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SOUTHERN COMPANY
Energy to Serve Your World℠

April 29, 2004

Docket Nos.: 50-348
50-364

NL-04-0715

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant Units 1 and 2
Application for License Renewal – Requests for Additional Information

Ladies and Gentlemen:

This letter is in response to your letters dated March 29, 2004 and April 1, 2004 requesting additional information for the review of the Joseph M. Farley Nuclear Plant, Units 1 and 2, License Renewal Application. Responses to the Requests for Additional Information (RAIs) are provided in Enclosures 1 and 2. By agreement between Ms. Tilda Liu, NRC Project Manager, NRR, and Mr. Jan Fridrichsen, SNC License Renewal Licensing Project Manager, both the March 29 and April 1 responses are provided by April 29.

Mr. L. M. Stinson states he is a vice president of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

If you have any questions, please contact Charles Pierce at 205-992-7872.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

Sworn to and subscribed before me this 29 *day of* April, *2004.*

Notary Public

My commission expires: 6-7-05

A099

LMS/JAM/slb

Enclosures: 1. Joseph M. Farley Nuclear Plant, Units 1 and 2
Application for License Renewal - Responses to March 29, 2004
Requests for Additional Information

2. Joseph M. Farley Nuclear Plant, Units 1 and 2
Application for License Renewal - Responses to April 1, 2004 Requests
for Additional Information

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley Jr., Executive Vice President
Mr. D. E. Grissette, General Manager – Plant Farley
Document Services RTYPE: CFA04.054; LC# 14023

U. S. Nuclear Regulatory Commission
Ms. T. Y. Liu, License Renewal Project Manager
Mr. L. A. Reyes, Regional Administrator
Mr. S. E. Peters, NRR Project Manager – Farley
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Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

ENCLOSURE 1

Joseph M. Farley Nuclear Plant Units 1 and 2

Application for License Renewal

Responses to March 29, 2004 Requests for Additional Information

RAI 2.5-1

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 the 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 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.

Response

Per ISG-2, the FNP LRA scope for station blackout (SBO) "includes the switchyard circuit breakers that connect to the offsite system power transformers (startup transformers), the transformers themselves, the intervening overhead or underground circuits between circuit breaker and transformer and transformer and onsite electrical distribution system, and the associated control circuits and structures" in the path to restore offsite power.

Startup transformers 1A, 1B, 2A, and 2B can be fed from multiple sources in the 230kV and 500kV switchyards. The path shown on the LRA boundary drawings (D-169970L sheets 1, 2, and 3 and D-173096L sheet 1) utilizes the preferred power source (230kV Bainbridge line) per the Farley Nuclear Plant procedures for restoring Units 1 and 2 offsite power following a wide area blackout. Switchyard circuit breakers 840, 934, 924, and 904 represent the first isolation device upstream from the startup transformers and demarcate the FNP SBO recovery path from the transmission system. This path allows for the restoration of all four startup transformers once the preferred 230kV source is restored. Breakers 830, 820, and 800 could provide a direct path to individual transmission lines, but the use of these breakers would require restoration of multiple transmission lines before all four startup transformers could be powered. The 4.16kV safety buses receive their power from one of the two secondary windings for each startup transformer as shown on D-173096L. A non-segregated cable bus duct system provides the path between the startup transformers and the 4.16kV safety buses.

RAI 3.1-4

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.

Response

For FNP, SNC normally manages cracking due to thermal fatigue through a TLAA or the fatigue monitoring program. However, the spray head assembly is not a part of the pressure retaining boundary as defined by the ASME Boiler and Pressure Vessel Code. Accordingly, SNC does not have fatigue calculations associated with this assembly and no corresponding cumulative fatigue usage factor prediction to maintain for this assembly. Therefore, there is no TLAA to address cumulative fatigue damage for the pressurizer spray head assembly.

Although there is no TLAA for the spray head assembly, the corresponding pressurizer nozzle forms part of the pressure retaining boundary and therefore has a fatigue calculation. Cracking due to thermal fatigue for this spray nozzle is managed through a TLAA and the fatigue monitoring program. The spray nozzle thermal transient cycles correlate to those experienced by the spray head assembly. The management of thermal fatigue for the spray nozzle confirms, for both the spray nozzle and the spray head assembly, the fatigue cycles experienced remain within the original design parameters.

The physical configuration and operating parameters for the FNP spray heads minimize the potential for cracking due to thermal fatigue cycling. The FNP cast austenitic stainless steel spray head is threaded onto a stainless steel coupling which is welded to the pressurizer hemispherically-shaped upper head. Loosening of the spray head threaded joint is precluded by a stainless steel locking bar that is tack welded to the spray head on one end, and the pressurizer upper head at the other end. In this configuration, thermal movement is not restrained with the possible exception of the locking bar tack weld areas. There are no significant pressure stresses present, since the operating pressures on both sides of the spray head are similar. Finally, FNP operates with a continuous flow of coolant through the spray head, thereby significantly reducing the number and magnitude of thermal transients affecting the spray head assembly. Since thermal movement is not restrained, and low fatigue usage is expected, SNC concludes that cracking of the FNP spray head assemblies due to fatigue cycling is not likely.

Table 3.1.2-3 of the LRA identifies cracking as an aging effect requiring management for the pressurizer spray head assembly. A one time visual VT-1 examination of higher stress regions associated with the pressurizer spray head assembly is credited to identify any cracking, whether SCC induced, or due to fatigue cycling.

RAI 3.5-13

For American Society of Mechanical Engineers (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 3.5.2-9 (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.

Response

In LRA Table 3.5.2-9 (page 3.5-62), the Component Type "ASME & Non-ASME Piping and Component Support Members" includes support members for ASME Class 1, 2 and 3 piping and components as well as Non-ASME. SNC offers the following clarification on the applicability of the Inservice Inspection Program (ASME Subsection IWF Program) and the Structural Monitoring Program to this component type.

SNC confirms the FNP ISI Program (which includes the ASME Section XI, Subsection IWF Program) is credited for ASME Class 1, 2 and 3 piping and components support members to manage aging effects due to loss of material. This is consistent with the GALL Report. In addition to the Section XI inspections (not instead of), support members of ASME piping and components (GALL Report items III B1.1.1-a and III B1.2.1-a) are also inspected under the FNP Structural Monitoring Program and therefore were identified as such in the LRA table.

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 mechanical function to be an aging effect requiring management based on the plant operating experience, contrary to that of 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.

Response

SNC offers the following clarification on the applicability of the Inservice Inspection Program (ASME Subsection IWF Program) and the Structural Monitoring Program to the component type "Constant and variable load spring hangers, guides, stops; sliding surfaces; vibration isolators (For ASME piping and components)" in LRA Table 3.5.2-9 (page 3.5-63).

Consistent with the GALL Report (items III B1.1.3-a and III B1.2.2-a), SNC confirms the FNP ISI Program (which includes the ASME Section XI, Subsection IWF Program) is credited to manage aging effects due to loss of material for this component type. SNC also credits the Structural Monitoring Program in addition to (not in lieu of), the Inservice Inspection Program for managing aging of these components. These components are also inspected under the FNP Structural Monitoring Program and therefore were identified as such in the LRA table.

As stated in the discussion column for Item 3.5.1-32 in Table 3.5.1 of the LRA, SNC does not consider loss of mechanical function (due to corrosion, distortion, dirt, overload, etc.) to be an aging effect requiring management at FNP for the Groups B1.1, B1.2 and B1.3 support members, anchor bolts, welds, spring hangers, guides, stops, and vibration isolators. Review of the FNP operating experience did not indicate loss of mechanical function due to corrosion or dirt as an aging effect at FNP. Loss of function due to distortion or overload is a design/corrective action issue and not the result of aging. Although SNC did not identify loss of mechanical function as an aging effect, the aging management program credited by SNC to manage aging in these components (i.e., ISI Program) is consistent with the GALL Report program for managing loss of mechanical function.

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.

Response

FNP FSAR Supplement Appendix A.2.18, NiCrFe Component Assessment Program, will be revised to add the following text:

> FNP will continue to participate in industry initiatives (such as the Westinghouse Owners Group and the EPRI Materials Reliability Program). Susceptibility rankings and program inspection requirements will be consistent with the latest version of the EPRI Materials Reliability Program safety assessment regarding Alloy 82 / 182 pipe butt welds or its successors.
>
> SNC will submit an inspection plan for the NiCrFe Component Assessment Program for NRC review and approval at least 24 months prior to entering the period of extended operation for the FNP units.

ENCLOSURE 2

Joseph M. Farley Nuclear Plant Units 1 and 2

Application for License Renewal

Responses to April 1, 2004 Requests for Additional Information

RAI 2.3.3.16-1

LRA Tables 2.3.3.16 and 3.3.2-16 list filter casings as components that are subject to an aging management review (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.

Response

The filters are included in the scope of license renewal in accordance with 10 CFR 54.4(a)(2) due to a potential spatial interaction with safety related components. Items included in scope for 10 CFR 54.4 (a)(2) are not highlighted on the system boundary drawings. For spatial interaction issues, license renewal boundary drawing D506447L identifies, for each applicable LRA System, the room(s) where the potential spatial interaction occurs.

In this case, the filters, N1P11F001 and N1P11F003, are located in Auxiliary Building Rooms 186 and 342 and have a spatial interaction with safety-related SSCs in those rooms. The filters are shown on license renewal boundary drawing D175047L Sheet 1 at coordinates E9 and F5.

RAI 2.3.3.19-4

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.

Response

The Liquid Waste and Drains features credited in FNP's current licensing basis (CLB) for prevention/mitigation of internal flooding are described in UFSAR Appendix 3K and Section 9.3.3. Features credited include sensors assigned to the Liquid Waste and Drains System that provide line break detection, and sump usage/level alarms for the sumps in the lower elevations of the Auxiliary Building. These features are discussed in the LRA system description for the Liquid Waste and Drains System (Section 2.3.3.19).

As stated in the LRA, SNC has included in scope the line break detection sensors assigned to the Liquid Waste and Drains system as part of the High Energy Line Break Detection System (LRA Section 2.3.3.17). These line break detection sensors are compartment/room pressure sensors that isolate the CVCS letdown line in the event of a CVCS letdown line rupture.

SNC also included in the scope of license renewal the features that provide sump usage indication and level alarms for the sumps in the lower elevations of the Auxiliary Building. UFSAR Section 9.3.3.3 provides the basis for the specific features relied upon in the CLB to monitor sump usage and provide sump high level alarms. In addition to the structural features of the rooms (which includes the sumps), frequency of sump pump operation, number of sump pumps operating, and the sump high level alarms are credited in the CLB for providing the operator with indication of the leak. SNC included the sump pump controls and sump level instrumentation in the scope of LR under the plant-wide electrical commodities. The level switches and mechanical alternators are active components and therefore do not require an aging management review (AMR). The sumps themselves and other physical features such as curbing, platforms, equipment pedestals, walls, and watertight doors are included in scope under the spaces approach used for civil/structural commodities in Section 2.5 of the LRA.

During review of the internal flooding prevention/mitigation scoping results for the Liquid Waste and Drains system in response to this RAI, SNC identified an omission in the LRA. Two rooms evaluated for flooding in the CLB analysis utilize drain piping to connect to a sump in an adjoining area that is relied upon to detect the line failure. This interconnecting piping was omitted from the LRA. This non safety-related piping functionally supports the safety-related use of the sump to detect a line failure and therefore is in the scope of license renewal in accordance with 10 CFR 54.4(a)(2).

For the first room, one of the drain lines is a six (6) inches diameter stainless steel line that is routed through a wall penetration to a fixed connection in the cover of a sump in an adjoining room. The room (without the sump) is water tight, so a leak in the room would fill the room to the height of the drain, the drain would then fill the sump, and the sump usage/level instrumentation would enable the leak to be detected. The inlet of this drain line is slightly more than six (6) inches above the floor (in the room without a sump) and the piping sloped downward to the sump cover to prevent water from pooling in the line. Therefore, the drain line is "dry" during normal operating conditions.

For the second room, the drain line connects the floor drain in the room to a sump in an adjoining area. The room without a sump is a watertight room, and any leak in the room would pass through the drain to the sump, where instrumentation would enable the leak to be detected. This drain is four (4) inches in diameter. One portion of the drain line is made from cast iron and the remainder, where the line enters the sump, is made from stainless steel. Both portions of the drain line are embedded in concrete. This line is routed from the floor drain to a point several feet above the bottom of the sump in the adjoining room, and sloped to prevent water pooling. Leakage in the room is normally insignificant, therefore this line is "dry" during normal operating conditions.

SNC has included these lines in the scope of the Rule, because they directly support the safety function of flooding detection. LRA Table 2.3.3.19 is unaffected since it already includes the applicable component types for these lines. Our evaluation did not identify any credible aging-effect that could potentially result in a failure of the drain piping (and surrounding concrete for one line) to direct flow to the sump. The Liquid Waste and Drains aging management review summary in Table 3.3.2-19 should have included the following:

Component Type *GALL Reference*	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Liquid Waste and Drains								
Piping (includes floor drain)	Pressure Boundary	Cast Iron	Air/Gas	None	None Required			J
			Embedded	None	None Required			J
		Stainless Steel	Air/Gas	None	None Required			J
			Embedded	None	None Required			J
			Inside	None	None Required			J

In the event of a fire, drains are available to serve as a support system to remove water used in fire suppression (UFSAR section 9B.4.1.21). However, the drains are a secondary support system to fire suppression. The waste disposal system was neither designed nor installed in accordance with the design specifications for the fire protection system. In scoping of SSCs for the regulated events of 10 CFR 54.4(a)(3) including fire protection (10CFR50.48), consideration of hypothetical failures or second-, third-, or fourth-level support systems is not required. This is consistent with the NRC's guidance on cascading for 10 CFR 54.4(a)(3) as described in Table 2.1-2 of NUREG-1800. Therefore, SNC has not included floor drains, equipment drains, or other waste disposal system components in scope for the regulated event of a fire under 10 CFR 54.4(a)(3).

In summary, SNC has included in the scope of license renewal the Liquid Waste and Drains features credited in the CLB for prevention/mitigation of internal flooding.

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.

Response

The non safety-related Auxiliary Steam and Condensate Recovery System is solely in scope for license renewal under 10 CFR 54.4(a)(2) for spatial interaction consideration with safety-related SSCs, including high energy piping considerations, as stated in LRA Section 2.3.4.5. The "strainers (shell)" component type listed in LRA Table 2.3.4.5 represents strainers, which are between the Auxiliary Steam Condensate Tank and the Auxiliary Steam Condensate pumps. The strainers have a potential spatial interaction with nearby safety related electrical components in Rooms 189 and 2189. License renewal boundary drawing D506447L sheet 1 identifies the spatial interaction concern in these rooms for this system.

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 structures and components (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).

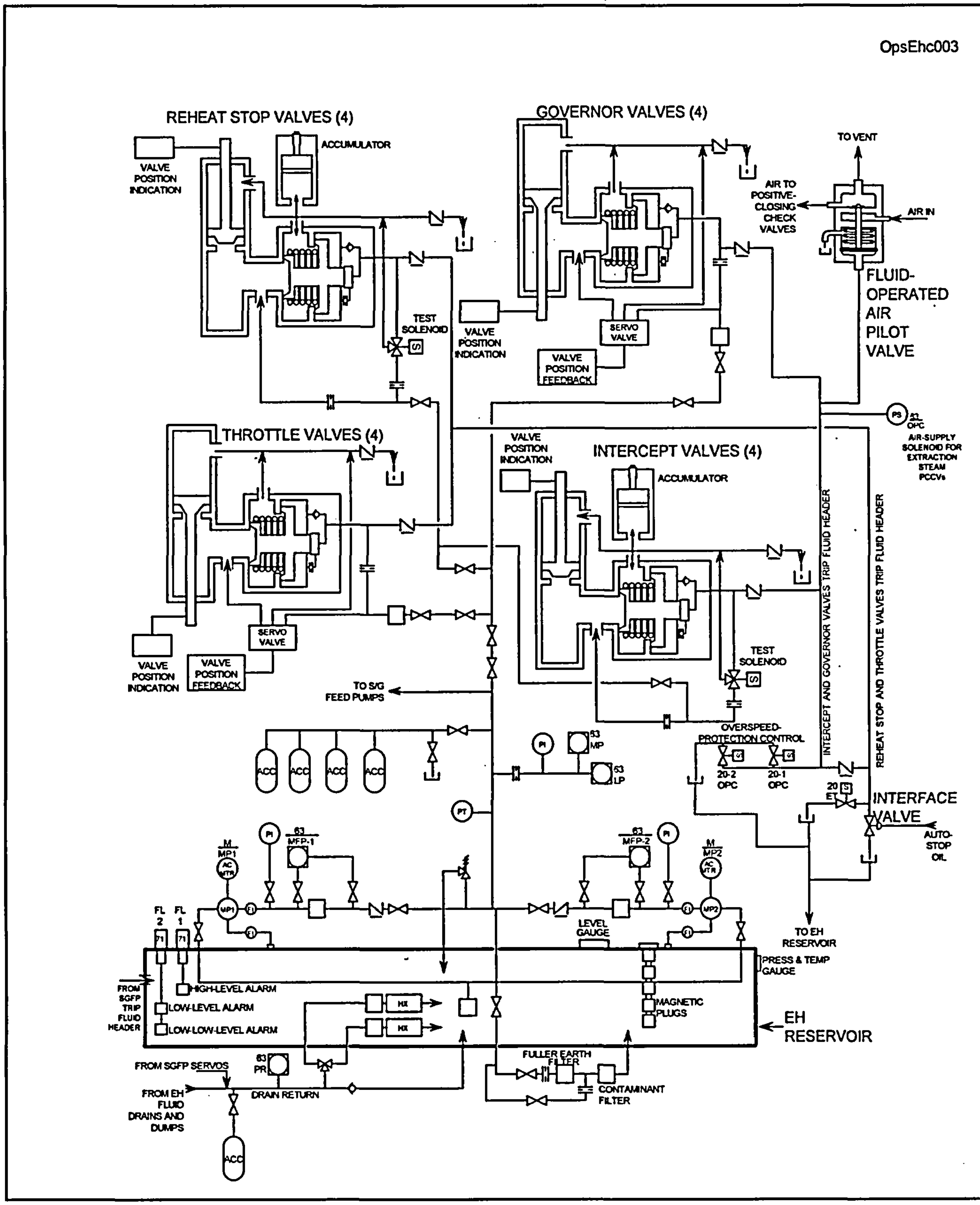
Response

A description of the turbine trip operation and diagrams from the Operations student lesson plans are provided to assist the staff in completing its review.

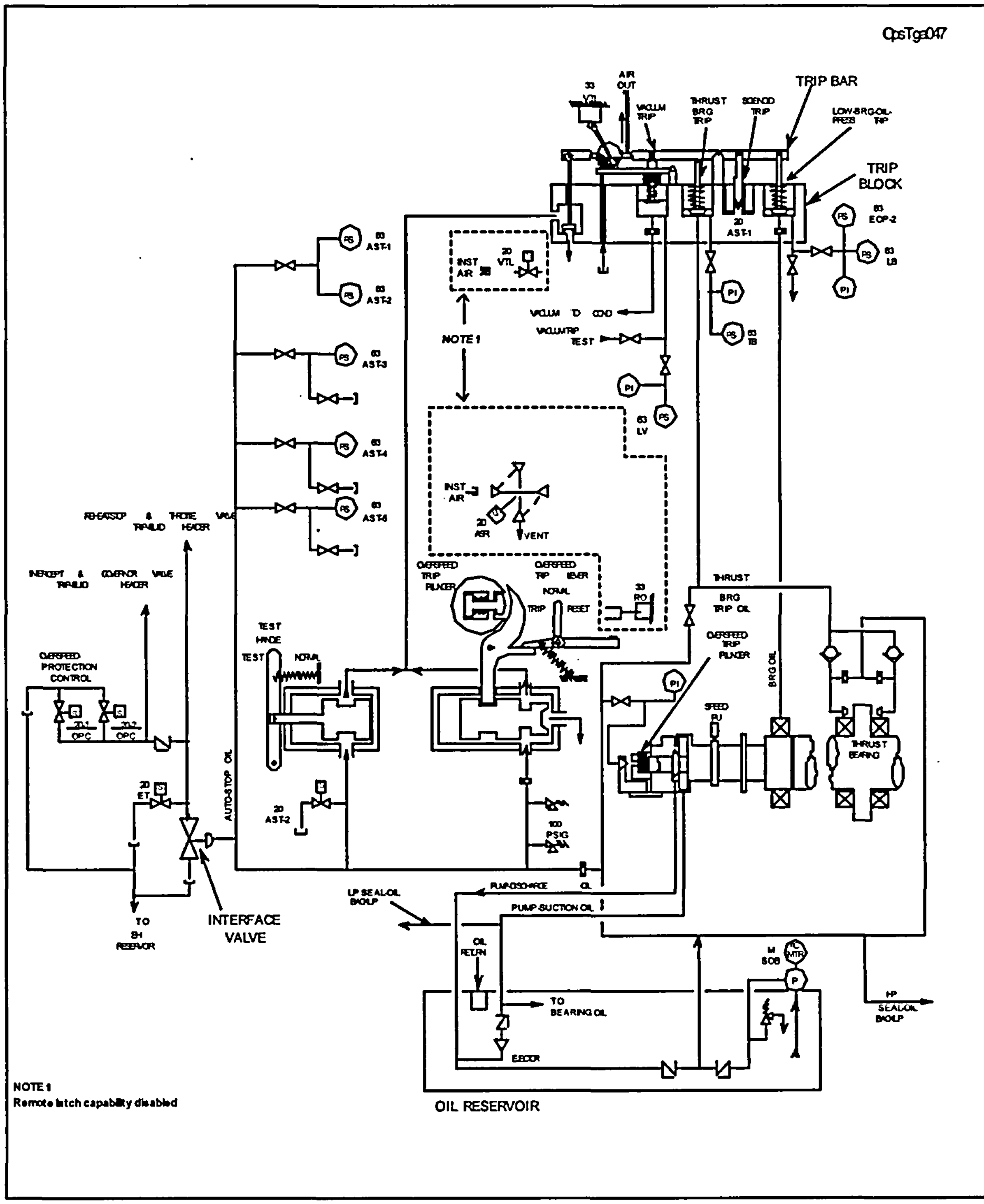
A turbine trip is initiated by dropping the fluid header pressure in either of the trip fluid headers (throttle & reheat stop valve trip fluid header and the intercept & governor valve trip fluid header) to a point where the hydraulic pressure is insufficient to overcome valve spring pressure thereby resulting in valve closure and termination of steam flow to the turbine. These trip fluid headers can function independently or together and are affected by the interface valve and by the 20/ET solenoid valve. Opening either the interface valve or 20/ET valve will drop fluid header pressure and close the throttle, reheat stop, intercept, and/or governor valves associated with the main turbine thus tripping the turbine. The turbine trip function is called upon in response to a normal turbine trip, reactor trip, turbine overspeed, and a turbine trip resulting from ATWS.

A review of the turbine controls design and component functions during the mechanical system screening process concluded that the trip functions are performed by active components, and that any passive component failure (loss of pressure boundary) would not prevent the performance of the system intended functions. On the contrary, loss of pressure boundary in the electrohydraulic (EH) Fluid or Auto Stop Oil subsystems would initiate a turbine trip, not prevent one. Therefore, the screening review concluded that the passive turbine controls components do not perform any intended functions for license renewal as described in 10 CFR 54.21(a)(1); therefore, none of the Turbine and Turbine Auxiliaries LRA system components are subject to an aging management review. Note that similar determinations by prior applicants (e.g., Robinson Nuclear Plant, Virgil C. Summer Nuclear Station) have been accepted by the staff.

EH Fluid System



Auto Stop Oil System



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 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.

Response

Alloy 82/182 weld locations are identified in FNP LRA Tables 3.1.2-1 and 3.1.2-3. The FNP RCS (ASME Class 1) piping Alloy 82/182 piping butt weld locations are associated with reactor pressure vessel (RPV) and pressurizer nozzle to safe-end welds. These locations are:

- RPV Primary inlet and outlet nozzle to safe-end welds (LRA Table 3.1.2-1);
- Pressurizer surge, safety (3 nozzles), relief, and spray nozzle to safe-end welds (LRA Table 3.1.2-3).

FNP Alloy 82/182 pipe butt welds are volumetrically examined in accordance with the FNP Risk-Informed Inservice Inspection Plan requirements for ASME Code Examination Category B-F welds. Volumetric examinations are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code as amended by 10 CFR 50.55a. Consistent with the requirements of Appendix VIII to ASME Section XI, examination techniques and personnel are qualified through performance demonstration on realistic mockups. Implementation of performance based requirements has resulted in development and application of improved procedures for volumetric detection and characterization of PWSCC in pipe butt welds.

Visual examinations of these weld locations for leakage continues to be performed in accordance with ASME Code Examination Category B-P.

In addition to these inspection requirements, the Materials Reliability Program (MRP) recommends that a direct visual inspection of the bare metal or an equivalent alternative inspection be performed once within the next two refueling outages. SNC is currently implementing this direct bare metal visual examination as follows: (1) The welds on the pressurizer and the reactor pressure vessel will be visually examined during the Unit 1 Fall 2004 outage; (2) The six welds on the pressurizer plus one of the RPV welds were visually examined during the Unit 2 Spring 2004 outage with no problems noted; and (3) The remaining five welds on the reactor pressure vessel will be visually examined during the Unit-2 Fall 2005 outage.

Long term inspection requirements will most likely change in response to new regulatory requirements from the NRC and industry recommendations from the MRP. SNC is an active participant in industry efforts to resolve issues associated with PWSCC of Alloy 82/182 pipe butt welds. After issuance and staff acceptance of a final Material Reliability Program Safety Assessment for Alloy 82/182 Pipe Butt Welds, SNC intends to implement the recommendations of this assessment at FNP.

SNC reviews both plant specific and industry-wide operating experience for applicability to FNP. Any applicable events would be evaluated for potential impact at FNP.

RAI 4.5.2-2

Section 4.5.2 of the LRA states that for the 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.

Response

A proprietary and non-proprietary version of WCAP-12825, Addendum 1, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plant for the License Renewal Program" was submitted to the staff for review in SNC letter NL-04-0654 dated April 22, 2004.

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 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.

Response

SNC clarifies that the FNP Reactor Vessel Internals Program scope includes all of the reactor internals components which perform an intended function as shown in LRA Table 3.1.2-2. The intent of the program is to manage these components through inspection of leading locations for the applicable aging effects.

In response to RAI B.5.1-3, SNC commits to implement industry recommendations regarding inspection locations (as applicable to FNP) and to submit an inspection plan for the FNP Reactor Vessel Internals Program for NRC review and approval at least 24 months prior to the periods of extended operation for the FNP units. Conforming changes to Section B.5.1 of Appendix B to the Farley LRA provided in the SNC response to RAI B.5.1-3 include clarification that the scope of the FNP Reactor Vessel Internals Program include those components shown in LRA Table 3.1.2-2.

RAI B.5.1-3

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 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.

Response

To address the staff's issues, SNC will revise the LRA FNP UFSAR Supplement A.2.13 to read as follows (***changes are in bold italics***):

A.2.13 REACTOR VESSEL INTERNALS PROGRAM

The new FNP Reactor Vessel Internals Program will be implemented prior to entering the period of extended operation to provide an integrated inspection program that addresses the reactor internals. It will be governed by administrative controls and procedures to supplement the inspection requirements of ASME Section XI, IWB Category B-N-3 to ensure that aging effects do not result in a loss of intended function of internal components during the period of extended operation.

The program will be used during the period of extended operation to manage the effects of crack initiation and growth due to irradiation assisted stress corrosion cracking; loss of fracture toughness due to irradiation embrittlement, thermal embrittlement, or void swelling; or changes in material properties (dimension) due to void swelling.

SNC will continue to participate in industry initiatives intended to clarify the nature and extent of aging mechanisms potentially affecting reactor vessel internals. SNC will incorporate the results of these initiatives (to the extent that they are applicable to the FNP reactor internals) into the scope, inspection requirements (inspection locations, methods, qualifications, and frequencies), acceptance criteria, and corrective actions of the Reactor Vessel Internals Program.

SNC will submit an inspection plan for the FNP Reactor Vessel Internals Program for NRC review and approval at least 24 months prior to entering the periods of extended operation for the FNP units.

The FNP Reactor Vessel Internals Program is consistent with the 10 attributes of the aging management program described in NUREG-1801, Sections XI.M13 and XI.M16, except as described above.

The "Farley Nuclear Plant – License Renewal Future Action Commitment" list will be revised accordingly.

LRA Appendix B.5.1, Reactor Vessel Internals Program, sections B.5.1.1 and B.5.1.3 should read as follows (deletions are indicated by strikethrough and additions are indicated in ***bold italics***):

LRA Appendix B.5.1 - Reactor Vessel Internals Program

B.5.1.1 Program Description

The new FNP Reactor Vessel Internals Program will be an integrated inspection program that addresses the reactor internals ***(as identified in LRA Table 3.1.2-2)***. It is intended to supplement the inspection requirements of ASME Section XI, IWB Category B-N-3 to ensure that aging effects do not result in a loss of intended function of internal components during the period of extended operation.

The Reactor Vessel Internals Inspection Program manages the effects of crack initiation and growth due to irradiation assisted stress corrosion cracking; loss of fracture toughness due to irradiation embrittlement, thermal embrittlement, or void swelling; or changes in material properties (dimension) due to void swelling.

~~While subject to change based on the results of ongoing industry research, accessible areas of the following Reactor Vessel Internals components will be included in the visual examination scope:~~

- ~~Baffle and Former Assemblies~~
- ~~Bottom Mounted Instrumentation Column Cruciforms (casting)~~
- ~~Core Barrel~~
- ~~Lower Core Plate and Fuel Alignment Pins~~
- ~~Lower Support Forging~~
- ~~Upper Support Column Bases (casting)~~

SNC supports development of improved industry data, models, and inspection methodologies through active participation in the EPRI Materials Reliability Program Reactor Vessel Internals Issue Task Group and the Westinghouse Owners Group. ***SNC will continue to participate in industry initiatives intended to clarify the nature and extent of aging mechanisms potentially affecting reactor vessel internals. SNC will incorporate the results of these initiatives (to the extent that***

they are applicable to the FNP reactor internals) into the scope, inspection requirements (inspection locations, methods, qualifications, and frequencies), acceptance criteria, and corrective actions of the Reactor Vessel Internals Program.

B.5.1.3 Exceptions to NUREG-1801

The scope, inspection requirements (inspection locations, methods, qualifications, and frequencies), acceptance criteria, and corrective actions will be based on the results of industry initiatives intended to clarify the nature and extent of aging mechanisms potentially affecting the reactor vessel internals. SNC will submit an inspection plan for the FNP Reactor Vessel Internals Program for NRC review and approval at least 24 months prior to entering the periods of extended operation for the FNP units.

~~Based on the recent replacement of selected FNP baffle bolting with an improved bolt design and a lack of data regarding the effectiveness of ultrasonic examinations, FNP takes exception to the requirement for ultrasonic examination of baffle bolting. FNP will perform VT-1 of these connections. Additionally, SNC will continue to participate in industry activities coordinated by the WOG and MRP and will update this inspection program as appropriate based on the results of future research initiatives.~~

~~FNP will limit VT-1 and enhanced VT-1 inspections to those leading locations as determined by industry research and operating experience.~~

~~The new inspection program will not follow the inspection cycles set forth in ASME Section XI IWB. A baseline inspection of the FNP internals will be performed during the 5th ISI inspection interval. The frequency of subsequent inspections, and the inspection methodologies utilized, will be based on the results of the baseline inspections.~~

~~The FNP Reactor Vessels Internals Program may contain additional inspection requirements and acceptance criteria to detect changes in critical reactor internals dimensions due to void swelling. These inspection requirements and acceptance criteria will be based upon the results of ongoing industry research regarding the significance of void swelling in the PWR environment.~~

RAI B.5.1-4

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.

Response

SNC does not take exception to the number of inspection cycles set forth in Section XI of the ASME Boiler and Pressure Vessel Code, Subsection IWB, for inspections of reactor internals components required under the FNP ISI Program. Inspection frequencies are specific to the program as approved by the NRC (see SNC's response to RAI B.5.1-3 wherein SNC commits to submit the program inspection plan for staff review and approval).

The statement that is the subject of concern in RAI B.5.1-4 was deleted by conforming changes to FNP LRA Appendix B, Section B.5.1, provided in SNC's response to RAI B.5.1-3.