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SOUTHERN COMPANY
Energy to Serve Your World℠

March 4, 2002

Docket No.: 50-348

NEL-02-0011

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

Farley Nuclear Plant, Unit 1
Request for Technical Specifications Change
Steam Generator Inspection Frequency Revision for the Spring 2003 Refueling Outage

Ladies and Gentlemen:

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Southern Nuclear Operating Company (SNC) is proposing a change to the Technical Specifications (TS) of Facility Operating License No. NPF-2 for the Farley Nuclear Plant Unit 1. The proposed one-time change revises the Steam Generator (SG) inspection frequency requirements in TS 5.5.9.3.a, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, to allow a 40 month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C-1 classification. This one-time change is proposed to eliminate unnecessary SG inspections during the upcoming Unit 1 Spring 2003 refueling outage, thus resulting in significant dose and cost savings as detailed in the attachment.

Farley Nuclear Plant, Unit 1, replaced SGs during a refueling outage that was completed in the Spring of 2000. The replacement SGs are Westinghouse design (Model 54Fs) and incorporate significant improvements, including thermally treated Alloy 690 tubing. During the Unit 1 Fall 2001 refueling outage, following the first cycle of operation after SG replacement, 100% of the tubing was inspected full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) with eddy current. The low row U-bends and a 20% sample of the hot leg top of tubesheet transition were inspected with the +Point rotating probe. There were no defective or degraded tubes. The Unit 1 Fall 2001 refueling outage inspection results along with the improved Westinghouse replacement SG design provide the

A001

basis for proposing an extension of the inspection interval for the Spring 2003 refueling outage from a maximum of 24 calendar months to a maximum of 40 months with one category C-1 inspection result.

We request approval of the proposed change prior to November 2002, to support planning efforts for the Unit 1, Spring 2003 refueling outage. SG inspections will be performed during the Unit 1 Fall 2004 refueling outage in accordance with the requirements in TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," and the Electric Power Research Institute's PWR Steam Generator Examination Guidelines.

Enclosure 1 gives a description and safety analysis of the proposed change. Enclosure 2 includes the marked-up TS page for the proposed change for the Farley Nuclear Plant, Unit 1. Enclosure 3 includes the associated TS pages with the proposed change incorporated for the Farley Nuclear Plant, Unit 1. Enclosure 4 provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92(c), "Issuance of amendment."

In addition, with respect to this proposed change there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed amendment satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment and the human environment is not affected by this amendment.

A copy of the proposed changes has been sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

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

Respectfully submitted,

DN Morey

Dave Morey

Sworn to and subscribed before me this 4th *day of* March *2002*

Glaria H. Buie

Notary Public

My Commission Expires: 06/07/2005

Enclosure 1: Description and Safety Analysis of the Proposed Changes
Enclosure 2: Marked-Up TS Page for Proposed Changes for Farley Nuclear Plant, Unit 1
Enclosure 3: Incorporated TS Pages for Proposed Changes for Farley Nuclear Plant, Unit 1
Enclosure 4: Information Supporting a Finding of No Significant Hazards Consideration

MJA/kw: Letter and Evaluation.doc

cc: Southern Nuclear Operating Company
Mr. D. E. Grissette, General Manager - Farley

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. F. Rinaldi, Licensing Project Manager - Farley

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. T. P. Johnson, Senior Resident Inspector – Farley

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

ENCLOSURE 1

FARLEY NUCLEAR PLANT, UNIT 1
DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

ENCLOSURE 1

FARLEY NUCLEAR PLANT, UNIT 1
DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Southern Nuclear Operating Company (SNC) is proposing a change to the Technical Specifications (TS) of Facility Operating License No. NPF-2 for Farley Nuclear Plant, Unit 1. The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in TS 5.5.9.3.a, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, to allow a 40 month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C-1 category. The proposed change is based, in part, on a 100% inspection of all SG tubes, which exceeds the current technical specification requirements of 18% over the first 40 months of SG operation. In accordance with TS 5.5.9.2, "Steam Generator (SG) Tube Surveillance Program, Steam Generator Tube Sample Selection and Inspection," C-1 category is defined as "Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective."

The proposed change is described in detail in Section E of this enclosure. Enclosure 2 includes the marked-up TS page for the proposed change for Farley Nuclear Plant, Unit 1. Enclosure 3 includes the associated TS page with the proposed change incorporated.

SNC requests approval of the proposed change prior to November 2002. This would support postponing SG inspections during the Unit 1, Spring 2003 refueling outage and would provide sufficient time for outage planning efforts. SG inspections will be performed during the Unit 1 Fall 2004 refueling outage in accordance with the requirements in TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," and the appropriate industry guidance, currently Electric Power Research Institute (EPRI) "PWR Steam Generator Examination Guidelines: Revision 5," Volume 1, September 1997. This proprietary document provides guidance for developing SG program inspection scope and frequency, identifying degradation mechanisms, and qualification of inspection techniques and personnel.

B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS 5.5.9.3.a, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," requires that subsequent inservice inspections of SG tubes after the first inservice inspection be performed "at intervals of not less than 12 nor more than 24 calendar months after the previous inspection." In accordance with the Extension Criteria in TS 5.5.9.3.a, if two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

C. BASES FOR THE CURRENT REQUIREMENTS

The inspection of the SG tubes ensures that the structural integrity of this portion of the Reactor Coolant System (RCS) will be maintained. Inservice inspection of SG tubes is essential in order to maintain surveillance of the condition of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to

corrosion. Inservice inspection of SG tubes also provides a means of characterizing the nature and cause of any tube degradation so that timely corrective measures can be taken.

D. NEED FOR REVISION OF THE REQUIREMENT

TS 5.5.9.3.a requires two consecutive inspection results in the C-1 category before the inspection interval can be extended from a maximum of 24 calendar months to a maximum of 40 months. This one-time change (i.e., extending the inspection interval for the Unit 1, Spring 2003 refueling outage from a maximum of 24 calendar months to a maximum of 40 calendar months) is proposed to eliminate unnecessary SG inspections during the upcoming Unit 1, Spring 2003 refueling outage. The one-time elimination of this SG inspection will result in a radiation dose savings of approximately 7 person-REM and a cost savings of approximately $2,600,000. These estimates are based on inspecting all three SGs.

Farley Nuclear Plant, Unit 1, replaced SGs during the refueling outage completed in the Spring of 2000. The replacement SGs are Westinghouse design (Model 54Fs) and incorporate significant design and material improvements, including thermally treated Alloy 690 tubing. During the Unit 1 Fall 2001 refueling outage following the first cycle of operation after SG replacement, 100% of the tubing was inspected full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) with eddy current. The low row U-bends (i.e., rows 1 and 2) and a 20% sample of the hot leg top of tubesheet transition were inspected with the +Point probe. There were no defective or degraded tubes. These inspection results along with the improved Westinghouse replacement SG design installed at FNP and industry experience regarding Westinghouse SG inspection results, provides the basis for extending the inspection interval that comes due during the upcoming Unit 1, Spring 2003 refueling outage from a maximum of 24 calendar months to a maximum of 40 months with one category C-1 inspection result.

E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed one-time change revises TS 5.5.9.3.a as follows.

- TS 5.5.9.3.a currently states, "If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months."

- TS 5.5.9.3.a will be revised to state, "If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months. ***[An exception to this Extension Criteria is that for Farley Unit 1 only, a one-time inspection interval extension of a maximum of once per 40 months is allowed for the inspection performed immediately following the 1R17 inspection. This is an exception to the Extension Criteria in that the inspection interval extension is based on the result of only one inspection result falling into the C-1 category."]***

F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The inspection of the SG tubes ensures that the structural integrity of this portion of the Reactor Coolant System (RCS) will be maintained. As discussed below, the Unit 1 Fall 2001 refueling outage inspection

results along with the improved Westinghouse replacement SG design installed at FNP and industry experience regarding Westinghouse SG inspection results provides the basis for proposing an extension of the inspection interval and demonstrating that the integrity of the RCS will be maintained.

As previously discussed, TS 5.5.9.3.a requires two consecutive inspections resulting in category C-1 classification before the inspection interval can be extended from a maximum of 24 calendar months to a maximum of 40 months. SNC is proposing that for the upcoming Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, the inspection interval be extended to a maximum of 40 months after only one SG inspection that resulted in all inspection results being classified in the C-1 category based on the following:

1) SG Design Improvements

Industry experience with recirculating SGs using mill annealed Alloy 600 tubing has lead to significant design improvements in replacement SG design and fabrication. Problems associated with tube degradation (e.g., stress corrosion cracking (SCC), intergranular attack (IGA), pitting, and wastage) have been addressed through changes in tube materials and stress relief. Problems associated with secondary system fouling and flow-induced vibration and wear have been addressed with changes to the tube bundle support system. These design improvements, along with others, have been incorporated into the Westinghouse replacement SG design and are discussed below.

- Thermally treated Alloy 690 tubing provides much improved resistance to stress corrosion cracking.

  Each of the three Farley Nuclear Plant, Unit 1, SGs contains 3,592 thermally treated Alloy 690 U-tubes that have a nominal outer diameter of 0.875 inches and a thickness of 0.050 inches. The development of thermally treated Alloy 690 tubing was prompted by the significant numbers of mill annealed Alloy 600 tubes being removed from service due to degradation. Thermally treated Alloy 690 tubing is similar to mill annealed Alloy 600 tubing but contains a 13% higher chromium content and correspondingly reduced nickel content. The higher chromium content reduces the degree of sensitization (i.e., the amount of chromium depleted in areas adjacent to the metal grain boundaries), thus increasing resistance to corrosion attack at the metal grain boundaries. Heat treatment of Alloy 690 for optimum SCC resistance involves mill annealing at temperatures sufficient to put all the carbon into solution, followed by a thermal treatment to precipitate carbides on the metal grain boundaries into the tube metal microstructure. Resistance to SCC is greatest when the metal grain boundaries are fully populated with carbides.

  Extensive testing has been performed which demonstrates thermally treated Alloy 690 tubing is superior to mill annealed Alloy 600 tubing in its resistance to both primary and secondary system SCC, pitting, and general corrosion. These tests demonstrated that thermally treated alloy 690 appears to be the most corrosion resistant tube alloy. Alloy 690 is already far better qualified than was either alloy 600 or alloy 800 when they were first selected for use in SGs. (Reference 1)

  The tubing procurement specification used in construction of the Farley Nuclear Plant, Unit 1 replacement SGs was designed to assure mill production of tubing that achieves the corrosion resistance properties as indicated by industry standards and research. The specification also outlines the physical, mechanical, and extensive inspection and qualification requirements necessary to limit fabrication defects. Cracks, laminations, scratches, draw-marks, pores, seams, laps, or stains are considered defects and are subject to rejection or conditioning in accordance with tested, approved, and controlled methods.

In addition to the thermal treatment process that was performed on all tubing, additional stress relief was performed on all U-bends up to a 12 inch centerline radius. The smallest centerline radius U-bend in the replacement SG design is 3.141 inches as compared to 2.1875 inches in the original SGs. This larger radius reduces residual stress in the low row U-bend region. The additional stress relief and larger minimum radius U-bend design provides added assurance that this region will not develop cracking.

Industry data (ie., reference 1) supports the laboratory test results demonstrating the superior performance of thermally treated Alloy 690 as compared to mill annealed Alloy 600 tubing.

During the Fall 2001 Farley 1 SG inspection, all tubes were inspected with eddy current. No stress corrosion cracking was detected.

- Enhanced anti-vibration bar (AVB) design provides for a more stable tube bundle, and limits potential for both wear and high cycle fatigue of tubes.

The original SGs installed at Farley experienced wear as a result of contact between the original AVBs and the tubes. Design improvements have reduced the probability of AVB wear in the replacement SGs.

Three sets of AVBs are installed in the U-bend region to stiffen the tube bundle, maintain proper tube spacing and alignment, and reduce tube vibration. The AVBs consist of V-shaped, rectangular cross section, bars of Type 405 stainless steel material.

The combined tube-to-AVB fitup gap is limited by use of tubes having tightened U-bend outside diameter ovality control at AVB crossover points, close tolerance AVB widths, and special AVB assembly procedures. Additionally, the AVBs allow tubes free thermal growth.

During the Fall 2001 Farley 1 SG inspection, all AVB/tube intersections were inspected with bobbin. There was no wear detected.

- Corrosion resistant tube support plate material limits potential for crevice corrosion product buildup.

The original Farley SGs experienced crevice corrosion product buildup and tube denting. Improved material selection has reduced the probability of this problem.

The tubes are intermittently supported on the secondary side by seven tube support plates. The support plates material is Type 405 stainless steel, a material with improved corrosion resistance over the carbon steel used in the replaced Farley SGs.

- Structural broach hole cutout in tube support plates improves axial flow within tube bundle and minimizes tube-to-tube support contact area.

The seven tube support plates contain structurally arranged, quatrefoil shaped holes. This design reduces tube dryout and chemical concentration where the tubes pass through the tube support plates.

- Increased number and types of external shell penetrations provides for better secondary side access for sludge and foreign object removal capabilities.

There are six secondary side handholes and two inspection ports. Four of the handholes are located 90 degrees apart at approximately 9 inches below the flow distribution baffle. Two of these handholes are on the tubelane centerline with the other two handholes perpendicular to the tubelane. The remaining two handholes are in line with the tubelane and provide access to the tube bundle between the flow distribution baffle and the first tube support plate. The two inspection ports are above the top tube support plate in line with the tube lane. Additionally, for access to the SG internals, there are two secondary side manways which provide access to the moisture separation equipment, the feedwater distribution ring, and the top of the tube bundle.

- Sludge collection provides a passive means of reducing the amount of sludge contained in the secondary-side flow, resulting in a reduction in the rate of sludge deposition on the tube bundle, thus enhancing steam generator reliability and performance.

  The operation of the sludge collector is based on the principle that suspended particles will settle if the flow velocity is less than the threshold settling velocity. Because the cross flow velocity in the sludge collector is less than the settling velocity, the suspended particles in the secondary side fluid will be "captured" in the sludge collector. The sludge collector is designed to operate passively during normal operating conditions.

- Flow distribution baffle (FDB) plate produces flow conditions on the secondary side of the tubesheet to minimize the size of the zone where sludge deposition can occur.

  The FDB is fabricated of Type 405 stainless steel and has octafoil shaped holes (instead of the quatrefoil holes in the tube support plates). The central region of the FDB is open. The result of this design is that the flow velocity across the tubesheet surface is increased and the low flow velocity region (i.e., sludge deposition zone) is in the center of the tube bundle, near the blowdown intake.

- Full depth hydraulic tube expansions minimize the depth of the crevice between the tubes and the top of the tubesheet, thus minimizing the accumulation of contaminants in the tubesheet crevice and minimizing the residual stresses.

  The tube-to-tubesheet joint accomplishes axial load resistance and the physical fastening of the tubing to the vessel. Original SG designs encountered severe corrosion problems with the SG tube-to-tubesheet joint region associated with open (i.e., unexpanded) crevices, and/or SCC at the high residual stress cold worked locations on the surface of the transition zone between the roller expanded and unexpanded tube. The Westinghouse replacement SG design incorporates the following features to address tube-to-tubesheet joint configuration concerns.

  - Full depth expansion to eliminate a tubesheet crevice that could result in accumulation of contaminants against the transition zone and minimize the risk of stress corrosion cracking.
  - Hydraulic expansion that leaves minimal residual stresses and cold work as compared to mechanical roller expansion techniques.

Overall, the improved design features incorporated in the Farley Nuclear Plant, Unit 1, replacement SGs provide reasonable assurance that SG tube integrity will be maintained over the proposed operating period.

2) First Outage Inspection Sampling

TS 5.5.9.3.a requires that the first inservice inspection of SG tubes be performed after six effective full power months but within 24 calendar months of initial criticality. This SG inspection requirement was satisfied during the Farley Nuclear Plant, Unit 1, Fall 2001 refueling outage. As required by TS 5.5.9.2, "Steam Generator (SG) Tube Surveillance Program, Steam generator Tube Sample Selection and Inspection," TS Table 5.5.9-1 which provides guidance on the minimum number of SGs to be inspected during inservice inspections, and TS Table 5.5.9-2, "Steam Generator Tube Inspection," the SG tube inspection requirements for the first inservice inspection were to inspect 3% of all the SG tubes in at least two SGs (4.5% of the tubes in the individual SGs). Farley Nuclear Plant inspected significantly more than the minimum TS requirement during the Unit 1 Fall 2001 refueling outage by performing 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) eddy current inspections of all three SGs. The low row U-bends and a 20% sample of the hot leg top of tubesheet transition area were inspected with the +Point rotating probe. This was in accordance with Section of 3.3.1, "Examination of Tubes," of the EPRI PWR Steam generator Examination Guidelines. The inspection was performed using ERPI PWR SG Examination Guidelines Appendix H, "Performance Demonstration for Eddy Current Examination," qualified techniques and Appendix G, "Qualification of Nondestructive Examination Personnel for Analysis of NDE Data," qualified data analysts and equipment. The inspection results showed no degraded or defective tubes, and all SGs were classified as category C-1. No tubes were plugged.

For the second inservice inspection of SG tubes, TS require that a minimum of 9% of the entire unit's tube population be inspected in at least one SG. By performing 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) eddy current inspection of all SG Tubes during the first outage after SG replacement (i.e., Unit 1 Fall 2001 refueling outage) Farley Nuclear Plant inspected significantly more SG tubes than would be required for both the first and second inservice inspections together, after SG replacement. Therefore, even though SNC is proposing a one-time extension of the interval between inspections, the scope of the inspections already performed during the Unit 1 Fall 2001 refueling outage (100% of SG Tubes) was significantly expanded from that required by the TS (18% of SG Tubes) over the first two refueling outages after SG replacement.

First outage inspection sampling results, along with industry experience and the operational assessment discussed below indicate that tube integrity will be maintained over the proposed operating period.

3) EPRI PWR Steam Generator Examination Guidelines

The EPRI PWR Steam Generator Examination Guidelines base SG inspection frequency on inspection results and performance criteria. We have followed the recommendations of Section 3.3.1 of the ERPI PWR Steam Generator Examination Guidelines, which contain the provisions regarding SG inspection frequency based on inspection results.

1. After the first cycle of operation (i.e., a duration not less than six effective full power months and not more than 24 effective full power months) for either new or replacement SGs, a 100% full-length (i.e., tube end to tube end) examination using general purpose eddy current probes shall be performed on all SGs.
2. During subsequent inservice inspections, if tube degradation (i.e., active damage mechanisms as defined in Appendix F, "Terminology," of the ERPI PWR SG Examination Guidelines) are identified, all SGs shall be examined at the end of each fuel cycle or 24 effective full power months, whichever is less, or as necessary to satisfy published regulatory requirements.

3. During subsequent inservice inspections, if active damage mechanisms are not identified, the number of SGs to be examined and/or the frequency of examination, shall be performed as required by Section 3.3.2, "Steam Generators Free from Active Damage Mechanisms," of the EPRI PWR Steam Generator Examination Guidelines.
4. 100% of the tubing and 100% of each type of repair shall be inspected within a rolling 60 effective full power month time frame. If 60 effective full power months occurs during an operating cycle, completion of that cycle is acceptable and is within the stated requirement.
5. No SG shall operate more than two fuel cycles between inspections.

As stated in Section 3.3.2 of the EPRI PWR Steam Generator Examination Guidelines, if the SGs are free from active damage mechanisms, some latitude is provided in terms of the number of SGs to be inspected and/or frequency of inspection. For these SGs, any of the following options may be performed.

1. Inspect ≥ 20% of the tubes and ≥ 20% of each type of repair in each SG at each refueling outage, or
2. Inspect ≥ 40% of the tubes and ≥ 40% of each type of repair in half the number of SGs at each refueling outage, or
3. Inspect ≥ 40% of the tubes and ≥ 40% of each type of repair in each SG at every other refueling outage.

Farley Nuclear Plant, Unit 1, has not detected any degradation during the first SG inspection after replacement, and, therefore, does not have an active SG damage mechanism as defined in Appendix F of the EPRI PWR Steam Generator Examination Guidelines. Therefore, Option 3 as indicated above will be met without performing SG inspections during the upcoming Unit 1, Spring 2003 refueling outage.

Inspection frequency in accordance with Section 5.0 "Steam Generator Assessments," of the EPRI PWR Steam Generator Examination Guidelines is also determined by measurement results against performance criteria. As part of our SG Program, both a Condition Monitoring Assessment and Operational Assessment are performed after each inspection and the results are compared to performance criteria. The performance criteria against which the results are compared involve SG tube structural integrity, accident-induced leakage, and operational leakage. The performance criteria are contained in Section 2.0 "Performance Criteria," of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" (Reference 2). The Condition Monitoring Assessment is retrospective and is intended to confirm that adequate SG integrity has been maintained since the previous inspection. The Operational Assessment is predictive and is intended to provide reasonable assurance that the performance criteria will be met throughout the next operating period. If it is determined that the performance criteria will not be met at the end of the next cycle of operation, then the operational cycle length must be adjusted accordingly.

4) Condition Monitoring Assessment

After completion of the Farley Nuclear Plant, Unit 1, Fall 2001 refueling outage SG inspections, a Condition Monitoring Assessment was performed in accordance with EPRI "Steam Generator Integrity Assessment Guidelines," Revision 1, March 2000. This proprietary document provides guidelines for evaluating the condition of SG tubes based on inspection results. The results showed that all performance criteria had been met based on full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) eddy current inspections of all of the tubes of all three SGs. Based on the Unit 1 Fall 2001 refueling outage inspection results of the "as found" condition of the SGs, all performance criteria were met.

5) Operational Assessment

An Operational Assessment was performed in accordance with EPRI Steam Generator Integrity Assessment Guidelines to evaluate the predicted condition of the SGs after two cycles of operation. The Operational Assessment is summarized below.

Farley Nuclear Plant, Unit 1, operated for 1.33 Effective Full Power Years (EFPY) during Cycle 17 (i.e., the first post-SG replacement cycle). The operational assessment evaluated Farley Nuclear Plant, Unit 1, for a combined Cycle 18 and Cycle 19 operating period. The combined length of both Cycle 18 and Cycle 19 is estimated to be about 2.7 EFPY.

A possible damage mechanism that could affect replacement SG tube integrity is wear from secondary side foreign objects. During the Unit 1 Fall 2001 refueling outage, sludge lancing was performed on the secondary side tubesheet region of all three SGs. Upon completion of sludge lancing, a video inspection was performed in this region to identify any foreign objects. No foreign objects detrimental to the SG tubing were identified. Minor debris was removed from the SGs as part of the inspection. At the present time, no foreign objects are known to be present in the SGs. This inspection along with improved SG design and Farley's Foreign Material Exclusion (FME) Program provides confidence that foreign object wear will not occur over the proposed operating period. In addition, Farley Unit 1 has an on line acoustic monitoring system to detect loose parts should they occur.

During the Unit 1 Fall 2001 refueling outage inspection, no forms of degradation were identified. Therefore, all structural and accident leakage performance criteria in Reference 2 are predicted to be met through the end of Cycle 19.

Farley Nuclear Plant meets the recommendations of the recently issued EPRI document, "PWR Primary-to-Secondary Leak Guidelines," Revision 2, dated April 2000. This proprietary document provides guidelines for detecting, monitoring, and mitigating SG primary-to-secondary leakage. Implementation of these guidelines provides assurance that proper monitoring and response will occur in the event primary-to-secondary leakage were to develop over the proposed operating period.

Farley Nuclear Plant, Unit 1, implements current industry guidelines with respect to primary and secondary water chemistry. No significant chemistry excursions have occurred which would indicate or lead to increased SG tube degradation.

6) Industry Data

Review of industry data for 54 plants with SGs containing thermally treated Alloy 690 tubing reveals no degradation mechanisms other than mechanical wear. Of the 54 plants, 49 of the Alloy 690 tubed SGs were placed in service prior to the Farley 1 SGs. Of the 49 plants with thermally treated Alloy 690 tubing that were in service prior to Farley Unit 1 SGs, 37 have hot leg operating temperatures in excess of the approximately 607°F at Farley Unit 1. Corrosion related degradation is not expected, particularly not early in the life of these SGs, due to the superior corrosion resistant properties of thermally treated Alloy 690 tubing. With regard to wear, there have been no reported instances of AVB wear in replacement SGs with the Westinghouse advanced AVB design incorporated in the Farley SGs. This industry experience covers a number of domestic nuclear power plants some with up to 7 EFPY of operation with replacement steam generators. Based on this information reasonable assurance exists that wear indications will not become structurally significant over the proposed cycle of operation.

7) Dose and Cost Impact

If the proposed change is not approved for the Unit 1, Spring 2003 refueling outage, the plan would be to perform 20% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of the three SGs. Assuming this scope, the following dose and cost impacts are predicted based on the Farley Nuclear Plant, Unit 1, Fall 2001 refueling outage inputs.

1. Accumulated personnel dose including SG platform setup, manway removal, eddy current inspection, and tube plugging is estimated to be approximately 7 person-REM.
2. The approximate cost associated with inspecting the three SGs including contractor craft support is $2,600,000.

Based on improved SG materials, design, first inservice inspection results, and industry experience, SNC has concluded that SG inspections are not necessary to meet the SG inspection objectives.

G. IMPACT ON PREVIOUS SUBMITTALS

SNC has reviewed the proposed changes regarding their impact on any previous submittals and has determined that there is no impact on any previous submittals.

H. SCHEDULE REQUIREMENTS

SNC requests approval of the proposed change prior to November 2002. This would support postponing SG inspections during the Unit 1, Spring 2003 refueling outage and allow for sufficient preplanning efforts. SG inspections will be performed during the Fall 2004 refueling outage in accordance with the requirements in TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," and the EPRI PWR Steam Generator Examination Guidelines.

I. REFERENCES

1. J. Peter N. Paine, ed. Steam Generator Reference Book, Revision 1, Volume 1. Electric Power Research Institute, EPRI TR-103824.

2. NEI 97-06, "Steam Generator Program Guidelines," Revision 1, January 2001.

ENCLOSURE 2

MARKED-UP TS PAGE FOR PROPOSED CHANGES FOR FARLEY NUCLEAR PLANT, UNIT 1

Affect Page
5.5-7

5.5 Programs and Manuals

5.5.9.2 Steam Generator Tube Sample Selection and Inspection (continued)

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

5.5.9.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

Insert 1 Here

(continued)

INSERT 1

[An exception to this Extension Criteria is that for Farley Unit 1 only, a one-time inspection interval extension of a maximum of once per 40 months is allowed for the inspection performed immediately after the Farley 1 1R17 inspection. This is an exception to the Extension Criteria in that the inspection interval is based on the result of only one inspection result falling into the C-1 category.]

ENCLOSURE 3

CLEAN TYPED INCORPORATED TS PAGES FOR PROPOSED CHANGES FOR FARLEY NUCLEAR PLANT, UNIT 1

List of Affected Pages

5.5-7
5.5-8
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5.5.9.2 Steam Generator Tube Sample Selection and Inspection (continued)

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

5.5.9.3 Inspection Frequencies

The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months. [An exception to this Extension Criteria is

(continued)

5.5.9.3 Inspection Frequencies (continued)

that for Farley Unit 1 only, a one-time inspection interval extension of a maximum of once per 40 months is allowed for the inspection performed immediately after the Farley 1 1R17 inspection. This is an exception to the Extension Criteria in that the inspection interval is based on the result of only one inspection result falling into the C-1 category.]

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of Specification 3.4.13.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

5.5.9.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.

(continued)

5.5.9.4 Acceptance Criteria (continued)

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube that contains imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspections.

(continued)

5.5.9.4 Acceptance Criteria (continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging of all tubes exceeding the plugging limit) required by Table 5.5.9-2.

Table 5.5.9-1

No. of Steam Generators per Unit	Three
First Inservice Inspection	Two
Second and Subsequent Inservice Inspections	One*

* The other steam generator not inspected during the first inservice inspection shall be reinspected. The third and subsequent inspections may be limited to one steam generator on a rotating schedule encompassing 3 N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the same sequence shall be modified to inspect the most severe conditions.

ENCLOSURE 4

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

ENCLOSURE 4

INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), "Issuance of amendment," a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Southern Nuclear Operating Company is proposing changes to the Technical Specifications (TS) of Facility Operating License No. NPF-2 for the Farley Nuclear Plant, Unit 1. The proposed changes revise the Steam Generator (SG) inspection frequency requirements in TS 5.5.9.3, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, to allow a one time only 40 month inspection frequency after one inspection, rather than after two consecutive inspections resulting in C-1 classification. This change is necessary to eliminate SG inspections during the Unit 1, Spring 2003 refueling outage, thus resulting in dose and cost savings.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed one-time change revises the steam generator (SG) inspection interval requirements in Technical Specification (TS) 5.5.9.3, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, to allow a 40 month inspection frequency after one inspection, rather than after two consecutive inspections with results that are within the C-1 category. C-1 category is defined as "less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective."

The proposed one-time extension of the Unit 1 SG tube inservice inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the Farley Nuclear Plants' Final Safety Analysis Report (FSAR). As such, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would not go undetected. Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last Farley Nuclear Plant, Unit 1, refueling outage (i.e., the first refueling outage following SG replacement) exceeded the TS requirements

for the first two refueling outages after SG replacement. That is, more tubes were inspected than were required by TS. Currently, Farley Nuclear Plant, Unit 1, does not have a SG damage mechanism, and will meet the current industry examination guidelines without performing SG inspections during the next refueling outage. Additionally, as part of our SG Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the Farley Nuclear Plant, Unit 1, Fall 2001 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with the improved SG design and reliability of Westinghouse replacement SGs, extending the SG tube inspection frequency does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change revises the SG inspection frequency requirements in TS 5.5.9.3.a, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Farley Nuclear Plant, Unit 1, Spring 2003 refueling outage, to allow a 40 month inspection interval after one inspection, rather than after two consecutive inspections, with inspection results within the C-1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during the last Farley Nuclear Plant, Unit 1, refueling outage (i.e., the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary-to-secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection frequency, and, therefore, will not give rise to new failure modes. In addition, the proposed change does not impact any other plant systems or components.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

The SG tubes are an integral part of the Reactor Coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system. The safety function of the SGs is maintained by ensuring the integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for second generation alloy 690 models of replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.