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SOUTHERN COMPANY
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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Joseph M. Farley Nuclear Plant
Response to Generic Letter 97-01

Ladies and Gentlemen:

Generic Letter (GL) 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations, was issued to request licensees to describe their program for insuring the timely inspection of PWR control rod drive mechanism (CRDM) and other closure head penetrations. Enclosed is the Southern Nuclear Operating Company (SNC) response for Farley Nuclear Plant (FNP) relative to the information requested by GL 97-01.

Prior to issuance of GL 97-01, SNC worked with the Westinghouse Owners Group (WOG), the Electric Power Research Institute (EPRI), and the Nuclear Energy Institute (NEI) to understand the operational experience, identify technical issues, cause factors, relative importance, and solutions to the Alloy 600 issues identified by the NRC. One of these tasks was the development of safety evaluations by each of the PWR Owners Groups (PWROGs) that characterized the initiation of damage, propagation, and consequences associated with Alloy 600 head penetration cracking. The WOG safety evaluation is contained in WCAP-13565, Alloy 600 Reactor Vessel Head Adaptor Tube Cracking Safety Evaluation, and is applicable to FNP. The NRC reviewed the safety evaluations prepared by the PWROGs and issued a safety evaluation report (SER) to the Nuclear Management and Resources Council (NUMARC) on November 19, 1993. The WOG safety evaluation and the resulting NRC SER establish the basis for the continued operation of FNP.

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Mr. D. N. Morey states he is vice president of SNC, is authorized to execute this oath on behalf of SNC, and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

DN Morey

Dave Morey

Sworn to and subscribed before me this 25th day of July 1997

Martha Gayle Dow
Notary Public

My Commission Expires: November 1, 1997

REM/clt:GL9701

Enclosure

cc: Mr. L. A. Reyes, Region II Administrator
Mr. J. I. Zimmerman, NRR Project Manager
Mr. T. M. Ross, Plant Sr. Resident Inspector
Mr. T. A. Reed, NRR - Materials and Chemical Engineering Branch
Dr. D. E. Williamson, State Department of Public Health

ENCLOSURE

JOSEPH M. FARLEY NUCLEAR PLANT
RESPONSE TO GENERIC LETTER 97-01

Response to Generic Letter 97-01

GL 97-01 Requested Information Item 1.1

A description of all inspections of CRDM nozzle and other VHPs (vessel head penetrations) *performed to the date of this generic letter, including the results of these inspections.*

SNC Response to Requested Information Item 1.1

Consistent with WCAP-13565, Alloy 600 Reactor Vessel Head Adaptor Tube Cracking Safety Evaluation, and the resulting NRC SER dated November 19, 1993, Southern Nuclear Operating Company (SNC) performs GL 88-05 walkdowns following shutdown for refueling of either of the Farley Nuclear Plant (FNP) units to identify boric acid deposits that could be indicative of reactor coolant system (RCS) leakage. These walkdowns include a visual inspection of the reactor vessel head and to date, no significant boric acid deposits have been identified that indicates leakage from the subject reactor vessel head penetration nozzles. In addition to the GL 88-05 walkdowns, SNC inspections of 29 out of 69 head penetrations on Unit 1 and 32 out of 69 head penetrations on Unit 2 between the reactor vessel head insulation and the top of the reactor head using a remote camera did not identify any evidence of leakage from the penetrations. These inspections were performed in 1995 and included penetrations for CRDMs, instrument columns, head adapter plugs, and latch housings. Also in 1995, field replication of one of the material heats used in Unit 1 and two of the material heats used in Unit 2 were performed to determine the material microstructure, a factor in the models used to assess the susceptibility of the reactor vessel head penetrations to primary water stress corrosion cracking (PWSCC). The penetrations selected for replication were chosen based on their location in the periphery of the closure head in the area of the highest residual stress.

GL 97-01 Requested Information Item 1.2

If a plan has been developed to periodically inspect the CRDM nozzle and other VHPs:

A. *Provide the schedule for first, and subsequent, inspections of the CRDM nozzle and other VHPs, including the technical basis for this schedule.*

B. *Provide the scope for the CRDM nozzle and other VHP inspections, including the total number of penetrations (and how many will be inspected), which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations.*

SNC Response to Requested Information Item 1.2

SNC had originally scheduled volumetric examinations for FNP 1 and 2 in the year 2001. WCAP-13565, Alloy 600 Reactor Vessel Head Adaptor Tube Cracking Safety Evaluation, and the resulting NRC SER provided the basis for the original SNC plan. Specifically, WCAP-13565 and the NRC SER concluded:

(1) axial flaws would be more likely than circumferential flaws;
(2) catastrophic failure of a penetration is extremely unlikely because a flaw would be detected during GL 88-05 boric acid leakage surveillance walkdowns before it reached the critical flaw size;

(3) the Westinghouse Owners Group (WOG) susceptibility model considers the appropriate parameters affecting IGSCC and should provide a reasonable ranking of plant susceptibilities;
(4) based on existing leakage monitoring requirements, there is reasonable assurance that leakage in excess of the 1.0 gpm technical specification limit would be detected prior to any unstable extension of the flaw; and,
(5) CRDM cracking at the reactor vessel heads is not a significant safety issue as long as the surveillance walkdowns in accordance with GL 88-05 continue.

In response to GL 97-01, SNC has elected to participate in the Nuclear Energy Institute (NEI)/WOG RPV head penetration integrated inspection program described in the SNC Response to Requested Information Item 1.4 below. As a participant in the NEI/WOG integrated inspection program, inspections of the FNP reactor vessel head penetration nozzles will be based on the NEI/WOG integrated inspection program.

GL 97-01 Requested Information Item 1.3

If a plan has not been developed to periodically inspect the CRDM nozzle and other VHPs, provide the analysis that supports why no augmented inspection is necessary.

SNC Response to Requested Information Item 1.3

Based on SNC participation in the NEI/WOG integrated inspection program, Requested Information Item 1.3 is not applicable to FNP.

GL 97-01 Requested Information Item 1.4

In light of the degradation of CRDM nozzle and other VHPs described above, provide the analysis that supports the selected course of action as listed in either 1.2 or 1.3, above. In particular, provide a description of all relevant data and/or tests used to develop crack initiation and crack growth models, the methods and data used to validate these models, the plant-specific inputs to these models, and how these models substantiate the susceptibility evaluation. Also, if an integrated industry inspection program is being relied on, provide a detailed description of this program.

SNC Response to Requested Information Item 1.4

The data, tests, and methods used in developing the crack initiation and crack growth models on which the SNC management strategy for addressing the FNP vessel head penetration cracking issue is based are provided in Sections 2 and 3 of WCAP-14901, Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group.

SNC is a participant in the WOG analysis program. A plant specific probability analysis using the methodology described in Section 4 of WCAP-14901 has been performed for FNP. The plant specific input parameters to the analysis are provided in Table 1 and Table 2 of this response for Unit 1 and Unit 2, respectively. The analysis results will be incorporated into the NEI/WOG integrated inspection program, for use in determining the need for a plant specific inspection. This integrated inspection program includes all three PWR owners groups, the Electric Power Research Institute (EPRI), and NEI who are cooperatively working to compile information on the estimated operating time from January 1, 1997, needed to initiate and propagate a crack 75% through wall in a vessel penetration for all operating PWRs in the United States. This information will be evaluated to determine if an adequate number of plants have or are planning to inspect in the near future. This evaluation is scheduled to be completed and detailed inspection plans for the industry to be provided to the NRC by the end of 1997. A copy of WCAP-14901 has been provided to the NRC by the WOG.

TABLE 1

FARLEY UNIT 1
INPUT VALUES USED FOR PROBABILISTIC ANALYSIS

Case	Penetration No.	Temperature	Setup Angle (°)	Y.S. (ksi)	Grain Boundary Coverage (%)
1	62-69	596.5 °F	42.6	32.7	50.2 [1]
2	58-61		40.0	32.7	50.2 [1]
3	50-57		38.6	32.7	50.2 [1]
4	46-49		37.3	32.7	50.2 [1]
5	38-45		33.1	32.7	50.2 [1]
6	30-37		28.6	32.7	50.2 [1]
7	26-29		27.0	32.7	50.2 [1]
8	22-25		25.4	32.7	50.2 [1]
9	18-21		19.8	39.0	44.7
10	14-17		19.8	32.7	50.2 [1]
11	10-13		17.6	32.7	50.2 [1]
12	6-9		12.4	32.7	50.2 [1]
13 [2]	2-5		8.7	32.7	50.2 [1]

[1] Values from field replication

[2] This case also used to bound penetration number 1

TABLE 2

FARLEY UNIT 2
INPUT VALUES USED FOR PROBABILISTIC ANALYSIS

Case	Penetration No.	Temperature	Setup Angle (°)	Y.S. (ksi)	Grain Boundary Coverage[1] (%)
1	62-69	596.9 °F	42.6	48.5	23.0
2	58-61		40.0	48.5	23.0
3	50-57		38.6	35.0	27.7
4	46-49		37.3	48.5	23.0
5	38-45		33.1	48.5	23.0
6	30-37		28.6	48.5	23.0
7	26-29		27.0	48.5	23.0
8	22-25		25.4	48.5	23.0
9	14-21		19.8	48.5	23.0
10	10-13		17.6	48.5	23.0
11	6-9		12.4	48.5	23.0
12	2-5		8.7	48.5	23.0
13	1		0	48.5	23.0

[1] Values from field replication

GL 97-01 Requested Information Item 2

Provide a description of any resin bead intrusions, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines recommendations for primary water sulfate levels, including the following information: (See Requested Information Items 2.1 through 2.6)

SNC Response to Requested Information Item 2

SNC has reviewed the plant historical records and determined that a resin ingress similar to those which occurred in 1980 and 1981 at the Jose Cabrera (Zorita) plant has not occurred at FNP. This data search was structured to identify any resin intrusion event into the primary coolant system that exceeded a magnitude of 1 cubic foot (approximately 30 liters). The threshold of 1 cubic foot was chosen as a conservative lower bound since it represents less than 15% of the estimated volume of resin released into the reactor coolant system during the two events at Jose Cabrera.

Routine analysis for sulfate in reactor coolant was not performed prior to June 1994 for FNP. For the period of plant operation following June 1994, a sulfate concentration in the range of 15 to 17 ppm was used as an indicator of action resin ingress equivalent to a volume of 1 cubic foot.

For the period of plant operation prior to the routine analysis for sulfate in reactor coolant, the data search was based on a review of the plant's reactor coolant chemistry records relative to conductivity of the reactor coolant. An elevation of 28 μS/cm in conductivity was the value used as an indicator of action resin ingress equivalent to a volume of 1 cubic foot.

One occurrence was identified where the RCS conductivity increase exceeded the 28 μS/cm criterion for approximately a two week period in February 1988. A review of additional RCS chemistry parameters during this period determined that the elevated conductivity was not the result of a resin bead intrusion. Specifically, the elevated conductivity was not accompanied by a corresponding depression of pH, elevation in lithium, or elevation in suspended solids.

Requested Information Item 2 implied that any increase above the limits specified in the EPRI PWR Primary Water Chemistry Guidelines were indicative of a potential resin bead intrusion. It is true that a resin bead intrusion would result in sulfate values that exceed the EPRI Primary Water Chemistry Guidelines, but it is not true that all sulfate measurements in excess of the EPRI guidelines are indicative of a resin intrusion. Therefore, SNC used the above criteria to determine that a resin bead intrusion has not occurred at FNP.

GL 97-01 Requested Information Item 2.1

Were the intrusions action, anion, or mixed bed?

SNC Response to Requested Information Item 2.1

Based on the determination that no resin bead intrusions have occurred at FNP, Requested Information Item 2.1 is not applicable to FNP.

GL 97-01 Requested Information Item 2.2

What were the duration's of these intrusions?

SNC Response to Requested Information Item 2.2

Based on the determination that no resin bead intrusions have occurred at FNP, Requested Information Item 2.2 is not applicable to FNP.

GL 97-01 Requested Information Item 2.3

Does the plant's RCS water chemistry Technical Specifications follow the EPRI guidelines?

SNC Response to Requested Information Item 2.3

The FNP Technical Specifications provide limits on RCS dissolved oxygen, chlorides, and fluorides, and these limits are consistent with Action Levels II and III of the EPRI Primary Water Chemistry Guidelines.

GL 97-01 Requested Information Item 2.4

Identify any RCS chemistry excursions that exceed the plant administrative limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium.

SNC Response to Requested Information Item 2.4

Based on the determination that no resin bead intrusions have occurred at FNP, Requested Information Item 2.4 is not applicable to FNP.

GL 97-01 Requested Information Item 2.5

Identify any conductivity excursions which may be indicative of resin intrusions. Provide a technical assessment of each excursion and any follow-up actions.

SNC Response to Requested Information Item 2.5

See SNC response to Requested Information Item 2 regarding review of RCS conductivity data.

GL 97-01 Requested Information Item 2.6

Provide an assessment of the potential for any of these intrusions to result in a significant increase in the probability for IAA of VHPs and any associated plan for inspections.

SNC Response to Requested Information Item 2.6

Based on the determination that no resin bead intrusions have occurred at FNP, Requested Information Item 2.6 is not applicable to FNP.