations, number of components inspected, acceptance criteria, and corrective actions, that result from the industry's initiatives on aging degradation of PWR RV internal components.
>
> c) A commitment to submit the inspection plan for the PWR Vessel Internals to the staff for review and approval two years prior to entering the periods of extended operation for the Farley units.
>
> The staff seeks confirmation that the commitment made on the Reactor Vessel Internals Program will incorporate the three elements discussed above and that the commitment will be docketed for the Farley units prior to the staff's issuance of the Safety Evaluation Report with Open Items for the Farley LRA. This RAI includes a request for confirmation that the FSAR supplement summary description for the Reactor Vessel Internals Program (Chapter A.2.13 of Appendix A to the LRA) will be amended to incorporate the changes to the program that the applicant stated will be made.

<u>D-RAI B.5.1-4</u>

SNC has taken an exception on the number of inspection cycles set forth in Section XI of the ASME Boiler and Pressure Vessel Code, Subsection IWB, for required inspections of RV internal components. This exception must be submitted by the applicant for review and approval in accordance with 10 CFR 50.55a. The staff therefore requests that the applicant withdraw this exception from the application and commit to following the ASME Code until and unless specific relief is granted under the relief request or alternative program provisions of 10 CFR 50.55a.

Discussion: The applicant indicated that this question was clear. This D-RAI will be sent as a formal RAI.

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

March 30, 2004 (Morning Call)

RAI Responses Received for Section 3.4

For its responses to RAI 3.4-1 and RAI 3.4-5, dated March 5, 2004, the applicant made references to Past Precedent Reviews PPR-AMS-M03, “Aging Management for Stainless Steel Components Exposed to a Steam or Treated Water Environment”, and PPR-AMS-M05, “Aging Management for Stainless Steel AFW System Lubricating Oil Coolers”. The staff indicated that these documents are not available to the staff. The applicant agreed and stated that the responses to RAI 3.4-1 and 3.4-5 will be revised to remove the respective references.

Section 4.5.2: Leak-Before-Break Analysis

D-RAI 4.5.2-1

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 ultrasonic testing (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 reactor coolant system (RCS) piping that contain Alloy 82/182 welds, and (2) describe actions it has taken to address this operating experience.

Discussion: The applicant indicated that this question was clear. This D-RAI will be sent as a formal RAI.

D-RAI 4.5.2-2

Section 4.5.2 of the LRA states that for the reactor coolant loop (RCL), Westinghouse revised the WCAP-12825 analysis of the primary loop piping to account for the additional thermal aging of the cast austenitic materials for the period of extended operation and issued Addendum 1 in December, 2002. The applicant is requested to provide Addendum 1 to WCAP-12825 which was reviewed and approved by the staff.

Discussion: The applicant indicated that this question was clear. This D-RAI will be sent as a formal RAI.

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

March 30, 2004 (Afternoon Call)

Section 2.3.3: Auxiliary Systems

<u>D-RAI 2.3.3.16-1</u>

LRA Tables 2.3.3.16 and 3.3.2-16 list filter casings as components that are subject to an AMR. However, license renewal boundary drawings D-175047L and D-205047L do not show any filter as being within the scope of license renewal. Provide drawings or descriptive information that identifies the filter casings in the demineralized water system that are within scope of license renewal and subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1), respectively.

Discussion: The applicant indicated that this question was clear. This D-RAI will be sent as a formal RAI.

<u>D-RAI 2.3.3.19-4</u>

Prevention of internal flooding is not listed as an intended function of the waste disposal system. Verify that none of the floor drains, equipment drains and waste disposal system components are credited in the FNP internal flooding analysis.

Discussion: The applicant indicated that this question was clear. This D-RAI will be sent as a formal RAI.

Section 2.3.4: Steam and Power Conversion Systems

<u>D-RAI 2.3.4.5-1</u>

LRA Section 2.3.4.5 under "License Renewal Drawings" makes reference to only one drawing (D-506447, Sheet 1). This is not a license renewal boundary drawing and, therefore, does not identify components considered to be within the scope of license renewal. An examination of the LRA boundary drawing index identified only one boundary drawing for each unit whose title includes the words "Auxiliary Steam System." (D-175033L, Sheets 1 and 2; D-205033L, Sheets 1 and 2, titled "Main Steam and Auxiliary Steam Systems"). Drawings D-175033L and D-205033L, however, do not demarcate the two systems. Additionally, no drawing listed in the index includes the condensate recovery system in its title.

LRA Tables 2.3.4.5 and 2.3.4.1 list the component types subject to an AMR for the Auxiliary Steam and Condensate Recovery System and the Main Steam System, respectively. In the absence of boundary drawings which demarcate the two systems, the components which comprise the Auxiliary Steam and Condensate Recovery System and those belonging to the Main Steam System cannot be identified. As a result, the staff is unable to determine whether all components subject to an AMR have been identified in these tables, in accordance with NUREG-1800, Section 2.3.3.2.

Provide drawings and/or other documentation that demarcate the boundaries of the Auxiliary Steam and Condensate Recovery System and that will allow the staff to determine whether all components within the scope of license renewal and subject to an AMR (in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1)) have been properly identified. Additionally, provide the locations of the specific components listed in Table 2.3.4.5.

Discussion: The staff indicated that the applicant clarified the drawings in its draft response by providing the appropriate nomenclatures. This D-RAI will not be necessary. Therefore this question will be WITHDRAWN and will not be sent as a formal RAI. The draft response received on February 2, 2004 was as follows:

D-RAI 2.3.4.5 -1

Draft Response:

FNP has two systems that contain the phrase "Auxiliary Steam" in the system title. One is the Auxiliary Steam System which supplies steam to the Auxiliary Feedwater System Turbine Driven Pump (FNP system number N12), and the other is the Auxiliary Steam and Condensate Recovery System (FNP system number P20).

The Auxiliary Steam System is included with the Main Steam System as described in LRA Section 2.3.4.1. The Auxiliary Steam System is the steam supply lines to the turbine-driven auxiliary feedwater pump turbine from the main steam lines. The Auxiliary Steam System is shown on license renewal boundary drawings D175033L sheet 2 (for Unit 1) and D205033L sheet 2 (for Unit 2) and described in UFSAR Sections 10.3.2.1 and 6.5.2.1. LRA Table 2.3.4.1 includes the component types subject to an AMR for the Auxiliary Steam System.

The non safety-related Auxiliary Steam and Condensate Recovery (AS&CR) System is in the scope of license renewal under 10 CFR 54.4(a)(2) considerations for spatial interaction with safety-related SSCs and high energy piping as stated in LRA Section 2.3.4.5. The portion of the system that is in the scope of license renewal for high energy line considerations is described in detail in UFSAR Section 3K.4.2.3.1. The portions of the AS&CR System that are brought into scope for license renewal per criterion 10CFR54.4(a)(2) for spatial interaction concerns were identified on license renewal boundary drawing D506447L (Sh. 1) by listing the room numbers where the spatial interaction concerns occur. Therefore, drawing D506447L sheet 1 identifies the locations of the components listed in LRA Table 2.3.4.5.

SNC elected to depict the 10 CFR 54.4(a)(2) scoping boundary results in a tabular format rather than on P&ID-based drawings since piping diagrams do not include the features necessary to present the results with clarity. For fluid-bearing mechanical SSCs that are only in-scope for a spatial interaction concern with a safety-related SSC, P&ID-based boundary drawings do not depict the spatial relationship with the safetyrelated SSCs that define the basis for the scoping decision. The piping scope would terminate in "mid-air" with no obvious basis without numerous notes. The boundary is defined by the physical layout and location in the plant (e.g., portion of the system that is located in room XYZ). Therefore, SNC has chosen to show the scope of systems with components brought into scope for 10CFR54.4(a)(2) in the tabular format shown on D506447L.

D-RAI 2.3.4.5-3

LRA Section 2.3.4.5 states that the auxiliary steam and condensate recovery system is non-safety related, but is considered within the scope of license renewal due to the potential spatial interactions of high energy piping/components (in the system) with safety related SSCs, in accordance with 10 CFR 54.4(a)(2).

LRA Section 2.3.4.1 (main steam system) states that the “Main Steam System also supplies steam via the Auxiliary Steam System to the Turbine Driven Auxiliary Feedwater Pump” and that “portions of the Main Steam System from each steam generator up to and including the main steam isolation valves and the supply to the Turbine Driven Auxiliary Feedwater Pump are necessary for the safe shutdown of the plant and accident mitigation.” Therefore, since the supply lines (and associated components) from the main steam line to the auxiliary feedwater pump turbine perform a safety-related function, they should be within the scope of license renewal in accordance with 10 CFR 54.4(a)(1)(ii).

Explain why the auxiliary steam and condensate recovery system is considered non-safety related in LRA Section 2.3.4.5.

Discussion: The staff indicated that the applicant clarified the drawings in its draft response by providing the appropriate nomenclatures. This D-RAI will not be necessary. Therefore this question will be WITHDRAWN and will not be sent as a formal RAI. The draft response received on February 2, 2004 was as follows:

D-RAI 2.3.4.5 -3

Draft Response:

The “Auxiliary Steam and Condensate Recovery System” described in LRA section 2.3.4.5 is not the same as the “Auxiliary Steam System” described in LRA section 2.3.4.1. The Auxiliary Steam and Condensate Recovery System draws steam from the main steam headers in the Turbine Building of each unit to supply steam loads such as the condenser air ejectors, ventilation heating, etc. The steam extraction points are downstream of the main steam isolation valves. The Total Plant Numbering System (TPNS) designator for the Auxiliary Steam and Condensate Recovery System is “P20.” As stated in LRA section 2.3.4.5, this system is non safety-related. Refer to LRA section 2.3.4.5 for a complete description of this system.

The “Auxiliary Steam System” described in LRA section 2.3.4.1 draws steam from the Main Steam System of each unit upstream of the main steam isolation valves to provide motive force for each unit’s turbine driven auxiliary feedwater pump. The TPNS designator for the Auxiliary Steam System is “N12.” This system is shown in its entirety on mechanical boundary drawings D-175033L, sheets 1 & 2, and D-205033, sheets 1 & 2. The Auxiliary Steam System piping which supplies steam to the turbine driven auxiliary feedwater pumps is shown as in scope on these drawings. Refer to LRA section 2.3.4.1 for a complete description of this system.

D-RAI 2.3.4.5-4

LRA Table 2.3.4.5 lists "strainers (shell)" as being subject to an AMR. However, after reviewing license renewal boundary drawings D-175033L, sheets 1 and 2; and D-205033L, sheets 1 and 2, the staff is unable to find components of this type on these drawings. The staff is concerned that other drawings (not referenced in the LRA) may contain components of this system that should be included within the scope of license renewal. Identify the drawings that contain the strainers referred to in LRA Table 2.3.4.5. If these drawings have not been provided to the staff previously, provide these drawings to the staff for review.

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

D-RAI 2.3.4.6-1

LRA Section 2.3.4.6 states that, in accordance with 10 CFR 54.4(a)(2), "The non-safety related SCs of the turbine and turbine auxiliaries that are required to trip the turbine in response to an anticipated transient without scram (ATWS) event and in response to a turbine overspeed event are conservatively included in the scope of license renewal for FNP." However, there are no mechanical components of the turbine and turbine auxiliaries system that are identified as being subject to an AMR.

Since LRA Section 2.3.4.6 does not provide or reference any boundary drawings associated with the turbine and turbine auxiliaries system, the staff is unable to confirm your determination that this system does not contain mechanical components subject to an AMR. For the staff to complete its review, provide a description or license renewal boundary drawing that identifies the components of the turbine and turbine auxiliaries system, and that shows which SCs are considered to be within the scope of license renewal in accordance with the requirements of 10 CFR 54.4(a). Justify the exclusion of the mechanical components of this system from being subject to an AMR in accordance with the requirements of 10 CFR 54.21(a)(1).

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

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

April 2, 2004

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

D-RAI 3.1-5

The feedwater ring in FNP's Westinghouse 54F replacement steam generators is equipped with J-tubes. The staff requests to clarify and/or confirm:

a. Are J-tubes included in the feedwater distribution assembly (see LRA Table 3.1.2-4, page 3.1-73) as components subjected to the aging effects of cracking and loss of material?

b. What material comprises the feedwater ring and J-tubes?

Discussion: The staff discussed this issue with the applicant because of potential concerns associated with J-tubes becoming loose parts. The applicant stated the feedwater distribution assembly in FNP's Westinghouse 54F replacement steam generators is not equipped with J-tubes. Instead, it is equipped with a feedwater ring and spargers; the feedwater ring performs the function of the J-tubes. The feedwater ring is made of alloy steel and the sparger tubes are made of Alloy 690. The staff indicated the question has been clarified and that this D-RAI will be WITHDRAWN.

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