March 17, 2000

Mr. D. N. Morey
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - CORRECTION TO SAFETY EVALUATION RE: STEAM GENERATOR REPLACEMENTS (TAC NOS. MA4393 AND MA4394)

Dear Mr. Morey:

The Nuclear Regulatory Commission's letter of December 29, 1999, issued Amendment No. 147 to Facility Operating License No. NPF-2 and Amendment No. 138 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments changed the Unit 1 and Unit 2 Improved Technical Specifications to address changes associated with replacing the current Westinghouse Model 51 steam generators with Westinghouse Model 54F steam generators. Your letter of January 28, 2000, identified some typographical errors and recommended a change to our December 29, 1999, Safety Evaluation (SE). Enclosed are revised pages (Errata page to Unit 2 and pages 6, 9 and 10 to the SE). I apologize for any inconvenience this has caused.

Sincerely,

/RA/

L. Mark Padovan, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosure: As stated

cc w/encl: See next page

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ATTACHMENT TO LICENSE AMENDMENT No. 138

TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of Facility Operating License No. NPF-8 with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating area of changes. Pages noted with an "*" have changed only due to information rolling over from one page to another.

Remove	Insert	Remove	Insert
3.3.1-17	3.3.1-17	B 3.6.5-3*	B 3.6.5-3*
3.3.2-11	3.3.2-11	B 3.6.6-3	B 3.6.6-3
3.4.5-2	3.4.5-2	B 3.7.16-1	B 3.7.16-1
B 3.4.5-5	B 3.4.5-5	5.5-5	5.5-5
B 3.4.5-6	B 3.4.5-6	5.5-6	5.5-6
3.4.6-2	3.4.6-2	5.5-7	5.5-7
B 3.4.6-5	B 3.4.6-5	5.5-8	5.5-8
3.4.7-1	3.4.7-1	5.5-9	5.5-9
3.4.7-2	3.4.7-2	5.5-10*	5.5-10
B 3.4.7-1	B 3.4.7-1	5.5-11	5.5-11
B 3.4.7-2	B 3.4.7-2	5.5-12*	5.5-12
B 3.4.7-4	B 3.4.7-4	5.5-13*	5.5-13
B 3.4.7-5	B 3.4.7-5	5.5-14	5.5-14
3.4.13-1	3.4.13-1	5.5-15*	5.5-15
B 3.4.13-2	B 3.4.13-2	5.5-16	5.5-16
B 3.4.13-3*	B 3.4.13-3*	5.5-17	5.5-17
B 3.4.13-4*	B 3.4.13-4*	5.5-18*	5.5-18
3.4.16-1	3.4.16-1	5.5-19	Delete
3.4.16-2	3.4.16-2	5.5-20	Delete
3.4.16-4	3.4.16-4	5.5-21	Delete
B 3.4.16-1	B 3.4.16-1	5.5-22	Delete
B 3.4.16-2	B 3.4.16-2	5.5-23	Delete
B 3.4.16-3	B 3.4.16-3	5.5-24	Delete
B 3.6.1-2	B 3.6.1-2	5.5-25	Delete
B 3.6.2-2	B 3.6.2-2	5.6-5	5.6-5
B 3.6.4-1	B 3.6.4-1	5.6-6	5.6-6
B 3.6.5-2	B 3.6.5-2		

(NSSS) residual heat generation. The analysis demonstrated that the auxiliary feedwater system capacity is adequate to remove core decay heat, prevent overpressurizing the RCS, and prevent uncovering the reactor core.

e. Steam Generator Tube Rupture

The steam generator tube rupture (SGTR) event was re-analyzed using the hand calculation method which reflects the current plant licensing basis. This is the same method described in Farley FSAR Section 15.4, and is consistent with the analyses performed to support the Farley power uprate which was approved in the NRC's letter of April 29, 1998 (Ref. 14). The SGTR analysis was performed to determine the quantity of primary-to-secondary leakage from the SGs and the quantity of steam released to the environment. The results were used in the radiological consequences analysis to verify that the postulated offsite dose consequences are acceptable.

f. Best Estimate Large-Break LOCA

Section 4.1.1 of Farley's Replacement SG Program NSSS Licensing Report (Ref. 10) discusses large-break LOCA (LBLOCA) re-analyses for replacing SGs, adapting the Farley Best Estimate (BE) LOCA model used to perform the re-analyses, the process implemented to determine the adaptation, and LBLOCA results. We reviewed this information as discussed below.

(i) Replacement SG Adaptation Assessment Process and Farley BE LBLOCA Model

Section 4.1.1.2 of Farley's Licensing Analysis Report described the process SNC used to adapt the existing approved Farley licensing Best Estimate LOCA (BELOCA) methodology to reflect installing the replacement SGs. The process included all elements of the BE methodology and considered input values, reference transient assumptions, various uncertainty response surfaces and distribution functions, effect of the changed containment conditions, and superposition correction. The adaptation process as used in this evaluation report does not necessarily include any specific finding resultant from its application. Based on sensitivity studies and comparative assessments, SNC concluded that the uncertainty elements of the methodology retained the basic characteristics of the current BELOCA licensing methodology. SNC had to re-perform only the reference peak cladding temperature (PCT) calculation and the superposition calculations in establishing the Monte Carlo structure for the replacement SG final PCT calculations.

Based on our review, we conclude that the process used to adapt the existing BELOCA methodology to reflect installing the replacement SGs is acceptable because it is comprehensive and effective in identifying the necessary changes. We also conclude that the same overall process described in Farley Licensing Report Section 4.1.1.2 is acceptable for future SG change/LBLOCA analysis methodology assessments and adaptations, such as steam generator plugging levels outside those already considered in the present analyses. Based on this conclusion we find the adaptation process (Farley Licensing Report Section 4.1.1.2) is acceptable for reference in the Farley Core Operating Limits Report (COLR), Technical Specifications, or other licensing documentation and may be used in future analyses in which SNC changes similar SG parameter values.

specifying analysis inputs continue to assure that PCT-specific input values bound the corresponding as-operated plant values.

h. Future LOCA Re-analyses

In order to more effectively implement 10 CFR Part 50.46 reporting requirements and to validate the uncertainty analyses in the BE LBLOCA methodology, for future LOCA re-analyses it is necessary for SNC either to determine the bounding analysis from comparative sensitivity analyses of re-analysis scenarios for both units or to perform a plant specific bounding licensing basis analysis for each unit.

2.0 Design Basis Accidents and Transients not Re-analyzed

SNC indicated that the following design basis accidents and transients did not require re-analysis since either (a) they were bounded by the previously approved power uprate analyses, or (b) the analyses were not adversely impacted by the SG replacement (i.e., replacing the SGs requires only a minimal change to the current analysis of record, and the analysis still meets applicable acceptance criteria):

- hot leg switchover
- post-LOCA long-term core cooling
- rod ejection accident
- uncontrolled rod cluster control assembly (RCCA) withdrawal from a subcritical position
- RCCA misalignment
- uncontrolled boron dilution
- partial loss of forced reactor coolant flow
- startup of an inactive RCL
- loss of external electrical load and/or turbine trip
- excessive heat removal due to feedwater system malfunctions
- excessive load increase accident
- accidental depressurization of the RCS
- accidental depressurization of the main steam system
- inadvertent operation of the emergency core cooling system during power operation
- minor secondary system pipe breaks
- inadvertent loading of a fuel assembly into an improper position
- complete loss of forced reactor coolant flow
- single rod cluster control assembly withdrawal at full power
- single RCP locked rotor
- rupture of a control rod drive mechanism housing
- steam system piping failure at full power
- anticipated transient without scram

3.0 Technical Specification Changes and Evaluation

SNC has proposed to change the SG water level low-low setpoint from 25 percent to 28 percent and the allowable value from 24.6 percent to 27.6 percent in TS table 3.3.1-1, "Reactor Trip System Instrumentation," and in ITS Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation." SNC also proposes to change the SG water level high-high setpoint from 78.5 percent to 82 percent, and the allowable value from 78.9 percent to 82.4 percent in ITS Table 3.3.2-1, “Engineered Safety Feature Actuation System Instrumentation.”

TS LCO 3.4.7, “RCS Loops Mode 5, Loops Filled,” currently specifies that the secondary side water level of at least two SGs shall be > 74 percent wide range (WR). SNC has proposed to change the minimum SG water level to > 75 percent WR. ITS surveillance requirements 3.4.5.2, “RCS Loops Mode 3,” require SNC to verify SG secondary-side water levels every 12 hours for required RCS loops. SNC proposed changing the required water level in the surveillance requirements from ≥ 28 percent narrow range to ≥ 30 percent narrow range. In addition, ITS surveillance requirements 3.4.5.2, “RCS Loops Mode 3,” 3.4.6.2, “RCS Loops Mode 4,” and 3.4.7.2, “RCS Loops Mode 5, Loops Filled,” require that SG secondary side water levels be verified every 12 hours for required RCS loops. SNC has proposed to change the required water level in the surveillance requirements from > 74 percent WR to > 75 percent WR consistent with the proposed limiting condition for operation.

These above proposed changes resulted from analytical values associated with replacement SG design differences and new analyses. These changes provide acceptable results for all effected transients and accidents. We find these changes to be acceptable.

4.0 Conclusion

The staff has reviewed SNC’s proposed TS changes associated with replacing the SGs and SNC’s supporting re-analysis and evaluation of design basis accidents and transients. Based on the review, the staff concludes that the proposed TS changes are acceptable.

C. Containment Integrity

SNC has performed containment integrity analyses for replacing the SGs at current uprated power to ensure that the maximum pressure inside the containment will remain below the containment building design pressure of 54 psig if a design bases LOCA or MSLB inside containment should occur during plant operation. The analyses also established the pressure and temperature conditions for environmental qualification and operation of safety-related equipment located inside the containment and the containment leak rate test pressure.

SNC indicated that the containment functional analyses included assuming the most limiting single active failure and the availability or unavailability of offsite power, depending on which resulted in the highest containment temperature and pressure. Bounding initial temperatures and pressures for analyses were selected to envelope the limiting conditions for operation.

1.0 LOCA Containment Integrity Analyses

SNC has performed analyses to determine the containment pressure and temperature response during postulated LOCAs using mass and energy releases which incorporate revised Model 54F SG design parameters at the current uprated power level of 2775 MWt. As in the current analyses, the postulated LOCA analyses were performed for the double-ended hot leg (DEHL) guillotine break of reactor coolant pipe and the double-ended pump suction (DEPS) break. SNC determined that the DEHL break results in the most limiting pressure during the blowdown phase and that the DEPS break yields the highest energy flow rates during the post-blowdown period.

Joseph M. Farley Nuclear Plant

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