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SOUTHERN COMPANY
Energy to Serve Your World℠

January 30, 2004

Docket Nos.: 50-348
50-364

NL-04-0115

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

Joseph M. Farley Nuclear Plant
Supplemental Information for
Relaxation Request to Order EA-03-009

Ladies and Gentlemen:

NRC Order EA-03-009, issued February 11, 2003, established interim inspection requirements for reactor pressure vessel (RPV) heads at pressurized water reactors. On December 3, 2003, Southern Nuclear Operating Company (SNC) submitted a request for a one-time relaxation of the Order, to extend by one refueling outage the required interval for performance of non-visual non-destructive examination (NDE) of the Farley Nuclear Plant (FNP) Unit 2 reactor pressure vessel (RPV) head penetration nozzles per Order item IV.C.(1)(b).

SNC representatives met with NRC staff members on January 16, 2004 to present and discuss this relaxation request. Topics, which were of particular interest to the NRC staff during that meeting, for which SNC agreed to supply additional information include 1) industry experience with the material heats used in manufacturing the penetration nozzles for the FNP RPV heads, 2) analysis of crack propagation in the nozzles above the attachment welds, and 3) the beneficial effects of zinc injection into the reactor coolant system (RCS) in mitigating the initiation of primary water stress corrosion cracking (PWSCC) in Alloy 600, the material used in the FNP RPV head nozzles.

Further information on these topics is included in Enclosure 1. Additionally, in Enclosure 2 SNC provides a copy of the Electric Power Research Institute (EPRI) Materials Reliability Program report, "Effect of Zinc Addition on Mitigation of Primary Water Stress Corrosion Cracking of Alloy 600 (MRP-78)," EPRI Report 1003522, October 2002.

Because MRP-78 contains information proprietary to EPRI, a copy of a letter and affidavit are provided in Enclosure 3, both dated January 26, 2004 and signed by an EPRI representative. These documents set forth the basis on which the information may be withheld from public disclosure by the Commission per the considerations listed in 10 CFR 2.790 (b)(4). Accordingly, it is respectfully requested that the information which is proprietary to EPRI be withheld from public disclosure in accordance with 10 CFR 2.790 (a)(4) of the Commission's regulations. Questions regarding this request for withholding

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and the contents of MRP-78 should be directed as indicated in the EPRI letter. A copy of a non-proprietary version of MRP-78 is provided in Enclosure 4.

SNC requests approval of the subject relaxation by March 13, 2004 which is the currently scheduled date for beginning the next Farley Unit 2 refueling outage.

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.

This letter contains no new NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L.M. Stinson

L. M. Stinson

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

Valerie A. O'Brien
Notary Public

My commission expires: 4/28/07

LMS/DWD/sdl

Enclosures:

1. Supplemental Information for Relaxation Request to Order EA-03-009
2. EPRI Materials Reliability Program Report, "Effect of Zinc Addition on Mitigation of Primary Water Stress Corrosion Cracking of Alloy 600 (MRP-78)," EPRI Report 1003522, October 2002 (Proprietary Version)
3. January 26, 2004 EPRI Letter with Affidavit regarding MRP-78
4. EPRI Materials Reliability Program Report, "Effect of Zinc Addition on Mitigation of Primary Water Stress Corrosion Cracking of Alloy 600 (MRP-78)," EPRI Report 1003522, October 2002 (Non-Proprietary Version)

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# 13940

U. S. Nuclear Regulatory Commission
Mr. L. A. Reyes, Regional Administrator
Mr. S. E. Peters, NRR Project Manager – Farley
Mr. C. A. Patterson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Enclosure 3

Joseph M. Farley Nuclear Plant

January 26, 2004 EPRI Letter
with Affidavit regarding MRP-78

Enclosure 1

Joseph M. Farley Nuclear Plant

**Supplemental Information for
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1. **Industry experience with the material heats used in manufacturing the penetration nozzles for the Farley Nuclear Plant (FNP) Reactor Pressure Vessel (RPV) heads (based on U.S. plants only)**

FNP Unit 1
No flaws were detected in any FNP Unit 1 penetration nozzle by the ultrasonic testing (UT) examination performed during the last refueling outage, in spring 2003. The Alloy 600 heats used for the 4-inch nozzles were Heat C2689 from B&W Tubular Products and Heat NX9420 from Huntington Alloys. Heat C2689 was used at only one other unit, with no cracking reported. Heat NX9420 was used at three other units, also with no cracking reported. Huntington Alloys Heat NX0706 was used for the 1-inch diameter head vent nozzle. This heat was also used at five other units. No U.S. plant has reported cracking in a head vent nozzle.

FNP Unit 2
No flaws were detected in any FNP Unit 2 nozzle by the UT examination performed during the last refueling outage, in fall 2002. The Alloy 600 heats used for the 4-inch diameter penetration nozzles were Heat M3935 from B&W Tubular Products and Heat NX3249 from Huntington Alloys. Heat NX3249 was used at only one other unit, with no cracking reported. Heat M3935 (used for 61 of the 69 4-inch nozzles) was used at four other units and all reported cracking. Huntington Alloys Heat NX2142 was used for the 1-inch diameter head vent nozzle. This heat was used at just one other unit. No U.S. plant has reported cracking in a head vent nozzle.

2. **Analysis of crack propagation in the nozzles above the attachment welds**

SNC submitted a letter to NRC on April 11, 2003 which included proprietary and non-proprietary versions of Westinghouse topical report WCAP-15925, "Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operation: Farley Units 1 and 2." Figures 6-5 through 6-7 in WCAP-15925 show crack growth predictions for axial flaws on the inside diameter (ID) of the nozzles near and above the J-groove attachment welds (pressurized water stress corrosion cracking (PWSCC) flaws in the nozzles are considered most likely to initiate axially). In all cases more than two operating cycles would be required for an ID flaw with an initial depth of 5% of the nominal 5/8 inch nozzle wall thickness to propagate through-wall. Demonstration testing was conducted by the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) of the UT equipment and techniques used during the fall 2002 examination of the FNP Unit 2 nozzles. This testing showed that ID flaws were reliably detected down to 5% of wall thickness; therefore, it is unlikely the fall 2002 examination missed any flaws which could propagate through-wall and cause leakage prior to the planned replacement of the FNP Unit 2 RPV head in fall 2005.

3. Beneficial effects of zinc injection into the Reactor Coolant System (RCS) in mitigating the initiation of PWSCC in Alloy 600

Enclosures 2 and 4 provide copies of the proprietary and non-proprietary versions of the EPRI Materials Reliability Program report, "Effect of Zinc Addition on Mitigation of Primary Water Stress Corrosion Cracking of Alloy 600 (MRP-78)," EPRI Report 1003522, October 2002. Pertinent points from MRP-78 are extracted and reproduced below (no proprietary material is included).

From the "Product Description" section:

> "EPRI and Southern Nuclear cosponsored an initial field demonstration of zinc addition at Farley Unit 2 in 1994-95. The results of that demonstration and other laboratory studies have been increasingly complemented by experience in operating pressurized water reactors (PWRs). These studies confirm the beneficial effects of zinc in mitigating radiation fields, with positive results observed in both domestic and German PWRs."

(Note - Zinc injection has been continued at both FNP units subsequent to the initial field demonstration; FNP Unit 2 has accrued approximately five and one-half years of operating time with zinc in the RCS to date.)

> "Essentially without exception, the results of laboratory testing indicate the beneficial effect of zinc addition in mitigating PWSCC initiation in alloy 600, with this benefit proportional to the zinc concentration. However, data for a beneficial effect on crack propagation are less certain, requiring additional laboratory work in carefully selected environments to resolve this issue."

> "Based on laboratory data, zinc injection appears to be a viable chemistry approach to mitigating PWSCC initiation and potentially propagation. Field data are consistent with a zinc benefit, but changing steam generator inspection practices make it difficult to ascertain a direct link. Nonetheless, reassessment of laboratory and field data support continued use and evaluation of zinc addition as a means to mitigate PWSCC."

> "This project incorporated additional laboratory and plant experience since February 2001 to reassess the effect of zinc addition on PWSCC of alloy 600. Laboratory data indicate a significant benefit of zinc addition with respect to crack initiation, with less certain results for crack propagation."

Enclosure 1

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From the "Abstract":

"A brief review of the mechanism by which zinc interacts with corrosion films on nickel-base alloys is followed by a review of the available literature dealing with zinc and PWSCC... The results of this review lead to the following conclusions:

- The mechanism by which zinc affects the corrosion of austenitic nickel-base alloys is by incorporation of zinc into the spinel oxide corrosion films.
- Reduction of general corrosion leads to reduced metal release rates and an associated dose rate reduction in operating steam generators by modifying the corrosion source term.
- Nearly without exception, the results of laboratory testing indicate a benefit of zinc injection in mitigating the initiation of PWSCC in Alloy 600. Early laboratory data suggest this benefit may vary with the concentration of zinc in the RCS.
- Data for a beneficial effect on crack propagation are mixed. The laboratory data vary from a substantial reduction in crack growth rates to no effect. Interpretations of these differences based on the nature of the crack tip oxides do not agree.
- The only substantial operating plant data are those from Diablo Canyon Units 1 and 2, where zinc has been injected for portions of three fuel cycles in each plant. Significant reductions in the initiation and propagation of cracks at TTS and tube-TSP locations have been observed in outages since zinc injection was adopted. Attributing these effects solely to the addition of zinc is complicated by the fact that concurrent changes have occurred in eddy current inspection equipment and scope, and in the plugging criteria, over this same period."

From Section 1, "Introduction":

"It is essentially universally accepted that zinc is very effective in mitigating the extent and consequences of general corrosion of both austenitic stainless steels and nickel-base alloys in primary water. This appears to be the result of substantial restructuring of the corrosion films that form on these materials in high temperature primary coolant."

"What is less clear, however, is the effectiveness of zinc in mitigating the occurrence of PWSCC. The results of the laboratory testing database developed to support the initial application of zinc to PWRs showed a clear benefit with respect to crack initiation in highly strained reverse U-bends, and also suggested a reduction in crack propagation rates (Ref. 1.11). More recent published research appears to support the crack initiation benefit, but is mixed with regard to crack propagation, with some data suggesting a benefit and other data indicating zinc has no effect on crack propagation rates."

Enclosure 1

Joseph M. Farley Nuclear Plant

**Supplemental Information for
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From Section 2, "Interaction of Zinc with Corrosion Films on Nickel-Base Alloys":

> "Examination by Auger electron spectroscopy of the surfaces of Alloy 600 and stainless steel that has been exposed to simulated primary water containing zinc clearly indicates efficient incorporation of zinc into the oxide corrosion films."
>
> "Auger examination of a tube pulled from Farley Unit 2 at the end of Cycle 10 (Ref. 2.8), following approximately nine months of zinc addition to the primary coolant at a concentration of approximately 40 ppb, gave results very similar to those seen in laboratory research."
>
> "Detailed knowledge of the precise mechanism notwithstanding, it is clear that the field experience is consistent with expectations based on laboratory tests and rational interpretation of the phenomena."