UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 147 AND 138 TO FACILITY

OPERATING LICENSES NPF-2 AND NPF-8

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1 AND UNIT 2

SOUTHERN NUCLEAR OPERATING COMPANY

DOCKET NUMBERS: 50-348 AND 50-364

Enclosure 3

Table of Contents

I. INTRODUCTION

By letter of December 1, 1998, as supplemented by letters of April 21, July 19, October 18, and November 11, 1999, the Southern Nuclear Operating Company, Inc. (SNC), et al., submitted a request for changes to the Joseph M. Farley Nuclear Plant, Units 1 and 2, Improved Technical Specifications (ITS). The requested changes would revise the ITS to address changes associated with replacing the current Westinghouse Model 51 steam generators (SGs) with Westinghouse Model 54F SGs. The April 21, July 19, October 18, and November 11, 1999, letters provided clarifying information that did not change the December 1, 1998, application and the initial proposed no significant hazards consideration determination.

II. BACKGROUND

The existing SGs in both Farley units are Westinghouse Model 51. The tubes are made of Alloy 600 and the tube support plates are made of carbon steel. Using these materials contributed, in part, to the existing SG tube degradation. As degradation occurred, SNC requested amendments to the plant technical specifications for various alternate tube repair criteria which the Nuclear Regulatory Commission (NRC) subsequently reviewed and approved. The various repair criteria are voltage-based alternate repair criteria, F* criteria, and sleeving repair criteria.

The new, replacement SGs will be Westinghouse Model 54F using thermally treated Alloy 690 tube material and stainless steel tube support plates. The Alloy 690 tubing material is more resistant to stress corrosion cracking than Alloy 600 tubing material. The Model 54F stainless steel tube support plates will be more resistant to magnetite formation than carbon steel support plates and minimize tube denting. Licensees that have used these materials in their replacement SGs have reported minimal tube degradation. SNC determined that the alternate tube repair criteria in the current ITS are unnecessary for the replacement SGs and requested removing this criteria from the Farley Units 1 and 2 ITS.

SNC plans to replace the Unit 1 SGs in spring 2000 and the Unit 2 SGs in spring 2001. The amendment request contains a proposed Unit 1 ITS changes set and a Unit 2 ITS changes set. The Unit 1 ITS changes set contains changed Unit 1 ITS and unchanged Unit 2 ITS. The approved Unit 1 ITS set will apply after SNC replaces the Unit 1 steam generators in spring 2000 until SNC replaces the Unit 2 steam generators in spring 2001. The Unit 2 ITS changes set contains the previously changed Unit 1 ITS and newly changed Unit 2 ITS. The approved Unit 2 ITS set will apply after SNC replaces both the Unit 1 and the Unit 2 steam generators.

III. EVALUATION

SNC performed analyses and evaluations in accordance with the Farley Nuclear Plant current licensing basis to support replacing the Westinghouse Model 51 SGs. Our evaluation of their associated ITS amendment request follows:

A. Mechanical and Structural Design

1.0 Structural Analysis

The basic design of the Model 54F SG is consistent with that of prior Westinghouse SGs. The Model 54F SG is considered a close replica of the current Model 51 SG. The structural design of the SG is based on the American Society of Mechanical Engineers Boiler & Pressure Vessel Code, Section III (ASME Section III), Subsections NB and NC, 1989 Edition. This edition is currently endorsed in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a and is, therefore, acceptable.

Westinghouse's WCAP-12825 (Ref. 1) technically justified applying leak-before-break (LBB) considerations to eliminate postulated loss-of coolant accident (LOCA) pipe ruptures from the Farley Nuclear Plant reactor coolant loop (RCL) design basis, in accordance with General Design Criterion (GDC) 4 of 10 CFR Part 50, Appendix A. In WCAP-12835 (Ref. 2) and its Supplement 1 (Ref. 3), Westinghouse similarly justified eliminating postulated pressurizer surge line pipe rupture. Our letters of August 12, 1991 (Ref. 4), and January 15, 1992 (Ref. 5), approved applying LBB considerations for eliminating breaks in the Farley Units 1 and 2 primary loop piping based on the Model 51 SGs.

Westinghouse excluded the thermo-hydraulic loads associated with RCL and pressurizer surge line LOCA ruptures from the structural design basis of the RCL piping and the SGs based on our Ref. 4 and Ref. 5 letters. In addition, Westinghouse did not postulate arbitrary, intermediate breaks in the main steamlines in accordance with revised Standard Review Plan (SRP), Section 3.6.2, Mechanical Engineering Branch Position 3-1 provisions.

Each of the current Model 51 SGs is restrained by 5 snubbers, for a total of 15 snubbers per unit. Westinghouse decided to install the Model 54F SGs without using snubbers since Westinghouse analyses eliminated the thermo-hydraulic loads resulting from the postulated RCL LOCAs. Westinghouse re-analyzed the reactor coolant system (RCS) with the existing RCL supports but excluded the five snubbers to verify that the RCS design meets current licensing basis (CLB) design criteria.

Westinghouse's analysis consisted of geometrically non-linear time-history dynamic structural analyses of the RCS subjected to various loading conditions with the Model 54F SG mass distribution and with the snubbers removed. Westinghouse performed these non-linear analyses to account for impact effects due to gapped bumper restraints in the RCS. The dynamic loading conditions consisted of time histories of the operating-basis earthquake (OBE), safe-shutdown earthquake (SSE), and thermo-hydraulic loads due to postulated pipe ruptures in either the main steamlines or the feedwater lines.

Westinghouse performed the time-history analyses using structural computer codes WECAN and WESTDYN. WCAP-8252 (Ref. 6) documents versions of these programs and the Farley Final Safety Analysis Report (FSAR) lists WCAP-8252 as a structural analyses reference. Westinghouse's analysis method is based on the pseudo-force approach which requires calculating the RCL system normal modes and frequencies for its implementation. Modal amplitudes and system response depend on the damping specified for the system. Westinghouse stated in their response of July 19, 1999 (Ref. 7), to our July 2, 1999, request for additional information (RAI) that their analysis used the CLB damping factors shown in Table 3.7-1 of the Farley Units 1 and 2 FSARs. The staff finds this acceptable.

Westinghouse performed a 10 CFR 50.59 LBB reevaluation of the RCL piping and the pressurizer surge line using the piping forces and moments resulting from the non-linear, seismic, structural analysis of the RCS. Westinghouse stated that the reevaluation showed that the recommended LBB margins in Draft Section 3.6.3 of the SRP (Ref. 8) were satisfied at the critical locations. This demonstrates the acceptability of applying LBB for the RCL piping with the Model 54F SGs. Westinghouse also found that the impact on the pressurizer surge line due to the Model 54F SG and a snubber elimination program was negligible. Therefore, the existing LBB analysis, based on the current RCS configuration with the Model 51 SG, remains valid.

2.0 RCS Stresses

The largest combined RCL piping stress resulting from the design loading combination of OBE, deadweight, and pressure was determined to be 21.1 ksi compared to the ASME Section III Code-allowable stress of 26.7 ksi. The largest combined stress under faulted conditions, consisting of SSE and either a main steamline break (MSLB) or a feedwater line break, was determined to be 39.5 ksi compared to the Code-allowable stress value of 53.4 ksi. The highest-stressed RCS equipment nozzle was found to be on the reactor pressure vessel (RPV). The largest nozzle stress intensity was found to be 92 percent of the allowable value under OBE conditions. The same nozzle under combined SSE and the highest loading from LOCA or MSLB or feedwater line break was found to be 69 percent of the allowable faulted condition allowable. The staff finds these results reasonable and acceptable.

3.0 RCL Supports

For the Model 54F SG, the highest stress in a column support was found to be 92 percent of the faulted allowable stress. For the reactor coolant pumps (RCPs), the stress was found to be 42 percent of the faulted allowable stress. The stress in the highest loaded bumper was determined as 31 percent of the upset allowable stress. The staff finds these results reasonable and acceptable.

In SNC's letter of July 19, 1999, Westinghouse also provided the design basis criteria for the support columns. These criteria are based on the structural code of record (Ref. 9) for Farley Units 1 and 2, modified to reflect faulted conditions. The staff has reviewed the Westinghouse response and finds it acceptable.

The reactions of the reactor vessel support structures to applied forces resulting from all load conditions were determined from a finite element analysis of these structures. The dynamic forces applied to these structures consist of a combination of forces obtained from the RCL analysis, reactor pressure cavity, and the reactor vessel internals analysis. The maximum stress intensity under normal or upset conditions was found to be 47 percent of the allowable stress intensity. The maximum stress intensity under faulted conditions was determined to be 38 percent of the faulted condition allowable stress intensity. The staff finds these results reasonable and acceptable.

4.0 Balance of Plant

SNC assessed the effects of installing the Model 54F SGs on balance of plant (BOP) safety-related structures and components not covered in WCAP-15098 (Ref. 10). The Farley Steam Generator Replacement Program BOP Licensing Report (Ref. 11) contains Farley Nuclear Plant BOP analyses and evaluations. Both SNC and Westinghouse provided input to these analyses and evaluations. SNC stated that CLB design criteria and analyses or previous power uprate submittals bound these structures and components. SG replacement impact on

BOP structure and component function and operation has been analyzed or evaluated and was found to have no significant adverse effect. The staff finds this reasonable and acceptable.

5.0 Conclusion

The staff finds that SNC has provided justification to support replacing the current Model 51 SGs with the Model 54F SGs based on the acceptability of applying LBB considerations to the RCL piping. The staff also finds that replacing the SG Model 51 with SG Model 54F will not significantly affect RCS structural integrity, and that the structural design of the RCLs and their supports meet Farley Nuclear Plant, Units 1 and 2, FSAR CLB design criteria.

B. Design Basis Accidents and Transients

SNC re-analyzed or evaluated design basis accidents and transients in support of the SG replacement as described below.

1.0 Re-analyzed Design Basis Accidents and Transients

SNC re-analyzed the following design basis accidents and transients to demonstrate that the applicable licensing criteria and requirements are satisfied considering the effects of the SG replacement. The analyses impacted by the SG replacement either support ITS changes or are not considered bounded by submittals previously reviewed by the NRC.

a. Loss of Normal Feedwater

Westinghouse re-analyzed the loss of normal feedwater transient using the RETRAN-02 computer code. Westinghouse's letter of June 6, 1997 (Ref. 12), submitted WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," to the NRC for review. Our letter of February 11, 1999 (Ref. 13), indicated our acceptance of WCAP-14882 for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation. SNC satisfactorily addressed each of these limitations as they relate to the Farley SG replacement.

The acceptance criteria for the loss of normal feedwater transient are as follows:

- The critical heat flux shall not be exceeded. Preventing departure from nucleate boiling (DNB) demonstrates this.
- Pressure in the reactor coolant and main steam systems shall be maintained below 110 percent of the design pressures.
- There shall be no propagation to a more serious event.

With respect to DNB, the loss of external electrical load event bounds the loss of normal feedwater transient. The loss of external electrical load event demonstrates that the minimum departure from nucleate boiling ratio (DNBR) is greater than the DNB acceptance criterion. With respect to RCS overpressurization, the loss of external load event also bounds the loss of normal feedwater transient. The loss of external load event demonstrates that the peak primary and secondary system pressures remain below 110 percent of design. Analysis results show that the pressurizer does not become water solid.

b. Loss of All AC Power to the Station Auxiliaries

The loss of all AC power event was re-analyzed using the RETRAN-02 computer code. The acceptance criteria for this event are as follows:

- The critical heat flux shall not be exceeded. Preventing DNB demonstrates this.
- Pressure in the reactor coolant and main steam systems shall be maintained below 110 percent of the design pressures.
- There shall be no propagation to a more serious event.

With respect to DNB, the complete loss of flow event bounds the loss of all ac power event. The complete loss of flow event demonstrates that the DNBR is greater than the DNB acceptance criterion. With respect to RCS overpressurization, the loss of external load event bounds the loss of all ac power to station auxiliaries event. The loss of external load event demonstrates that peak primary and secondary system pressures remain below 110 percent of design. Analysis results show that the pressurizer does not become water solid.

c. Main Steamline Rupture at Zero Power

The main steamline rupture was re-analyzed using the RETRAN-02 computer code. The event was conservatively analyzed at zero power with no decay heat. The acceptance criterion for this event is that the critical heat flux shall not be exceeded, which preventing DNB demonstrates. The results of the re-analysis demonstrated that the minimum DNBR for the steamline break event initiated at zero power remains above the limit value.

d. Major Main Feedwater Pipe Rupture

The major feedwater line rupture event was re-analyzed using the RETRAN-02 computer code. The acceptance criteria for this event are as follows:

- Pressure in the reactor coolant and main steam systems shall be maintained below 110 percent of the design pressures.
- Any fuel damage that may occur during the transient should be of sufficiently limited extent so that the core will remain in place and geometrically intact with no loss of core cooling capability.
- Any activity release must result in calculated doses at the site boundary being within a small fraction of 10 CFR Part 100 guidelines.

To ensure that these criteria are met, Westinghouse established an internal criterion that no bulk boiling occurs in the RCS following a feedwater line rupture before SG heat removal capability (being fed by the auxiliary feedwater system) exceeds nuclear steam supply system

(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 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 calculation 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.

(ii) Replacement SG LBLOCA Methodology

The Farley BE LBLOCA methodology is based on the Westinghouse BE LBLOCA methodology described in WCAP-12945-P-A, which acceptably considers within its uncertainty processes input parameter values over ranges that bound their as-operated plant values. In SNC's letter of November 22, 1999, SNC confirmed that SNC and Westinghouse have processes which ensure that the PCT-sensitive parameters (values) used as input to the BE LBLOCA analyses bound the as-operated plant values for Farley. From our review of the Farley BE LBLOCA methodology adaptation process, we noted quantitative effects due to the input of parameter values reflecting the replacement SGs. However, the approved Farley BE LBLOCA methodology itself has not changed because none of its elemental processes have changed. Therefore, based on the continued acceptability of the LBLOCA methodology and SNC's use of processes to ensure that PCT-sensitive input parameter values bound their as-operated plant values, we conclude the Farley replacement SG BE LBLOCA methodology is acceptable and applicable to Farley. We find that SNC's LBLOCA methodology is suitable for reference in the Farley TS and COLRs as long as the processes for specifying analysis inputs continue to assure that PCT-specific input values bound the corresponding as-operated plant values.

(iii) LBLOCA Analyses

SNC performed licensing basis replacement SG LBLOCA analyses using the acceptable BE LBLOCA methodology discussed in Section f. (ii) of this safety evaluation. The licensing basis case is a cold-leg split break with a break discharge coefficient of 1.0 and 20 percent SG tube plugging. SNC identified that Farley Unit 2 continued to be bounding based on sensitivity studies performed as part of the previous power uprate analyses and on qualitative assessments of applying those analyses findings to the replacement SG analyses.

SNC determined that ZIRLO fuel in the hot assembly bounded the other fuel types. The calculated licensing basis PCT is 2065 ^{o}F; the maximum total (pre-transient plus transient) cladding oxidation is 12 percent; and the maximum core-wide hydrogen generation is 0.6 percent. These values fall below the criteria specified in 10 CFR Part 50.46 (b). Compliance with the 10 CFR Part 50.46 (b) criteria that the core remains amenable to cooling and for long-term cooling is qualitatively the same as the acceptable results from the previous Farley (power uprate) BE LBLOCA analyses.

SNC used these bounding analyses for both units. Using the Farley Unit 2 analyses as a basis for both plants is acceptable because SNC has determined that the Farley 2 analysis is bounding for both plants. This is based on extensive sensitivity studies preformed for the power uprating of both Farley units with the approved licensing basis BE LBLOCA methodology and based on qualitative assessments adapting the sensitivity analyses results. However, significant differences in the two Farley unit designs could affect the validity of mutual portions of the uncertainty analyses in the BE LBLOCA model. Therefore, for future re-analyses, SNC must justify the applicability of the analytical results either by determining the bounding analysis from comparative sensitivity analyses of re-analysis scenarios for both units or by performing a plant-specific bounding licensing basis analysis for each unit.

We conclude that the Farley replacement SG BE LBLOCA analyses are acceptable because of the following:

- SNC used an applicable, acceptable LBLOCA model.
- Parameter input values and ranges bound the as-operated plant values and ranges.
- Calculated results comply with 10 CFR Part 50.46 (b) criteria.

We find that SNC's LBLOCA methodology is suitable for reference in the Farley TS and COLRs as long as the processes for specifying analysis inputs continue to assure that PCT-specific input values bound the corresponding as-operated plant values.

g. Small-Break LOCA Analyses

Section 4.1.2 of Farley's Licensing Analysis Report discusses small-break LOCA (SBLOCA) re-analyses to reflect replacing the SGs, identifies the SBLOCA methodology used to perform the re-analyses, indicated that sensitivity studies were used to determine the limiting SBLOCA case, and gave SBLOCA analyses results.

(i) SBLOCA Methodology

The SBLOCA methodology used to analyze both Farley units for the replacement SG configuration is the Westinghouse NOTRUMP model discussed in WCAP-10054-P-A, as adapted to include the COSI Safety Injection/Steam condensation model discussed in WCAP-10054-P-A/WCAP-11767-P, Addendum 2, Revision1, July 1997. This methodology has been approved for application to Westinghouse power reactor designs, including the Farley units. Therefore, the NOTRUMP with COSI SBLOCA analysis methodology applies to both Farley plants.

SNC's November 22, 1999, letter indicated that, for SBLOCA analyses, Westinghouse and SNC have processes to ensure that the values of PCT-sensitive parameters input into SBLOCA analyses bound their as-operated plant values for Farley. This assures that the input values for Farley SBLOCA analyses parameters will be appropriate, and therefore, the SBLOCA analysis methodology applies to Farley.

(ii) Sensitivity Studies and Results

SNC described sensitivity analyses, including break size, operating temperature, time-in-life and fuel type studies, that they performed to identify the worst-case SBLOCA scenario. The results for the bounding 3-inch SBLOCA analysis identify that the fuel with ZIRLO cladding is limiting. The calculated PCT is 2030 ^{o}F, and the calculated maximum local oxidation is 11.88%.

(iii) SBLOCA Conclusions

We conclude that the Farley Replacement SG SBLOCA analyses are acceptable because they were performed with an applicable acceptable SBLOCA model, with parameter input values and ranges that bound the as-operated plant values and ranges, and because the calculated results show compliance with the criteria of 10 CFR Part 50.46(b). We find that SNC's SBLOCA methodology is suitable for reference in the Farley TS and COLRs as long as the processes for 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 LBB
- 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. 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, 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.

SNC indicated that Westinghouse calculated the mass and energy releases in the containment for the Model 54F SG using Topical Report WCAP-10325-P-A. In this analyses, the 1979 ANS 5.1 decay heat model with 2 sigma uncertainty factor was used. The same Westinghouse Topical Report and decay heat model were used in the current licensing basis analyses. The mass and energy release analyses were presented in WCAP-15098 (Ref. 10).

The containment pressure and temperature response analyses for Model 54F SG were performed using the GOTHIC computer code. The use of GOTHIC computer code and input assumptions for Farley was determined acceptable in the current power uprate analyses. The primary change to the Model 54F SG containment analysis model from the power uprate model were the blowdown mass and energy releases because of replacement SG design. No other changes were made to the analyses which would significantly affect the calculated peaks. Only small changes to the residual heat removal heat transfer area and flow were modeled to represent plant design data with a margin for tube plugging.

Farley Model 54F SG analyses calculated a containment peak pressure of 43.8 psig and a peak temperature of 264°F for the LOCA (DEHL and DEPS breaks). The current uprate LOCA peak pressure and temperature were 43.0 psig and 263°F. SNC indicated that the change in the LOCA peak pressure from present 43.0 psig to 43.8 psig is mainly due to the increased blowdown mass and energy releases associated with the Model 54F SG design (more tubes in Model 54F than Model 51). The calculated peak LOCA pressure of 43.8 psig and temperature of 264°F remains below the containment design pressure of 54 psig and design temperature of 280°F. SNC has proposed to revise the ITS to change the containment leak rate test pressure from 43.0 psig to 43.8 psig in accordance with Appendix J requirements.

2.0 Main Steamline Break Containment Integrity Analysis

SNC has performed the Model 54F SG analyses to determine the containment pressure and temperature response during postulated MSLBs inside containment for limiting conditions of operation at current uprated power. As in the current uprated analyses, the Model 54F SG analyses were evaluated for initial power levels of 102 percent, 70 percent, 30 percent, and 0 percent and a spectrum of break sizes. SNC indicated that the Model 54F SG MSLB mass and energy releases were calculated using the RETRAN computer code as described in WCAP-14882-P under the limitations delineated in the report and the associated NRC safety evaluation. Earlier analyses used the LOFTRAN computer code. SNC presented the MSLB mass and energy release analyses in WCAP-15098. The staff has found the use of Non-LOCA RETRAN analyses performed in accordance with the methodologies as described in WCAP-14882 and the associated NRC safety evaluation acceptable.

Containment temperature and pressure were calculated using the GOTHIC computer code. The same code is used in the current uprate analyses. SNC indicated that the primary change to the Model 54F SG containment analysis model from the power uprate model was the blowdown mass and energy releases input due to Model 54F SG design. Another key input change was specifying 8 percent condensate revaporization, as allowed by NUREG-0588, Appendix B for the time in which the atmosphere is superheated. No other changes were made to the model which would significantly affect the calculated peaks. Bounding limiting initial containment conditions were assumed for maximum pressure and temperature results.

The Model 54F SG analyses calculated a peak containment pressure of 52.0 psig and a peak temperature of 367°F for the MSLB. The current uprate MSLB peak pressure and temperature were 52.4 psig and 383°F. SNC indicated that the small changes in peak pressure and temperature are mainly due to difference in MSLB mass and energy releases and 8 percent revaporization as allowed by NUREG 0588, Appendix B. The Model 54F SG MSLB peak calculated containment pressure of 52.0 psig remains below the containment design pressure of 54 psig. The MSLB peak air temperature of 367°F will last for < 6 minutes above 280°F and the containment structure temperature will remain below the containment design temperature of 280°F. Also, updated calculated pressure and temperature curves for LOCA and MSLB cases will remain bounded by the curves used for equipment qualifications.

Based on the above discussion, the staff finds the proposed ITS changes for replacement SGs will not affect the containment integrity as the calculated peak containment pressure and temperature remain below the containment design pressure and temperature and; therefore, are acceptable.

3.0 Short-Term Subcompartment Analysis

SNC indicated that they reviewed the short-term LOCA-related mass and energy releases to assess the effects associated with SG replacement. Although LBB approval for Farley included the pressurizer surge line, pressurizer surge line blowdown was used as bounding input for the RCL subcompartment analyses using GOTHIC computer code. Changes to reflect the Model 54F SG and new grating platforms were included in the model. The result demonstrated that the RCS loop subcompartment pressure remains bounded by previous design basis analysis. The pressurizer compartment was not re-analyzed since the pressurizer spray and surge line conditions do not change for SG replacement.

Based on the above review, the staff concludes that the SG replacement is acceptable as the subcompartment pressure loading analysis remains bounded by the current design basis subcompartment analysis.

D. Post-LOCA Hydrogen Generation

Hydrogen is generated following a LOCA inside containment from the zirconium-water reaction, corrosion of materials inside containment, radiolytic decomposition of core and sump solution, and hydrogen present in the reactor coolant and pressurizer vapor space. SNC indicated that the effect of SG replacement was reviewed for the above modes of post-LOCA hydrogen production and for combustible gas control system capability to maintain acceptable hydrogen concentration inside containment.

SNC stated that two 100 percent capacity hydrogen recombiners, post accident containment venting, post-accident containment mixing, and post-accident combustible gas sampling systems are provided to maintain hydrogen concentration below four volume percent within the containment. Farley Nuclear Plant procedures specify placing the recombiners in service within 1 hour after the LOCA, but SNC takes credit for their operation 1 day after the LOCA start.

SNC used the hydrogen generation model based on NRC Regulatory Guide 1.7 to evaluate the changes from SG replacement. SNC used the LOCA temperature profile from the containment temperature accident analysis as an input for evaluating the corrosion rate of containment materials which generate hydrogen. The analysis results indicate that the post-accident hydrogen concentration inside containment will not exceed the lower flammability limit of 4 volume percent with the operation of one hydrogen recombiner 24 hours after the start of the accident. Plant operators would initiate containment purging at day 7 following an accident in the unlikely event both recombiners fail to start. The dose due to containment purging added to the doses from other post-accident consequences would remain within the limits of 10 CFR Part 100. SNC determined the increase in hydrogen generation rate due to Model 54F SGs has a negligible effect on the hydrogen recombiners, post-accident venting, post-accident hydrogen mixing or sampling systems.

Based on the above review, the staff finds that the SG replacement is acceptable as it will not impact the post-LOCA hydrogen control system.

E. Radiological Dose Consequences

SNC proposes to increase both the instantaneous and the 48-hour ITS values for dose equivalent I-131 (DEI-131) in the reactor coolant to 0.5 mCi/gm. SNC performed analyses and evaluations to demonstrate that all acceptance criteria, including the dose criteria continue to be met for the proposed SG design change and DEI-131 increase.

SNC identified the following design basis accident analyses as being impacted by the change in SG design:

- SGTR
- MSLB
- Loss of Offsite Power
- Turbine Trip/Loss of Load
- RCP Locked Rotor

SNC indicated that the SG design change would not affect any other accident analyses. SNC calculated the doses associated with these accidents for individuals located offsite at the exclusion area boundary (EAB) and at the Low Population Zone (LPZ). SNC used assumptions consistent with the NUREG-0800 SRP, and also used the following major assumptions:

- Dose conversion factors are taken from International Committee on Radiation Protection Publication 30 in lieu of the Regulatory Guides or TID-14844.

- Iodine spike models as described in the appropriate sections of the SRP are used. Although not specifically required by the SRP, a pre-existing iodine spike is also modeled for the loss of offsite power analysis.

- RCS specific activity is assumed to be 1.0 μCi/gm DEI-131, and the RCS operational leakage is assumed to be 1 gpm to the SGs, therefore bounding the proposed ITS limits.

The staff reviewed SNC's calculation assumptions and performed confirmatory calculations. The following sections provide the results of the staff's assessment of SNC's re-analysis of the

FSAR Chapter 15 accidents affected by the SG design change and the change in primary coolant TS levels for dose equivalent I-131.

1.0 Steam Generator Tube Rupture

SNC evaluated SGTR radiological consequences using the guidance of SRP Section 15.6.3. Two cases were analyzed as described in the SRP:

Case 1 - An SGTR with a pre-accident iodine spike of 60 µCi/gm DEI-131 in the primary coolant.

Case 2 - An SGTR with an accident initiated iodine spike with an appearance rate 500 times the equilibrium RCS activity of 1.0 µCi/gm DEI-131.

Both cases assumed no tube uncovery and immediate flashing of primary-to-secondary leakage in the ruptured SG.

SNC's calculated results meet the SRP acceptance criteria for each case as shown in Table 1 below.

Table 1
SNC-Calculated SGTR Doses

		Dose (rem)		SRP Acceptance Criteria (rem)
		EAB	LPZ	
Thyroid	*Case 1	131	49	300
	**Case 2	18	7.3	30
Whole Body		0.26	0.1	2.5

* Case 1 = pre-accident iodine spike
**Case 2 = accident initiated iodine spike

SNC also evaluated the impact of SG tube uncovery during the event for the new SG design. This evaluation assumed 1.0 µCi/gm DEI-131 and a 1 gpm primary-to-secondary leak rate released directly to the environment for the first 30 minutes. The offsite dose results were within 10 percent of 10 CFR Part 100 limits. The staff reviewed SNC's assumptions and determined them to be acceptable. The staff also performed a confirmatory calculation using SNC's assumptions as submitted and confirmed the results. The staff finds the SGTR evaluation acceptable.

2.0 Main Steamline Break

SNC performed an evaluation of the radiological consequences of the MSLB using SRP Section 15.1.5 guidance, except that partition factors in the intact SGs were assumed to be limited to 10. Two cases were analyzed as described in the SRP:

Case 1 - An SGTR with a pre-accident iodine spike of 60 µCi/gm DEI-131 in the primary coolant.

Case 2 - An SGTR with an accident initiated iodine spike with an appearance rate 500 times the equilibrium RCS activity of 1.0 µCi/gm DEI-131.

Both cases assumed no tube uncovery nor immediate flashing of primary-to-secondary leakage in the affected SG. SNC's calculated results meet the SRP acceptance criteria for each case as shown in Table 2 below.

Table 2
SNC-Calculated MSLB Doses

		Dose (rem) EAB	Dose (rem) LPZ	SRP Acceptance Criteria (rem)
Thyroid	***Case 1**	7.4	3.4	300
	****Case 2**	6.9	3.3	30
Whole Body		$1x10^{-2}$	$4.6x10^{-3}$	2.5

* Case 1 = pre-accident iodine spike
**Case 2 = accident initiated iodine spike

SNC also evaluated the impact of SG tube uncovery during the event for the new SG design. This evaluation assumed 1.0 µCi/gm DEI-131 and a 1 gpm primary-to-secondary leak rate released directly to the environment for the first 30 minutes. The offsite dose results were within 10 percent of 10 CFR Part 100 limits. The staff reviewed SNC's assumptions and determined them to be acceptable. The staff also performed a confirmatory calculation using SNC's assumptions as submitted and confirmed the results. The staff finds the MSLB evaluation acceptable.

3.0 Loss of Offsite Power, Loss of Load, and Turbine Trip

SNC evaluated the radiological consequences of a loss of offsite power, loss of load, and turbine trip bounding steam releases for all three events. SNC followed the guidance given in SRP Sections 15.2.1 - 15.2.6, except the partition factors are assumed to be 10. There is no tube uncovery nor immediate flashing of primary-to-secondary leakage. SNC's calculated results meet the SRP acceptance criteria for each case as shown in Table 3.

Table 3
SNC-Calculated Loss of AC Power Doses

	Dose (rem) EAB	LPZ	SRP Acceptance Criteria (rem)
Thyroid	1.2	0.89	30
Whole Body	$2x10^{-3}$	$1x10^{-3}$	2.5

SNC also evaluated the impact of SG tube uncovery during the events for the new SG design. This evaluation assumed a pre-accident spike of 60 μCi/gm DEI-131 and a 1 gpm primary-to-secondary leak rate released directly to the environment for the first 30 minutes. The results were within 10 percent of 10 CFR Part 100 limits. The staff reviewed SNC's assumptions and determined them to be acceptable. The staff also performed a confirmatory calculation using SNC's assumptions as submitted and confirmed the results. The staff finds the evaluation acceptable.

4.0 Reactor Coolant Pump Locked Rotor

SNC performed an evaluation of the radiological consequences of a single RCP locked rotor assuming 20 percent of the fuel clad/pellet gap gas is released to the primary coolant with subsequent leakage to the SGs and secondary coolant. SNC followed the guidance given in SRP Section 15.3.3. SNC's calculated results meet the SRP acceptance criteria for each case as shown in Table 4 below. The staff reviewed SNC's assumptions and determined them to be acceptable. The staff also performed a confirmatory calculation using SNC's assumptions as submitted and confirmed the results. The staff finds the RCP locked rotor accident evaluation acceptable.

Table 4
SNC-Calculated RCP Locked Rotor Doses

	Dose (rem) EAB	LPZ	SRP Acceptance Criteria (rem)
Thyroid	6.0	8.7	30
Whole Body	0.65	0.34	2.5

5.0 Control Room Operator Doses

SNC stated that control room doses for accidents other than the large break LOCA are less than those for the LOCA and were not explicitly calculated for SG replacement. SNC previously calculated LOCA control room doses for power uprate. In the NRC's Farley power uprate safety evaluation of April 29, 1998, the staff performed confirmatory control room dose calculations for several design basis accidents (including the LOCA and all accidents SNC analyzed for the current SG replacement) to confirm that LOCA control room doses were bounding. The staff finds that since the control room atmospheric dispersion factors are the same for all postulated accidents and since the offsite dose results for the SG replacement indicate that the previous LOCA offsite doses bound the re-analyzed accidents, then the previously calculated LOCA control room doses are still bounding for the SG replacement.

6.0 Conclusion

The staff has assessed SNC's evaluations of those accidents where the change from Westinghouse Model 51 to Westinghouse Model 54F SGs would impact offsite dose consequences. The staff has concluded that for those accidents impacted, the resultant doses would not exceed the acceptance criteria given in the SRP for each accident. Therefore, the staff finds the proposed replacement of the Model 51 SGs with the Model 54F acceptable from an offsite radiological dose standpoint. The staff also finds the change to the reactor coolant activity level of dose equivalent I-131 in TS 3.4.16 and Figure 3.4.16-1 to be acceptable.

F. Operational Leakage and Steam Generator Tube Surveillance Program

For the Unit 1 SG replacement, SNC proposed to delete the requirements and references regarding sleeving repair criteria, F* criteria, and voltage-based alternate repair criteria in ITS 5.5.9. These alternate repair criteria will still apply to the Unit 2 SGs until their replacement in 2002. SNC also proposed to change the RCS operational leakage limit in ITS 3.4.13 for Unit 1 from 140 gallons per day to 150 gallons per day. This is acceptable because the limit of 150 gallons per day is consistent with the limit for Unit 2 and with the limit the staff recommended for other pressurized-water reactors.

For the Unit 2 SG replacement, SNC proposed to delete all requirements and references regarding sleeving repair criteria, F* criteria, and voltage-based alternate repair criteria in ITS 5.5.9 in Farley Units 1 and 2.

The sleeving repair criteria, F* criteria, and voltage-based alternate repair criteria are less restrictive than the traditional tube plugging limit which requires that any tube having a crack that is 40 percent through wall (tube wall thickness) or greater be removed from service. The staff finds that removing these alternate repair criteria is acceptable because the 40 percent through wall plugging limit in ITS 5.5.9 will remain the applicable plugging limit for defective tubes.

SNC also proposed to delete preservice tube inspection in ITS 5.5.9.4 which requires SNC to perform a preservice inspection after the field hydrostatic test and prior to initial power operation. SNC stated that all tubes in the replacement SGs will undergo a shop-performed baseline eddy current examination after an ASME Section III hydrostatic pressure test. The Section III hydrotest will be conducted at a test pressure of 1.25 times the design pressure. Under the current ITS requirement, a field hydrotest is performed in accordance with ASME Code Section XI with a lower test pressure than that of the Section III hydrotest. SNC stated that the field hydrotest will not affect the results of the shop-performed baseline tube inspection results. SNC concluded that the current ITS requirement for preservice tube inspection is unnecessary considering that all tubes will have undergone an ASME Section III hydrotest followed by a baseline eddy current examination before installation. The staff approved a similar technical specification change regarding preservice inspection for the North Anna Units 1 and 2 replacement SGs in a letter to Virginia Power Corporation on December 4, 1991. The staff agrees with SNC and finds that SNC proposed changes to preservice inspection in ITS 5.5.9.4 acceptable.

The staff finds that removing sleeving, F* criteria, and voltage-based alternate repair criteria to repair defective tubes from ITS 5.5.9 is acceptable because the more restrictive tube plugging limit of 40 percent through wall will remain as the applicable requirement in the ITS. The staff finds that revising the RCS operational leakage limit for Unit 1 is acceptable because the proposed new limit is consistent with past staff recommendation. The staff also finds that eliminating the preservice inspection requirements in ITS 5.5.9.4 acceptable because the replacement SGs will have undergone a hydrotest and a baseline eddy current inspection before installation. Thus, the staff finds the proposed amendment of the Farley Units 1 and 2 ITS to be acceptable.

IV. STATE CONSULTATION

In accordance with the Commission's regulations, on December 27, 1999, the NRC notified the Alabama State official, Mr. Jim McNees of the Office of Radiation Control, Alabama Department of Public Health, of the proposed issuance of the amendment. Mr. McNees had no comments.

V. ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (64 FR 56533, dated October 20, 1999). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

VI. CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and, (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: A. Cubbage, M. Hartzman, J. Tsao, R. Goel, M. Hart, F. Orr

Date: December 29, 1999

REFERENCES

1. WCAP-12825, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as a Structural Design Basis for the Joseph M. Farley Units 1 and 2 Nuclear Power Plants," January 1991 (Proprietary).

2. WCAP-12835, "Technical Justification for Eliminating Pressurizer Surge Line Rupture from the Structural Design Basis for Farley Units 1 and 2," April 1991 (Proprietary).

3. WCAP-12835, Supplement 1, September 1991 (proprietary).

4. NRC letter of August 12, 1991, "Safety Evaluation of Elimination of Dynamic Effects of Postulated Primary Loop Pipe Ruptures from Design Basis for Joseph M. Farley Units 1 and 2."

5. NRC letter of January 15, 1992, "Safety Evaluation of Elimination of Dynamic Effects of Postulated Pipe Ruptures in the Pressurizer Surge Line from Structural Design Basis for Joseph M. Farley Nuclear Plant, Units 1 and 2."

6. WCAP-8252 "Documentation of Selected Westinghouse Structural Analysis Computer Codes," Revision 1, May 1977.

7. Letter of July 19, 1999, from D. Morey, Southern Nuclear Operating Company, to the NRC Document Control Desk.

8. NUREG-800, Standard Review Plan for the Review of Safety Analysis Reports of Nuclear Plants, LWR Edition.

9. AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, 1969.

10. WCAP-15098, "Farley Nuclear Plant Units 1 and 2 Replacement Steam Generator Program NSSS Licensing Report," November 1998 (Proprietary).

11. Southern Company Services, "Farley Nuclear Plants Units 1 and 2 Steam Generator Replacement Program BOP Licensing Report," November 1998.

12. Letter from Liparulo, N.J., Westinghouse Electric Corporation, to NRC, forwarding WCAP-14882, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," dated June 6, 1997.

13. Letter from Frank Akstulewicz, NRC, to Hank Sepp, Westinghouse Electric Corporation, "Acceptance for Referencing of Licensing Topical Report WCAP-14882, RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," dated February 11, 1999.

14. Letter from Jacob I. Zimmerman, NRC, to Southern Nuclear Company, "Issuance of Amendments - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos: M98120 and M98121)," dated April 29, 1998.