Progress Energy

James Scarola
Vice President
Harris Nuclear Plant
Progress Energy Carolinas, Inc.

DEC 0 8 2003

Serial: HNP-03-116
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/LICENSE NO. NPF-63
REQUEST FOR LICENSE AMENDMENT
TECHNICAL SPECIFICATION 4.4.5.3a.

Ladies and Gentlemen:

In accordance with the Code of Federal Regulations, Title 10, Part 50.90, Progress Energy Carolinas, Inc. requests a license amendment for the Harris Nuclear Plant (HNP) to Technical Specification 4.4.5.3a., "Steam Generator Surveillance Requirements." The proposed one-time change revises the Steam Generator (SG) inspection frequency requirements in Technical Specification 4.4.5.3a. for the HNP Fall 2004 refueling outage (RFO) 12. Specifically, the proposed change would allow a 40-month inspection interval after the completion of the first inservice inspection following SG replacement, rather than after two consecutive inspections resulting in C-1 classification. This proposed change is consistent with a similar license amendment for the Joseph M. Farley Nuclear Plant, Unit 2, approved by NRC letter dated July 14, 2003.

The HNP SGs were replaced during RFO-10, completed in January 2002. The replacement SGs are Westinghouse model Delta 75's which incorporate significant improvements including thermally-treated Alloy 690 tubing. The SG tubing was inspected to establish a baseline condition prior to operation (i.e., the "preservice inspection"). The first inservice inspection of the SG tubing was performed during the subsequent refueling outage completed in May 2003 (RFO-11). The results of this inspection showed that there were no active degradation mechanisms present in the SG tubes. Therefore, a one-time inspection interval of a maximum of 40 months is being proposed for the inspection performed immediately following RFO-11.

Attachment 1 provides the description, background, and technical analysis for the proposed change to the Technical Specifications.

Attachment 2 details, in accordance with 10 CFR 50.91(a), the basis for Progress Energy Carolinas, Inc.'s determination that the proposed change to the Technical Specifications does not involve a significant hazards consideration.

Attachment 3 provides the proposed Technical Specification change.

Attachment 4 provides the revised Technical Specification page.

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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. The proposed change to the Technical Specifications meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), therefore, no environmental assessment or environmental impact statement is required for approval of this application.

In accordance with 10 CFR 50.91(b), Progress Energy Carolinas, Inc. is providing the State of North Carolina with a copy of the proposed license amendment. Progress Energy Carolinas, Inc. requests that the proposed amendment be issued prior to May 1, 2004 to support planning for HNP RFO-12 which is scheduled for October 16, 2004.

Please refer any question regarding this submittal to Mr. John Caves at (919) 362-3137.

Sincerely,

James Scarola

JS/ajp

Attachments:

1. Description, Background, and Technical Analysis
2. 10 CFR 50.92 Evaluation
3. Proposed Technical Specification Change
4. Revised Technical Specification Page

Jim Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge, and belief and the sources of his information are employees, contractors, and agents of Progress Energy Carolinas, Inc.

Darlene S. Yarborough

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

Notary (Seal)
My commission Expires:
2-21-2005

c:
Mr. R. A. Musser, NRC Senior Resident Inspector
Ms. B. O. Hall, N.C. DENR Section Chief
Mr. C. P. Patel, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

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REQUEST FOR LICENSE AMENDMENT
TECHNICAL SPECIFICATION 4.4.5.3a.

DESCRIPTION, BACKGROUND, AND TECHNICAL ANALYSIS

Description

This letter is a request to amend the Harris Nuclear Plant (HNP) Technical Specifications. The proposed one-time change revises the Steam Generator (SG) inservice inspection frequency requirements in Technical Specification 4.4.5.3a. for the Fall 2004 refueling outage (RFO-12). Specifically, the proposed change would allow a 40-month inspection interval after the first inservice inspection following the SG replacement outage in RFO-10, rather than after two consecutive inspections resulting in C-1 classification. The proposed change is based, in part, on a 100% inspection of all open tubes. This inspection significantly exceeded the current technical specification requirements over the first 40 months of SG operation (i.e, 3% of the total number of tubes in all steam generators in the first inservice inspection and 3% of the total number of tubes in all steam generators in the second inservice inspection). In accordance with Technical Specification 4.4.5.2, "Steam Generator Tube Sample Selection and Inspection," the 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."

HNP requests approval of the proposed change by May 1, 2004 to support postponing SG inspections in the Fall 2004 refueling outage (RFO-12). The reason for this one-time change is to eliminate unnecessary SG inspections which will result in a savings of at least 4 REM in dose, one day of critical path time in the outage schedule, and approximately $1 million dollars in cost with no significant impact on nuclear safety.

Currently, Technical Specification SG Surveillance Requirement 4.4.5.3a. states:

"The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of Steam Generator Replacement. 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, 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;"

The proposed change is to add a one-time allowance following Technical Specification 4.4.5.3a. as follows:

"A one-time inspection interval of a maximum of once per 40 months is allowed for the inspection performed immediately following RFO-11. This is an exception to Specification 4.4.5.3a. in that the interval extension is based on the results falling into the C-1 category after one inspection."

Background

Technical Specification 4.4.5.3a. requires two consecutive SG tube inspection results in the "C-1" category following SG Replacement, not counting the preservice inspection, before the inspection interval can be extended from 24 calendar months to a maximum of 40 months.

This one-time extension change is proposed to eliminate unnecessary SG inspections during RFO-12. During the first outage following SG replacement (RFO-11), 100% of the open SG tubing was inspected. The SG tube inspection did not identify any degradation of the tubing from the first cycle of operation.

The HNP SG eddy current examination results following the first cycle after SG replacement are similar to other Westinghouse recent model Replacement SGs: J. M. Farley Units 1 and 2, Arkansas Nuclear One Unit 2, South Texas Project Unit 1, and V. C. Summer Nuclear Station. The Harris Replacement SGs incorporate many design and material improvements including Alloy 690 thermally-treated tubing, similar to the above-mentioned utilities, to mitigate potential tube degradation.

Technical Analysis

Steam Generator Design Improvements

The HNP replacement Westinghouse Model Delta 75 steam generators have significant improvements over the original Westinghouse Model D4 steam generators. Most of the problems with the original steam generators centered on the Alloy 600 mill annealed tubing. This tubing was subject to tube degradation mechanisms such as stress corrosion cracking (SCC), intergranular attack (IGA), pitting, and wastage. The problems associated with the early model steam generators were researched by the industry. The results involved a variety of design improvements in materials selection and fabrication. These changes have been incorporated into the Delta 75 steam generator.

The Westinghouse Delta 75 steam generator includes the following design improvements: Alloy 690 thermally-treated tubing, tube support plate design and material, full-depth hydraulically expanded tubes in the tubesheet, anti-vibration bars design and material, an internal sludge collector, and feedwater nozzle design. These features are discussed below:

Alloy 690 Thermally-Treated Tubing

The replacement Delta 75 steam generator has 6307 Alloy 690 thermally treated (690TT) tubes. The tubing material has undergone much research by the industry and has been the only material used for tubing the recent replacement steam generators (in the United States). Alloy 690TT tubing has been tested over a range of operating conditions and has demonstrated a resistance to primary water stress corrosion cracking. One of the contributing reasons for Alloy 690TT tubing to resist stress corrosion cracking has been attributed to the higher percentage of chromium in the material composition. The higher chromium content results in the material being less susceptible to grain boundary chromium depletion and is therefore more resistance to corrosion attack.

In addition to the heat treatment received by the all tubes, the first 17 rows (those less than 24" bend diameter) received an additional heat treatment after bending to reduce the residual stresses in this region. Also, in contrast to the old model D4 steam generator, the smallest U-bend in the Delta 75 steam generator has a larger radius; 3.25" (D75) vs 2.25 (D4). The smaller U-bend tubes typically have had the higher residual stress and therefore more degradation issues.

Tube Support Plate Design and Material

The tube support plate design has a flat-contact broached trefoil hole design. The support plate material is 405 stainless steel. The broached hole design is intended to reduce the tube-to-tube support plate crevice area while providing for maximum steam/water flow in the open areas adjacent to the tube. The support plate material selection is to minimize the potential for tube denting.

Technical Analysis (Continued)

Full-Depth Hydraulically-Expanded Tubes in the Tubesheet

The tubes in the Delta 75 are full-depth expanded within the tubesheet. The tubes have been expanded close to the top of the secondary tubesheet in order to minimize the tubesheet crevice. The tubesheet hydraulic expansion process was designed to minimize the residual stresses in the tube.

Anti-Vibration Bars (AVBs) Design and Material

There are four sets of 405 stainless steel rectangular-shaped anti-vibration bars installed in the U-bend region of the tubes to provide additional support to the tubes. The AVBs are staggered to minimize pressure drop in the U-bend region in order to minimize the impact on the secondary water circulation ratio, and also to reduce the potential for steam blanketing. The rectangular shape provides for a greater tube-to-AVB contact area and the gap between the AVB and support is tightly controlled so as to reduce the potential for tube wear.

An Internal Sludge Collector

The Delta 75 has an internal sludge collector located above the tubes. As the secondary water circulates around the inside of the steam generator, a portion of the water flows over the sludge collector. The region above the sludge collector was designed to allow solids to settle into the sludge collector. The sludge collector has been demonstrated in other similarly designed SGs to reduce the amount of suspended solids in the steam generator bulk water and the amount of sludge deposited on the tubesheet.

Feedwater Nozzle Design

Feedwater enters the steam generator via vertical spray tubes located on the top of a Feedwater header. The spray tubes are made from Alloy 690 material and are spread uniformly around the header. The spray tubes have 0.38" holes through which the feedwater flows into the steam generator. The 0.38" holes limit the size of foreign material that might be introduced into the steam generators.

Water Chemistry

HNP maintains plant primary and secondary water chemistry in accordance with current industry guidelines (e.g., NEI 97-06, Steam Generator Program Guidelines; EPRI TR-105714-V1, PWR Primary Water Chemistry Guideline; and EPRI TR-102134V, PWR Secondary Water Chemistry Guidelines). Proper water chemistry is essential for the health of the steam generator internal components.

Technical Analysis (Continued)

First Outage SG Inservice Inspection following SG Replacement

Technical Specification 4.4.5.3a. requires that the first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of Steam Generator Replacement. The steam generator replacement outage was completed January 3, 2002 (RFO-10). Two tubes were plugged during the steam generator manufacturing process. The first inservice inspection was performed during RFO-11 (May 7, 2003), within approximately 16 months of the steam generator replacement. During the first inservice inspection following steam generator replacement, Technical Specification Table 4.4-1, "Minimum Number of Steam Generators to be Inspected during Inservice Inspection" and Table 4.4-2, "Steam Generator Tube Inspection", require a minimum inspection of two steam generators and 4 ½% of the open tubes in both steam generators (Note: 4 ½% population in two steam generators is equivalent to the Technical Specification requirement 4.4.5.2 of 3% of the total number of tubes in all steam generators). These requirements were significantly exceeded by inspecting 100% of the open tubes in all three steam generators during RFO-11. This inspection did not detect evidence of tube degradation or evidence of defective tubes, and no additional tubes were plugged as a result of this inspection. Based on this inspection, the results are classified in the "C-1" category.

For the second inservice inspection of SG tubes, the Technical Specification Tables 4.4-1 and 4.4-2 require that a minimum of 9% of the tubes be inspected in at least one steam generator (NOTE: 9% population in one steam generator is equivalent to the Technical Specification requirement 4.4.5.2 of 3% of the total number of tubes in all steam generators). The 100% tube inspection performed in the first inservice inspection significantly exceeded the Technical Specification requirements for both the first and second inservice inspections after steam generator replacement. Therefore, even though this technical specification amendment is proposing a one-time extension of the interval between inspections, the scope of the inspections performed during RFO-11 (100% of SG tubes) was significantly greater than that required by the Technical Specifications over the first two outages following steam generator replacement.

An assessment of the "as-found" condition of the steam generator tubing was performed based on the first inservice inspection results (know as the "Condition Monitoring Assessment" in EPRI's Steam Generator Examination Guideline document). Tube corrosion mechanisms were not detected during the inspection. Evidence of tube degradation was not found from the tubing eddy current analysis. Benign signals identified during the pre-service inspection, such as manufacturing burnish marks, dings/dents, laps, were observed during the inservice inspection. Some of the benign signals were noted as having a slight signal phase rotation as compared to the preservice inspection. Phase rotation after the first cycle of plant operation has been observed by the eddy current analysts at other plants with replacement steam generators with similar tubing and is not evident of tube degradation.

Technical Analysis (Continued)

There were no signals indicative of loose parts detected during the inspection. In addition, the secondary side of the steam generators were sludge lanced leaving the tubesheets in a clean condition. A visual inspection was performed in all three steam generators in the tubesheet periphery and blowdown lane. There were no objects observed that had any size or mass during the inspection that would cause degradation to the SG tubing. Two "items" were retrieved that had magnetic properties, but they had no significant weight or size. These two metallic "remnants/dust" potentially came from the manufacturing process.

The areas of contact between the anti-vibration bars and the tubes did not have indications of mechanical wear. This is consistent with other steam generators that have similar U-bend support systems as the Westinghouse Model Delta 75.

An assessment of the condition of the steam generator tubing for two cycles of operation before the next inservice inspection indicates that the tubing is still expected to meet integrity criteria for this length of period. This is based on no detectable service-induced degradation from the RFO-11 inspection results, no detectable primary-to-secondary steam generator tube leakage and a review of industry data of similar SGs with thermally treated Alloy 690 tubing.

Replacement model SGs with thermally-treated Alloy 690 tubing have shown few indications of mechanical wear. Some of the tube wear involved loose parts in the secondary of the steam generator. The small diameter holes in the Delta 75 feedring spray tubes (discussed above) mitigate the potential for large detrimental objects to inject into the steam generator secondary side. In addition, there is a metal impact monitoring system installed on the HNP steam generators to detect loose parts should they occur. Another mode of mechanical wear noted has been due to contact with the anti-vibration bars. The HNP RFO-11 inspection noted there was no detectable wear at the anti-vibration bar area. This is consistent with VC Summer's model Delta 75 steam generator inspection results and South Texas model Delta 94 (Unit 1) inspection results. It was reported that VC Summer had three wear indications at the anti-vibration wear contact areas, but that they could be traced back to pre-service baseline data and were therefore not service induced. With no evidence of service-induced degradation from the first inservice inspection at HNP, it is expected that the tubing will still meet their structural requirements at the next inservice inspection, which would be during RFO-13 (April 2006) pending approval of this amendment request.

Conclusion

Progress Energy Carolinas, Inc. 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Precedent

There are other recent utilities that have replaced steam generators with Alloy 690 tubing that have requested a one-time change to the steam generator tube inspection frequency in their technical specifications. The change allowed a 40-month inspection interval after the first inservice inspection, rather than after two consecutive inspections resulting in a C-1 classification. The following is a listing of the known utilities with the appropriate correspondence letters and the associated NRC approval of the requests.

Farley Unit 2

Southern Nuclear Operating Company, Inc. letter NL-03-0134 to the NRC dated February 11, 2003, "Joseph M. Farley Nuclear Plant, Unit 2, Request for Technical Specifications Change, Steam Generator Inspection Frequency Revision for the Spring 2004 Refueling Outage" ML030520106

NRC letter to Mr. J. B. Beasley, Jr., Southern Nuclear Operating Company, Inc. dated July 14, 2003, "Joseph M. Farley Nuclear Plant, Unit 2 Re: Revising the Steam Generator Inspection Frequency (TAC No. MB7938)" ML031950574

ANO Unit 2

Entergy Letter 2CAN110204 to the NRC dated November 22, 2002, "Arkansas Nuclear One, Unit 2, Docket No. 50-368, Operating License Amendment Request to Modify Steam Generator Tube Inspection Frequency" ML023600429

NRC letter to Mr. Craig G. Anderson, Entergy Operations, Inc. dated May 28, 2003, "Arkansas Nuclear One, Unit 2 – Issuance of Amendment Re: One-Time Change of Steam Generator Tube Inspection Frequency (TAC No. MB6808)" ML031490475

Farley Unit 1

Southern Nuclear Operating Company, Inc. letter NL-02-0011 to the NRC dated March 4, 2002, "Farley Nuclear Plant, Unit 1, Request for Technical Specifications Change, Steam Generator Inspection Frequency Revision for the Spring 2003 Refueling Outage" ML020650389

NRC letter to Mr. D. N. Morey, Southern Nuclear Operating Company, Inc. dated September 20, 2002, "Joseph M. Farley Nuclear Plant, Unit 1 Re: Issuance of Amendment (TAC No. MB4310)" ML022340746

Precedent (Continued)

South Texas Unit 1

South Texas Project letter NOC-AE-02001351 to the NRC dated June 20, 2002, "South Texas Project, Units 1 and 2, Docket No. STN 50-498 and STN 50-499, License Amendment Request – Revised Proposed Amendment to Technical Specification 4.4.5.3a" ML021780019

NRC letter to Mr. William T. Cottle, STP Nuclear Operating Company dated July 31, 2002, "South Texas Project, Unit 1 – Issuance of Amendment on Steam Generator Surveillance Requirements (TAC No. MB3963)" ML022040265

Braidwood Unit 1

Exelon letter RS-01-011 to the NRC dated February 9, 2001, "Request for Technical Specifications Change, Braidwood Station, Unit 1, Steam Generator Inspection Frequency Revision for the Fall 2001 Refueling Outage" ML010470080

NRC letter to Mr. Oliver D. Kingsley, Exelon Nuclear dated August 9, 2001, "Issuance of Amendments – Technical Specifications Changes to revise Steam Generator Inspection Frequency, Braidwood Station , Units 1 and 2 (TAC Nos. MB1226 and MB1227)" ML012040245

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TECHNICAL SPECIFICATION 4.4.5.3a.

10 CFR 50.92 NO SIGNIFICANT HAZARDS EVALUATION

A written evaluation of the significant hazards consideration of a proposed license amendment is required by 10 CFR 50.92. Progress Energy Carolinas, Inc. has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92, 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

The basis for this determination is presented below.

Proposed Change

The change involves a one-time revision to the steam generator (SG) inservice inspection frequency requirements in Technical Specification 4.4.5.3a. to allow a 40-month inspection interval after the first inservice inspection following SG replacement, rather than after two consecutive inspections resulting in C-1 classification. The proposed change will add a one-time allowance following Technical Specification 4.4.5.3a. as follows:

"A one-time inspection interval of a maximum of once per 40 months is allowed for the inspection performed immediately following RFO-11. This is an exception to Specification 4.4.5.3a. in that the interval extension is based on the results falling into the C-1 category after one inspection."

Basis

This change does not involve a significant hazards consideration for the following reasons:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed amendment revises the steam generator inspection frequency to allow a 40-month inspection frequency after the first inservice inspection following SG replacement, rather than after two consecutive inspections resulting in C-1 classification. The "C-1" category is defined in the Technical Specifications as having inspection results that indicate "less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective."

   The 100% inspection of the open steam generator tubes performed during RFO-11 represents a quantity of tubes inspected that is significantly greater then the amount required by the Technical Specifications over two successive inspective periods (i.e., 3% of the total number of tubes in all steam generators required in the first inspection following SG replacement and the same quantity of the tubes to be examined in the second inspection). The RFO-11 100% tube inspection did not indicate the tubes had experienced degradation from the cycle of operation.

   The assessment of the condition of the steam generator tubes indicated the structural condition of the tubing had not changed during the first cycle of operation following steam generator replacement and these results that indicated the tubes would still meet their structural criteria over the proposed inspection frequency. The steam generator tube inspection meets the current industry examination guidelines without performing inspections during the next refueling outage.

   The steam generator inspection frequency extension does not introduce a new failure mode or impact any other plant systems or components. The proposed change does not alter plant design. The HNP steam generator tubes do not have an active damage mechanism which could lead to the potential of primary-to-secondary steam generator leakage.

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

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change to extend the steam generator tube inspection frequency does not impact the design or operation of the steam generators or any other plant structure, system or component. Extending the inspection frequency of the steam generator tubes does not introduce any new failure modes. The proposed change does not alter plant design basis, or alter any potential accident previously evaluated.

Basis (Continued)

The proposed change revises the steam generator inspection frequency to allow a 40-month inspection interval after the first inservice inspection following SG replacement, rather than after two consecutive inspections resulting in C-1 classification. The first steam generator inspection following replacement inspected 100% of the open tubing in all three steam generators. This inspection exceeded the existing technical specification inspection over the two consecutive inspections. This inspection indicated there was no service-induced degradation in the steam generator tubes. The HNP first cycle inspection results were comparable with other recent Westinghouse model replacement steam generators.

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

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The steam generator tubes are an integral part of the reactor coolant system pressure boundary. The tubes are expected to maintain primary system pressure and inventory. The tubes are a barrier to keep radioactive fission products in the reactor coolant system from transferring to the secondary system. The steam generator tubes transfer the heat from the primary system to the secondary system. The ability of the steam generator tubes to perform these functions depends on the integrity of the tubes.

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

Therefore, the proposed inspection frequency change does not involve a significant reduction in a margin of safety.

Based on the above, Progress Energy Carolinas, Inc. concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

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PROPOSED TECHNICAL SPECIFICATION CHANGE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 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 Steam Generator Replacement. 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, 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~~;~~. Add 1 Delete

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b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 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 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and 1 Delete

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions: 1 Delete

1. Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

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

3. A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

4. A main steam line or feedwater line break.

A one-time inspection interval of a maximum once per 40 months is allowed for the inspection performed immediately following RFO-11. This is an exception to Specification 4.4.5.3a. in that the interval extension is based on the results falling into the C-1 Category after one inspection.

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REVISED TECHNICAL SPECIFICATION PAGE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 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 Steam Generator Replacement. 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, 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. A one-time inspection interval of a maximum once per 40 months is allowed for the inspection performed immediately following RFO-11. This is an exception to Specification 4.4.5.3a. in that the interval extension is based on the results falling into the C-1 Category after one inspection;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 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 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

    1. Reactor-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

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

    3. A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

    4. A main steam line or feedwater line break.