J. Barnie Beasley, Jr., P.E.
Vice President

Southern Nuclear Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7110
Fax 205.992.0341

SOUTHERN COMPANY
Energy to Serve Your World℠

October 9, 2003

Docket Nos.: 50-348
50-364

NL-03-2098

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

Joseph M. Farley Nuclear Plant
Application for License Renewal

Ladies and Gentlemen:

Southern Nuclear Operating Company (SNC) submitted the license renewal application for Joseph M. Farley Nuclear Plant, Units 1 and 2, by SNC Letter No. NL-03-1657 dated September 12, 2003. In recent telephone conversations, the NRC Staff requested that SNC expand on the Time Limited Aging Analysis (TLAA) information in Section 4.5 of the application. By this letter, SNC is providing that requested information.

Should you provide additional questions, please contact Charles Pierce at telephone number 205-992-7872.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

J. B. Beasley, Jr.

Sworn to and subscribed before me this 9th *day of* October, *2003.*

Notary Public

My commission expires: 11/10/06

JBB/JAM/sdl

Enclosure: LRA Section 4.5 Additional Information

A099

cc: Southern Nuclear Operating Company
Mr. J. D. Woodard, Executive Vice President
Mr. D. E. Grissette, General Manager – Plant Farley (w/ 1 enclosure)
Document Services RTYPE: CFA04.054; LC# 13849 (w/ 1 enclosure)

U. S. Nuclear Regulatory Commission
Mr. R. William Borchardt, Acting Director – NRR (w/ 1 enclosure)
Mr. L. A. Reyes, Regional Administrator (w/ 1 enclosure)
Mr. F. Rinaldi, NRR Project Manager – Farley (w/ 1 enclosure)
Mr. T. P. Johnson, Senior Resident Inspector – Farley (w/ 1 enclosure)

Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer (w/ 1 enclosure)

Local Libraries
Houston Love Memorial Library (w/ 1 enclosure)

Mayors and County Agents
Mr. B. Alloway, Mayor, City of Ashford (w/ 1 enclosure)
Mr. O. Smith, Mayor, Town of Gordon (w/ 1 enclosure)
Mr. J. N. Green, Mayor, Town of Columbia (w/ 1 enclosure)
Mr. A. Howard, Chairman, Early County Commission (w/ 1 enclosure)
Mr. L. Turner, Chairman, Henry County Commission (w/ 1 enclosure)
Mr. M. Culver, Chairman, Houston County Commission (w/ 1 enclosure)

ENCLOSURE

LRA SECTION 4.5 ADDITIONAL INFORMATION

4.5.1 ULTIMATE HEAT SINK SILTING

The ultimate heat sink (UHS) for Farley Nuclear Plant (FNP) is a pond in which silt deposition (silting) may occur. Excessive silting could adversely impact the capability of the Service Water System to perform its safety function to achieve safe shutdown and maintain long term cooling of the plant following a design basis accident.

As described in Section 2.4.8.1 of the FNP Updated Final Safety Analysis Report (UFSAR), SNC conducts routine soundings to confirm the water volume in the pond. These soundings include a detailed and extensive sounding of the UHS once every five years. The soundings are evaluated for acceptability in part to confirm the Basis for Technical Specification 3.7.9. Under this technical specification, the pond elevation is checked every 24 hours to ensure that the surface elevation of the pond is at least 184 feet, above sea level. The soundings confirm that, for this pond elevation, enough pond water volume exists to safely shutdown and maintain long term cooling.

Part of the acceptability criteria for the evaluation of the soundings is a curve (see attached Figure 4.5.1) showing the acceptable surface area of the pond versus the pond level height (depth) based on the requirements of the UHS. This establishes a minimum acceptable volume (with margin) of 1325 acre-feet at a surface elevation of 184 feet. SNC has validated this curve by trending the rate of change of the volume of the pond using data from prior surveillance soundings and extrapolating out to 40 years. As might be expected if silting is occurring, the calculation currently shows a reduction in water volume trend over time, and the curve for 40 years shows less volume available than is currently available in the UHS pond for a given pond level height. The data used to originally generate the curve for the current term covered the period from 1981 through to 1989.

For license renewal, SNC extrapolated to 60 years the calculation that validates the acceptability curve to determine if an update was necessary. SNC elected to treat this calculation as a time-limited aging analysis (TLAA) for renewal.

To extrapolate the calculation for the period of extended operation, SNC incorporated additional surveillance data from soundings performed in 1993 and 1998 into the calculation. An additional surveillance was performed in August 2003, but this data was not available at the time SNC prepared the License Renewal Application (LRA) materials for submittal. The calculation uses the method of least squares to fit a curve through the data. With the 1993 and 1998 data included, the existing required pond volume remains conservative for the

renewal term and adequately assures enough pond water volume exists to safely shutdown and maintain long term cooling.

Through this review and the revision of the analysis, SNC has demonstrated that the TLAA is acceptable, as modified, for the renewed license term in accordance with 10 CFR 54.21(c)(1)(ii).

FIGURE 4.5.1

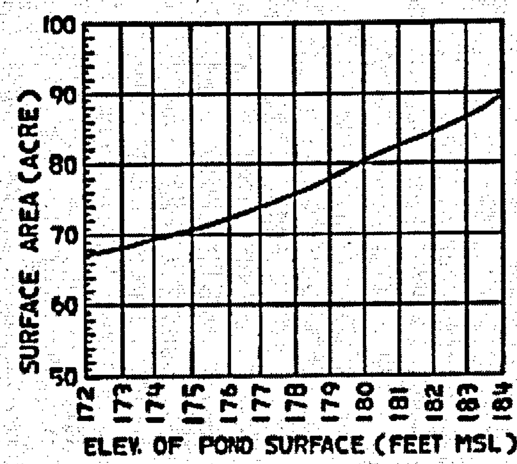
THE FOLLOWING REQUIREMENTS CONCERNING THE VOLUME OF WATER IN THE SERVICE WATER STORAGE POND AND THE SURFACE AREA OF THE SERVICE WATER STORAGE POND ARE BASED ON ASSUMPTIONS MADE IN THE ULTIMATE HEAT SINK EVALUATION:

1. THE VOLUME OF WATER IN THE SERVICE WATER POND, BASED ON A POND ELEVATION OF EL. 184'-0" MSL, MUST NOT BE LESS THAN 1,325 ACRE-FEET.

2. THE SURFACE AREA OF THE POND MUST NOT BE LESS THAN THE AREA SHOWN ON THE GRAPH AT RIGHT FOR ALL POND ELEVATIONS BETWEEN EL. 172'-0" MSL AND EL. 184'-0" MSL.

THE DATA POINTS USED TO PREPARE THIS GRAPH ARE PRESENTED BELOW:

POND ELEVATION (FT. MSL)	SURFACE AREA (ACRES)
172'-0"	67.1
173'-0"	68.3
174'-0"	69.5
175'-0"	70.9
176'-0"	72.3
177'-0"	73.8
178'-0"	75.7
179'-0"	77.8
180'-0"	80.1
181'-0"	82.2
182'-0"	84.1
183'-0"	86.2
184'-0"	89.1



REFERENCES:

1. ENGINEERING STUDY ES 89-1500, ULTIMATE HEAT SINK REEVALUATION
2. CALCULATION SM-ES-89-1500-001, ULTIMATE HEAT SINK - DEPTH VS. VOLUME AND DEPTH VS. SURFACE AREA CURVES
3. CALCULATION SM-ES-89-1500-002, ULTIMATE HEAT SINK POND 30-DAY POST-ACCIDENT HEAT LOADS AND FLOWS
4. CALCULATION SM-ES-89-1500-003, ULTIMATE HEAT SINK POND 30-DAY NORMAL SHUTDOWN HEAT LOADS
5. CALCULATION SM-ES-89-1500-004, SERVICE WATER POND - ULTIMATE HEAT SINK ANALYSIS

REV.	DATE	BY	CHK'D	DESCRIPTION	APPR. 1	APPR. 2	APPR. 3	APPR. 4	APPR. 5
0	9-5-90	[illegible]	[illegible]	APPROVED, ISSUED PER ES89-1500, REV. 0.	[illegible]	[illegible]			[illegible]

REMARKS

Southern Company Services, Inc. FOR ALABAMA POWER COMPANY

FARLEY NUCLEAR PLANT, UNITS 1 & 2
SERVICE WATER STORAGE POND - REQUIRED VOLUME AND SURFACE AREA

DESIGNED DSM | DRAWN REO | TYPED — | CHECKED [illegible] | SCALE NONE | CONTINUED ON SHEET —

DRAWING NUMBER B508559 | SHEET 1/1 | REV. 0

4.5.2 LEAK BEFORE BREAK ANALYSIS

Plant specific leak-before-break (LBB) analyses have been performed for both units of FNP and are summarized in the FSAR in Section 3.6. These analyses provide the technical justification for elimination of postulated breaks in the reactor coolant loop (RCL) piping (except for the accumulator and residual heat removal branch connections) and pressurizer surge line from the structural design basis. For the RCL, Westinghouse completed WCAP-12825, "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 Plants," in January 1991 (a supplement to WCAP-12835 also exists and is included in the TLAA evaluation). Subsequently, Westinghouse performed additional analyses of the RCL to evaluate the revised loadings associated with steam generator (SG) replacement and SG snubber elimination. Westinghouse completed WCAP-12835, "Technical Justification for Eliminating Pressurizer Surge Line Rupture from the Structural Design Basis for Farley Units 1 and 2" in April 1991. The Westinghouse Proprietary Class 3 version of WCAP-12825 is WCAP 12826, and the Proprietary Class 3 document for WCAP 12835 is WCAP-12834.

Both proprietary and non-proprietary versions of these WCAP documents were submitted to NRC for approval. WCAP-12835, its supplement, and the non-proprietary version were accepted by NRC through letter dated January 15, 1992, submitting the safety evaluation report (SER) for the analyses. Similarly, NRC accepted WCAP-12825 and WCAP-12826 through the SER in a letter dated August 12, 1991. These SERs accepted both the methodology and the results of both LBB analyses. A summary of the additional RCL LBB analysis performed to support SG replacement and SG snubber elimination was included in Section 2.1.2.2.3 of WCAP-15098, "Farley Nuclear Plant Units 1 and 2 Replacement Steam Generator Program NSSS Licensing Report"." SNC provided this Licensing Report to the NRC as Attachment 4 to FNP's "Steam Generator Replacement Related Technical Specifications Change Request," submitted December 1, 1998.

Since the methodology and results of these analyses have already been submitted to the NRC, they were not re-summarized in this response.

The aging effect that is addressed in these analyses is cracking. Specifically, leak-before-break crack stability evaluations were performed for enveloping critical locations. The enveloping critical locations were determined based on loading, pipe geometry, and fracture toughness considerations. A fatigue crack growth analysis was also carried out to demonstrate that fatigue crack growth is negligible. Assumptions in these analyses having a potential basis in the original 40-year term of operation are fracture toughness properties for cast austenitic stainless steel (CASS) materials (due to thermal aging considerations), and the design transients cumulative cycles.

For license renewal, SNC performed a TLAA evaluation of the primary loop and pressurizer surge line LBB analyses. SNC determined that no update of the pressurizer surge line LBB analysis was required for license renewal since the surge line does not contain CASS materials and since the transients assumed for 40 years are bounding for 60 years (see Section 4.3 of the FNP LRA). SNC determined that the primary loop analysis should be updated to account for the extended term since CASS materials are present.

At the request of SNC, Westinghouse revised the WCAP-12825 analysis of the primary loop piping to account for the additional thermal aging of the CASS materials for the period of extended operation and issued Addendum 1 in December 2002. The addendum is a Westinghouse Proprietary Class 2 document. The analysis accounts for the effects of thermal aging degradation of the CASS materials due to 60 years of operation. The analysis demonstrates, using faulted loads, that a margin of at least 2 exists between the critical flaw and the flaw having a leak rate of 10 gallons per minute (the detectable leakage flaw). A margin of 10 exists between the calculated leak rate from the detectable leakage flaw and the FNP leak detection capability of 1 gallon per minute. At the critical locations, the detectable leakage flaw is stable using the faulted loads. As required by action item 10 of the NRC Final Safety Evaluation Report for WCAP-14575-A, a margin on loads of 1 is satisfied. No CASS material for FNP Unit 1 and 2 primary loop piping has been replaced; therefore, the second component of action item 10 is not applicable to FNP.

In conclusion, the analysis for the pressurizer surge line was reviewed and determined to be acceptable as-is for the extended license term (demonstration in accordance with 10 CFR 54.21(c)(1)(i)). The analysis for the primary coolant loop has been evaluated and updated to address operation through 60 years (demonstration in accordance with 10 CFR 54.21 (c)(1)(ii)).

4.5.3 RHR RELIEF VALVE CAPACITY VERIFICATION CALCULATIONS

As described in section 5.2.2.4 of the FNP UFSAR, FNP has no dedicated cold-overpressure mitigation system. Instead, overpressure mitigation is provided for reactor coolant system (RCS) pressure excursions initiated by inadvertent mass and/or heat additions at low RCS temperatures through the use of the residual heat removal (RHR) system relieve valves (RHRSRVs). The RHRSRVs are aligned for RCS overpressure protection by opening the RHR suction valves inside containment at low RCS temperatures. SNC employs administrative procedures to minimize the potential for challenges to the system.

SNC has an analysis (the cold over-pressure mitigation analysis or COMA) that demonstrates the RHRSRVs flow capacity and setpoint are sufficient to mitigate a RCS cold overpressure transient. Three transients are addressed in the pressure analysis. The worst case mass input event assumed in the pressure transient analyses is the operation of three (3) charging pumps due to inadvertent emergency core cooling (ECCS) initiation with the RCS above 180°F. Below 180°F, power is locked out to two (2) of the three (3) charging pumps consequently inadvertent initiation of only one charging pump is assumed. The inadvertent starting of one of the main reactor coolant pumps (RCPs) with an initial differential temperature of 50°F existing between the RCS and the steam generator (steam generator secondary temperature greater than RCS primary temperature) prior to the RCS start is also evaluated (cf. UFSAR Section 5.2.2.4.3). The RCP start transient is the worst case heat input event evaluated.

The cold over-pressure mitigation analysis determines the RHRSVs flow capacity under various conditions of temperature and RCS pressure upstream of the relief valves. The RHRSV flow capacity is compared to the injection flow rates for each of the three analyzed transients to determine the RCS pressures necessary to achieve a flow exceeding each transient's injection flow. The pressure necessary is then compared to the pressure –temperature (P-T) heat-up and cooldown limit curves (required under Appendix G to 10 CFR Part 50 - see Section 4.2.5 of the LRA) to demonstrate RCS pressure will remain within these allowable limits. Currently, the COMA evaluates the P-T limit curves up to 48 EFPY. Since the COMA does not use a 54 EFPY P-T curve, and since the other criteria 1, 2, 4, 5, and 6 of 10 CFR 54.3 are met for the COMA, SNC has determined that the analysis is a TLAA.

As described in LRA Section 4.2.5, the P/T curves for FNP have not been updated for 54 EFPY (i.e., for the extended period of operation). At this time, the Unit 1 reactor vessel surveillance coupon that best models the end-of-life neutron irradiation of the vessel has been pulled and is being analyzed. This coupon was pulled after the cut-off date for materials used in support of the LRA preparation. The Unit 2 coupon that best models the end-of-life neutron irradiation is still in the reactor vessel. SNC plans to pull one of the two remaining coupons in the

Unit 2 reactor vessel at the next refueling outage (Spring 2004). Analysis of the coupons should provide acceptable data for further updating of the P/T curves. SNC intends to wait until the data from both coupons has been evaluated for acceptability (lack of excessive data-point scatter) before updating the P/T curves to 54 EFPY.

After the P/T curves are updated to address the extended period of operation, the COMA will also be updated to project the TLAA to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). SNC will update this analysis prior to entering the period of extended operation.