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2CAN110204

CORRECTED COPY

November 22, 2002

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

SUBJECT: Arkansas Nuclear One, Unit 2
Docket No. 50-368
Operating License Amendment Request to Modify Steam Generator Tube Inspection Frequency

REFERENCES:

1 Entergy letter to NRC dated June 26, 2002, *Revision of Section 6.0, Administrative Controls for Consistency with ANO-1 Improved Technical Specification* (2CAN060203)

2 Entergy letter to NRC dated June 29, 2000, *Response to Request for Additional Information Regarding the August 18, 1999, Steam Generator Inspection Requirements License Amendment Request* (2CAN060013)

Dear Sir or Madam:

Pursuant to 10CFR50.90, Entergy Operations, Inc. (Entergy) hereby requests an operating license amendment for Arkansas Nuclear One, Unit 2 (ANO-2) to Technical Specification (TS) 4.4.5.3.a. The proposed one-time change revises the steam generator (SG) inservice inspection frequency requirements in TS 4.4.5.3.a to allow a 40-month inspection interval after one inspection given a C-1 classification rather than after two consecutive inspections. This is considered acceptable since the steam generators and tubes are new and the 2R15 inspection resulted in category C-1.

The ANO-2 steam generators were replaced during the 2R14 refueling outage in the fall of 2000 with Westinghouse Delta 109 SG designed steam generators. The replacement SG tube material is thermally treated (TT) Alloy 690. Based on extensive industry-wide test programs, Alloy 690 TT has been proven through both laboratory testing and operational experience to provide increased corrosion resistance compared to Alloy 600. The SG tubes were preservice examined prior to initial installation. The first in-service steam generator inspection was performed during the subsequent 2R15 refueling outage (spring of 2002). 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 once per 40 months is being proposed for the inspection performed immediately following 2R15. This is an exception to 4.4.5.3.a in that the interval extension is based on the results falling into the C-1 category after

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one inspection. The details of this request are provided in Attachment 1. The revised mark-up TS page is provided in Attachment 2.

A proposed change to the ANO-2 Operating License was requested on June 26, 2002 (Reference 1) to relocate various portions of the ANO-2 TSs to the Administrative Controls section (6.0) of the TSs. Technical Specification 4.4.5.3 was one of those sections being proposed for relocation to Section 6.0. Therefore, there is an outstanding amendment request that affects the proposed change being requested in this amendment request.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. There are no commitments being made in this amendment request.

Entergy requests approval of the proposed amendment by August 15, 2003. Once approved, the amendment shall be implemented within 60 days.

If you have any questions or require additional information, please contact Steve Bennett at 479-858-4626.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 22, 2002.

Sincerely,

CGA/sab

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)

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Attachment 1

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Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2). The proposed one-time change revises the steam generator (SG) inservice inspection frequency requirements in TS 4.4.5.3.a to allow a 40-month inspection interval after one inspection given a C-1 classification rather than after two consecutive inspections. This is considered acceptable since the steam generators and tubes are new and the 2R15 inspection resulted in category C-1.

The reason for this one-time change is to eliminate the SG inspection during 2R16, which, if performed, would result in added dose, schedule, and cost with no safety benefit.

2.0 PROPOSED CHANGE

Currently, TS 4.4.5.3.a states, in part:

> 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.

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

> A one-time inspection interval of a maximum of once per 40 months is allowed for the inspection performed immediately following 2R15 This is an exception to 4.4.5.3.a in that the interval extension is based on the results falling into the C- I category after one inspection.

Since this proposed change is a one-time change to allow only one inspection cycle after the ANO-2 SG replacement and since TS 4.4.5.3.a is being moved to the Administrative section of the TSs per Reference 1, no TS Bases changes are being proposed.

3.0 BACKGROUND

Technical Specification 4.4.5.3.a requires that subsequent inservice inspections of SG tubes after the first inservice inspection be performed at intervals of not less than 12 calendar months nor more than 24 calendar months after the previous inspection. Furthermore, if two consecutive inspections following service under all volatile treatment (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.

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.

4.0 TECHNICAL ANALYSIS

ANO-2 Steam Generator Design Improvements

Industry experience with recirculating SGs using mill annealed Alloy 600 tubes has led to significant design improvements in replacement steam generator (RSG) design and fabrication. Problems associated with tube degradation, such as 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, and through design of the emergency feedwater headers for loose parts control. These design improvements, along with others, have been incorporated into the Westinghouse Delta 109 SG design now in service at ANO-2.

The steam generators were replaced in the 2R14 outage (fall of 2000). Even though ANO-2 is a Combustion Engineering unit the replacement steam generators (RSGs) were designed by Westinghouse and many of the design features are based primarily on other RSGs designed by Westinghouse. In establishing the design features for the RSGs, the key design features of both the ANO-2 original steam generators (OSGs) and other Westinghouse manufactured replacement steam generators were evaluated in order to establish the appropriate technical design requirements and design features for ANO-2. Design improvements of the ANO-2 RSGs, includes:

- improved tube support plate design to minimize the entrapment of corrosive materials,
- improved tubing composition and fabrication controls to minimize the susceptibility to corrosion mechanisms (including SCC, intergranular corrosion, and wastage), and
- improved tube-to-tubesheet expansion process to minimize corrosion potential.

The RSG tube bundle was sized to provide greater heat transfer capability than the OSG while essentially maintaining the primary side flow resistance. The lower shell diameter was increased by 4-inches and the tubes were distributed uniformly. The overall tube bundle height was raised approximately 18-inches. With these changes in combination with changing the tube size from 0.75-inch to 0.688-inch outside diameter (OD), it was possible to produce a tube bundle with a nominal surface area of 108,700 ft^2 (vs. the OSG surface area of 86,559 ft^2). The tube pitch remained triangular for the RSG; however, the pitch was reduced from 1-inch to 0.95-inch. This RSG tube diameter/pitch creates a slightly larger sludge lancing lane for the RSG than existed for the OSG.

The RSG tube material is thermally treated Alloy 690 (Alloy 690 TT). Based on extensive industry-wide test programs, Alloy 690 TT has been determined to be the material of choice for steam generator applications. Alloy 690 TT has been demonstrated through both laboratory testing and operational experience to provide increased corrosion resistance compared to Alloy 600, with regard to primary water stress corrosion cracking (PWSCC) and outer diameter stress corrosion cracking. No steam generator tube degradation due to SCC has been known to occur using Alloy 690 TT tube material.

Regulatory Guide 1.121 "*Bases for Plugging Degraded PWR Steam Generator Tubes*" requires an analysis be completed to establish the minimum acceptable tube wall thickness (structural limit) and the unacceptable defects (plugging limit). Additionally, an analysis is required to substantiate the acceptability of the Tech Spec limit for steam generator leakage. These analyses were completed for the ANO-2 replacement steam generators using the standard Westinghouse methodology that confirms the 40% tube plugging limit. The analysis also confirms the acceptability of the 150 gpd leakage limit. The defined structural limit was determined to be 57.5% allowable defect depth as noted below. A 15% allowance for NDE uncertainty and growth between inspections was established, resulting in a value that bounds the 40% value listed in the ANO-2 Technical Specifications. The leakage rate that corresponds to burst at main steam line break (MSLB) pressure differential (ΔP) was determined to be 403 gpd. This information was provided to the NRC along with a copy of the Westinghouse report in non-proprietary correspondence dated June 29, 2000 (Reference 2).

The primary function of the tube-to-tubesheet joint is to provide a structurally sound and leak tight barrier between the primary and secondary sides of the steam generator. The new SG tubes are hydraulically expanded full depth through the thickness of the tubesheet to minimize the tube-to-tubesheet crevice while locating the transition as close as practical to the top of the tubesheet. The hydraulic expansion process is preferred to other techniques based on the analysis of test results and experience. Hydraulic expansion provides the best optimization of tight dimensional control and minimum residual stresses. The new SG tubes are supported by eight tube support plates of the flat-contact broached trifoil tube hole design. All tube supports are made of Type 405 ferritic stainless steel. The broached tube support plates are designed 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. Materials are selected to prevent the potential for tube denting due to tube support plate corrosion. Tube support plate material is selected to prevent the potential for tube denting due to tube support plate corrosion, while maintaining acceptable wear characteristics.

Strict tube ovality control is implemented during manufacture to limit dimensional variability of the RSG tubes in the U-bend region. Five sets of Type 405 SS anti-vibration bars (AVBs) are assembled in the U-bend region of the tubes to provide support in the regions potentially susceptible to degradation due to U-bend vibration and wear. The thickness of the five sets of U-bend AVBs is also tightly controlled. The AVBs in the U-bend provide sufficient support so that all the tubes remain elastically stable even if it is assumed that some of the support points are inactive. The AVBs in adjacent columns are inserted to different depths to minimize the U-bend pressure drop and to discourage the formation of flow stagnation regions. The AVBs are nearly perpendicular to the centerline of the tubes at all five locations in the U-bend region to provide support without unnecessary tube contact. These features of the U-bend support system provide significant margin against flow stagnation, corrosion, and tube vibration.

Reduced Potential for New Steam Generator Tube Failure Mechanisms

Alloy 690 tubes and the RSG tube bundle design are different than the OSG. The use of Alloy 690 provides high confidence that there are no new damage mechanisms since Alloy 690 tubing has been the material of choice for replacement steam generators. Field experience with Alloy 690 tubing has been excellent, with little degradation identified for the tubing except for wear or loose parts.

Loose parts are a potential concern following any construction and installation program. Westinghouse took the necessary clean room conditions and procedural controls on components utilized in the RSGs final assembly to minimize the chance of loose parts being fabricated into the RSGs. Although actions have been taken to minimize loose parts, loose parts are difficult to completely eliminate. It has been recognized that loose parts can result in damage to tubes on original steam generators, as well as steam generator replacement installations at other sites. Wear damage resulting from a loose part will typically be found in the following outage after replacement and will be detected by eddy current inspections.

Under NEI 97-06, (Steam Generator Management Program) utilities are required to develop management programs that include primary and secondary side chemistry control, primary to secondary leakage monitoring, testing and repair of the steam generators. These programs have been adopted and plant procedural controls are in place on ANO-2 to mitigate degradation of the tubing. Secondary side parameters such as metals transport, contaminant ingress, pH and Oxygen control are monitored in accordance with the applicable EPRI guideline. The Alloy 690 TT tubing material was designed to be resistant to corrosion in an operating steam generator environment. Alloy 690 tubing has been in service for greater than 12 years without any corrosion based degradation.

2R15 Steam Generator Outage Inspection

The 2R15 steam generator inspection (spring of 2002) was the first in-service inspection following replacement of the generators in 2R14. The industry guidelines require a 100% bobbin inspection. Westinghouse was the vendor who performed the inspection. A Westinghouse manway mounted manipulator was used in each channel head with dual guide tubes. All analyses were performed onsite. Table 1 lists the inspection scope of the baseline examination and Table 2 contains the detected indications that were identified.

Table 1
2R15 Eddy Current (ECT) Inspection Scope

ECT Examination Type	Inspections Conducted		% Scope
	A	B	
Bobbin	10637	10636 Note 1	100
U-bend Plus Point 1&2	181	181	100
Special Interest	128	143	

Note 1 One tube was plugged during fabrication due to an equipment failure. An inconel-690 welded plug was installed in both ends of the tube.

Table 2
2R15 Inspection Results

	Generator	
Damage Mechanism	A	B
Tube Dings Note 2	1017	596
Tube Dents	0	0
Manufacturing Buff Marks (MBMs)	139	216
AVB Wear	0	1 Note 3
Loose Part Impact	0	2 Note 4

Note 2 A volumetric indentation of the tube wall due to fabrication processes

Note 3 There was one AVB wear identified

Note 4 One loose part that was removed which affected two tubes

There were no degradation mechanisms other than minor wear identified during the 2R15 outage. A loose part, approximately 12 inches above the Cold Leg tubesheet in the "B" SG, was lodged between two tubes. The object was identified as a small corkscrew shaped metal shaving, which was likely, a result of the replacement SG activities during the previous outage. The part was removed and the impacted tubes were further analyzed and determined to have no significant wear as noted below. There were no tubes repaired during the 2R15 inspection and there were no in-situ tests performed during the baseline examination.

Therefore, based on the 2R15 inspection results, the ANO-2 steam generators were clearly Category C-1.

2R15 Condition Monitoring Assessment

The purpose of this condition monitoring evaluation is to evaluate the 2R15 RSG tubing inspection results and ensure the performance criteria contained in the Nuclear Energy Institute Steam Generator Program Guidelines (NEI 97-06) are satisfied. This includes evaluating any detected degradation from a structural and leakage standpoint. A comprehensive eddy current examination was performed on the RSGs to look for various degradation mechanisms. The techniques used were qualified per the EPRI PWR Steam Generator Examination Guidelines.

The only form of in-service degradation identified was mechanical wear at the AVBs and wear on two tubes from a loose part. The depth of the wear indications was minimal (maximum of 18% throughwall) which is well below the limits for leakage and burst. The number of MBMs and dings were also bounded by the operational assessment and degradation assessment. The conclusion of this condition monitoring evaluation is that none of the performance criteria in NEI-97-06 was exceeded and based on the comprehensive exams performed it is reasonable to conclude that the end of cycle (EOC) condition would not exceed the performance criteria. The combined EOC accident induced leakage value was zero.

Cycle 16 Operational Assessment

Based on the results of ECT examinations and analysis described in this report, detection capabilities and growth rates are quantifiable and therefore, future steam generator tube performance can be evaluated. The 2R15 Operational Assessment evaluated two full-cycles of operation. A run time of 3.0 effective full power years was used to determine the end of cycle conditions. The steam generators were evaluated to be safe to operate until the next proposed inspection (2R17) in the spring of 2005. All performance criteria were shown to be maintained with added margin.

The only existing damage mechanism that impacts the operational assessment is wear. The anticipated number of wear indications is expected to increase; however the extent of wear is not expected to increase. Similar designed generators such as at DC Cook, Farley, and South Texas have not experienced large degrees of wear to date.

A conservative factor of ten was applied for the predicted number of indications that could possibly develop. The calculated leakage at main steam line break conditions is zero. Since wear at the AVBs is typically self-limiting, the calculational results are considered conservative. It was determined that the ANO-2 SGs meet their design basis and operational leakage assessment until the 2R17 inspection in the spring of 2005.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

General Design Criteria (GDC)

GDC 32 (Inspection Of Reactor Coolant Pressure Boundary) states that *"Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel."* The proposed change only impacts the frequency of inspection of the steam generator pressure boundary and does not affect the ability of the boundary to be inspected.

GDCs 14, (Reactor Coolant Pressure Boundary), and 31 (Fracture Prevention of Reactor Coolant Pressure Boundary) are similarly unaffected by the proposed change.

Regulatory Guides (R.G.)

Regulatory Guide 1.83 (Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes) paragraph C.6.d states that *"if two consecutive inspections, not including the preservice inspection, result in less than 10% of the tubes with detectable wall penetration (20%) and no significant (10%) further penetration of tubes with previous indications, the inspection frequency should be extended to 40-month intervals."* In addition, *"if it can be demonstrated through two consecutive inspections that previously observed degradation has not continued and no additional degradation has occurred, a 40-month inspection interval may be initiated."* This regulatory position is being impacted in that Entergy proposes that after replacement of the ANO-2 SGs only one inspection after the initial inservice inspection is necessary for extending the inspection interval. The impact to the discussion of the regulatory guide is the same as that being requested for the ANO-2 technical specifications.

Regulatory Guide 1.121 (*Bases for Plugging Degraded PWR Steam Generator Tubes)* describes a method acceptable to the NRC Staff for establishing the limiting safe conditions of tube degradation of steam generator tubing, beyond which defective tubes as established by inservice inspection should be removed from service by welding plugs at each end of the tube. This regulatory guide is also unaffected by the proposed change.

NEI Guidance

The proposed change does not affect the ANO-2 commitment to the guidance of NEI 97-06, *Steam Generator Program Guidelines*, Rev. 1.

ANO-2 Final Safety Analysis Report (FSAR)

ANO-2 FSAR section 5.5.2.3.4, *Allowable Tube Wall Degradation*, states that the program for the inservice inspection of steam generator tubing is in conformance with the requirements of Regulatory Guide 1.83, Rev. 1 and the ASME Code, Section XI. R.G. 1.83 provides guidance on the sample size and the inspection interval. ASME Code, Section XI establishes the standards for examination and allowable flaws in tubing and is unaffected by the proposed change. Except as noted above for R.G. 1.83, ANO-2 is still in compliance with the ANO-2 FSAR.

Standard Review Plans (SRPs)

SRP 5.4.2.2 (*Steam Generator Tube Inservice Inspection)* states that the inservice inspection program for steam generator tubes, which constitute part of the reactor coolant pressure boundary, is based on the detailed positions of Regulatory Guide 1.83. In addition, this SRP states that the provisions made for baseline inspection prior to startup, the methods to be used for the inspections, and the inservice inspection program are reviewed in the final safety analysis report and plant technical specifications. ANO-2 is not an SRP plant; however, except as noted above for R.G. 1.83, the proposed TS change is still in accordance with this SRP.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any GDC differently than described in the SAR. R.G. 1.83 compliance is affected the same as that of the proposed TS change and no additional action is considered necessary.

5.2 No Significant Hazards Consideration

The proposed one-time change revises the steam generator (SG) inservice inspection frequency requirements in TS 4.4.5.3.a to allow a 40-month inspection interval after one inspection given a C-1 classification rather than after two consecutive inspections. This is considered acceptable since the steam generators and tubes are new and the 2R15 inspection resulted in category C-1. The proposed change will add a one-time allowance following the current TS 4.4.5.3.a, as follows:

> *A one-time inspection interval of a maximum of once per 40 months is allowed for the inspection performed immediately following 2R15. This is an exception to 4.4.5.3.a in that the interval extension is based on the results falling into the C-1 category after one inspection.*

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   There are no damage mechanisms that are active in the ANO-2 SGs that would prematurely create an accident or increase SG leakage. The scope of inspections performed during 2R15, the first refueling outage following SG replacement, exceeded the TS requirements for ensuring that the ANO-2 steam generator fell into the C-1 category. The ANO-2 steam generators meet the current industry examination guidelines without performing inspections during the next refueling outage. The results of the Condition Monitoring Assessment performed during 2R15 demonstrated that all performance criteria were met. The results of the 2R15 Operational Assessment show that all performance criteria are being met over the proposed operating period.

   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?

   Response: No.

   The proposed change will not alter any plant design basis or postulated accidents resulting from potential SG tube degradation. The scope of inspections performed during the 2R15 outage, the first refueling outage following steam generator replacement, exceeded the TS requirements.

   The proposed change does not affect the design of the SGs, the method of 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 previously evaluated.

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

Response: No.

Steam generator tube integrity is a function of design, environment, and current physical condition. Extending the steam generator tube inservice inspection frequency by one operating cycle will not alter their function or design. Inspections conducted prior to placing the SGs into service and inspection during the first refueling outage following SG replacement demonstrate that the SGs do not have fabrication damage or an active damage mechanism. The scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for the same model of RSGs 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 change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed amendment(s) present 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.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

Exelon Corporation proposed a one-time change to the TS to revise the SG inspection frequency requirements in TS 5.5.9.d.2 for the Braidwood Unit I Fall 2001 refueling outage. The change allowed a 40-month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in a C-1 classification. The NRC approved the Exelon request in a safety evaluation dated August 9, 2001.

In addition, South Texas Project Nuclear Operating Company proposed a license amendment for the South Texas Project on January 28, 2002. Their request to amend Operating Licenses NPF-76 and NPF-80 included a similar one-time change to revise the steam generator inservice inspection frequency requirements.

The proposed changes by Entergy are similar to those above.

Attachment 2

2CAN110204

Proposed Technical Specification Changes (mark-up)

REACTOR COOLANT SYSTEM

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 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.

   A one-time inspection interval of a maximum of once per 40 months is allowed for the inspection performed immediately following the 2R15 outage. This is an exception to 4.4.5.3.a in that the interval extension is based on all of the results of one inspection falling into the C-1 category.

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 into 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 4.4.5.3.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 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   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.