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1CAN090401

September 30, 2004

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

SUBJECT: License Amendment Request
Proposed Technical Specification Change for Revision to ANO-1 Steam Generator Tube Inservice Inspection Program
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the Operating License amendment for Arkansas Nuclear One, Unit 1 (ANO-1) to replace the existing steam generator tube surveillance program with that being proposed by the Technical Specification Task Force in TSTF 449, Draft Revision 2 as discussed in Attachment 1. The proposed change in Attachments 2 and 3 revises the Technical Specifications (TS) and associated Bases for Specification 3.4.13, *RCS Operational LEAKAGE*, Specification 5.5.9, *Steam Generator Tube Surveillance Program,* and Specification 5.6.7, *Steam Generator Tube Surveillance Reports*, and adds a new TS 3.4.16 entitled *Steam Generator (SG) Tube Integrity*. In addition, Entergy is replacing the ANO-1 steam generators in refueling outage 1R19 which is scheduled to commence in the fall of 2005. As a result, Entergy is revising the current TSs and Bases to reflect the new Alloy 690 thermally treated tubing design. Both the TSs and Bases are being provided for NRC review and approval.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards consideration. The basis for these determinations is included in Attachment 1. The proposed change includes new commitments as identified in Attachment 4.

Entergy requests approval of the proposed amendment by August 1, 2005. Once approved, the amendment shall be implemented prior to resumption of operation from the 1R19 refueling outage scheduled for the fall of 2005.

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

A047

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 30, 2004.

Sincerely,

for Jeff Forbes

JSF/sab

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Proposed Technical Specification Bases Changes (mark-up)
4. List of Regulatory Commitments

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Attachment 1

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Analysis of Proposed Technical Specification Change

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

The proposed change revises the Technical Specifications (TS) for Arkansas Nuclear One, Unit 1 (ANO-1) related to steam generator inspections. The reason for the proposed changes is twofold. The first is that ANO-1 will be replacing the existing steam generators (SGs) in refueling outage 1R19 (scheduled to commence in the fall of 2005) with replacement SGs having thermally treated Alloy 690 SG tubes. Second, Entergy is applying the draft Revision 2 of Technical Specification Task Force (TSTF) 449 format and content which is being finalized for industry application.

Specifically, Entergy will be revising ANO-1 Specification 3.4.13, *RCS Operational LEAKAGE*, Specification 5.5.9, *Steam Generator Tube Surveillance Program,* and Specification 5.6.7, *Steam Generator Tube Surveillance Reports*, and adding a new specification 3.4.16 for *Steam (SG) Generator Tube Integrity.*

2.0 PROPOSED CHANGE

The proposed change revises TS 3.4.13, *RCS Operational LEAKAGE*, by removing the 4 hour Action Statement when reactor coolant system (RCS) primary to secondary LEAKAGE is not within limits. The primary to secondary LEAKAGE is being revised to conservatively restrict the Completion Time to an immediate shutdown. The Completion Time for RCS unidentified or identified LEAKAGE not within limits, is not being proposed for change and will retain the ANO-1 currently licensed Completion Time of 18 hours. The Action Statements are renumbered accordingly.

Surveillance Requirement (SR) 3.4.13.2 is being changed from verifying SG tube integrity to requiring verification that primary to secondary LEAKAGE is within limit. Steam generator tube integrity is verified under a new Limiting Condition for Operation (LCO) in TS 3.4.16. A new Note is being added to surveillance requirement (SR) 3.4.13.1 to indicate that this surveillance is not applicable to primary to secondary LEAKAGE. A Note is being added to SR 3.4.13.2 stating that the SR is not required to be performed until 12 hours after establishment of stable plant conditions. This is consistent with the existing Note for SR 3.4.13.1. Due to the level of detail that is represented by implementation of the new and revised TS changes, NRC review of the associated Bases to this Amendment Request is requested. The Bases sections of TSs 3.4.5, 3.4.6, and 3.4.7 reference the *Steam Generator Tube Surveillance Program.* Since these TSs and Bases are not affected except by the reference to this program, these Bases will be revised under the ANO-1 Bases Control Program.

The proposed change adds a new Technical Specification 3.4.16 entitled *Steam Generator (SG) Tube Integrity.* The proposed Specification requires that SG tube integrity be maintained and requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the *Steam Generator Program.* The revised Table of Contents is attached for completeness and is proposed to be issued with the NRC Operating License Amendment.

The proposed change revises TS 5.5.9, *Steam Generator Tube Surveillance Program (*now changed to *Steam Generator Program)* to reflect the Steam Generator Program which describes SG tube integrity maintenance, SG condition monitoring, performance criteria, and

inspection intervals. The current Alternate Repair Criteria applicable to the existing SGs will be eliminated.

The proposed change to TS 5.6.7, *Steam Generator Tube Surveillance Reports (*now changed to *Steam Generator Tube Inspection Program)*, provides the revised requirements and contents of the SG tube inspection report. The reporting requirements are revised to require a report within 180 days of initial entry into MODE 4 following a steam generator inspection.

Except as noted above, changes being proposed herein are in accordance with Draft Revision 2 to TSTF 449.

3.0 BACKGROUND

3.1 Replacement Steam Generators

ANO-1 currently utilizes two steam generators that were supplied as a part of the original nuclear steam supply system by Babcock and Wilcox. The ANO-1 SGs are straight-tube, vertical, counter-flow, once-through heat exchangers with shell-side boiling of secondary fluid. Primary fluid from the reactor enters through an inlet nozzle in the top head, flows down through the tubes, is collected in the bottom head, and exits through two primary outlet nozzles. The use of straight tubes results in almost pure counter-flow properties.

The Replacement SGs (RSGs) are being manufactured by Framatome ANP (AREVA) in Chalon, France. The RSGs are being designed, manufactured, and tested in accordance with the 1989 Edition, no Addenda, of Section III of the ASME Code (Alloy 690 tubing is in accordance with ASME Boiler & Pressure Vessel Code 1998 Edition, 2000 Addenda). The design, procurement, and manufacturing process are performed under a Quality Assurance Program that complies with the requirements of Appendix B to 10 CFR 50 and with the current NRC requirements that relate to steam generator design. Significant design improvements include (1) use of thermally treated Alloy 690 tube material with full depth tube sheet expansion, (2) addition of an integral flow restrictor in each main steam nozzle, (3) slightly greater secondary side volume due to thinner shell, (4) more corrosion resistant material for the tube support plates and (5) slightly higher secondary side pressure and temperature rating. The replacement steam generators occupy essentially the same physical space as the original steam generators and are very similar in thermal and hydraulic performance to the original SGs. Therefore, the RSGs are similar to the original SGs and will be replaced under the requirements of 10 CFR 50.59. The following is a comparison of the original SGs to the RSG preliminary design.

Parameter	Original SGs	RSGs
Number of Tubes per SG	15,531	15,597
Tube Outside Diameter, inches	0.625	0.625
Length of Tube Expansion, Inches	1 (minimum)	24 (nominal)
Tube Wall Thickness, inches	0.037	0.037

Parameter	Original SGs	RSGs
Tube Active Heat Transfer Surface Area per SG (Secondary), sq. ft.	132,400	133,000
Tube Material	SB-163 Alloy 600 stress relieved	SB-163 Alloy 690 TT
Primary Side Volume per SG, cu. ft.	2030	2026
Secondary Side Volume per SG, cu. ft.	3264	3666
Primary Side Design Pressure, psig	2500	2500
Secondary Side Design Pressure, psig	1050	1150 (Secondary side steam line design pressure is unchanged @ 1050 psig).
Primary Side Design Temperature, °F	650	650
Secondary Side Design Temperature, °F	600	605
Shell Material	SA 516 Gr. 70	SA 508 Cl. 3A
Tube Sheet Thickness, feet	2	2
Tube Support Plate Thickness, inches	1.5	1.18
Tube Support Plate Material	SA 515 Gr. 70	SA 240 type 410

3.2 Steam Generator Program

For the last several years, the industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP), has developed a generic approach to improving SG performance referred to as *Steam Generator Degradation Specific Management* (SGDSM). Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG Program as provided in References 2 through 7.

These EPRI Guidelines, along with NEI 97-06, tie the elements of the *Steam Generator Program* in TS 5.5.9 together, while defining a comprehensive, performance based approach to managing SG performance. In parallel with the industry efforts, the NRC pursued resolution of SG performance issues. In December of 1998, the NRC Staff acknowledged that the *Steam Generator Program* described by NEI 97-06 and its referenced EPRI Guidelines provide an acceptable starting point to use in the resolution of differences between it and the staff's proposed Generic Letter and draft Regulatory Guide (DG-1074). In 2004, the NRC Staff and NEI have reached mutual agreement to the scope of what is to be Revision 2 to TSTF 449.

4.0 TECHNICAL ANALYSIS

The proposed TS format is not directly affected by the design of the RSGs, except for the extended inservice inspection periods for thermally treated Alloy 690 tubing and elimination of the alternate repair criteria (ARC). The primary coolant activity limit and its assumptions are not affected by the proposed changes to the standard technical specifications. The proposed changes are an improvement to the existing steam generator inspection requirements and provide additional assurance that the plant licensing basis will be maintained between steam generator inspections. Alternate repair criteria and sleeve inspections not applicable to the replacement steam generator are being removed.

For design basis accident for the main steam line break (MSLB) the SG tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). The offsite dose analyses assume a primary to secondary LEAKAGE for the SGs of 1 gallon per minute (bounding for that resulting from accident induced stresses and pressure differential). For accidents that do not involve fuel damage, the reactor coolant activity levels are conservatively assumed to be at the technical specification limit or consistent with 1% failed fuel. For accidents that do involve fuel damage, the primary coolant activity values are a function of the amount of activity released from the damaged fuel.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. However, the primary to secondary LEAKAGE rate is relatively inconsequential for the SGTR analysis.

The consequences of these design basis accidents are, in part, functions of the radioactivity levels in the primary coolant and the accident primary to secondary LEAKAGE rates. As a result, limits are included in the plant technical specifications for operational LEAKAGE and for DOSE EQUIVALENT I-131 in primary coolant to ensure the plant is operated within its analyzed condition.

The current ANO-1 limit of 150 gallons per day of primary to secondary LEAKAGE through any one SG is based on operating experience as an indication of one or more tube leaks. This LEAKAGE limit provides additional assurance that leaking flaws will not propagate to burst prior to plant shutdown and will remain within the ANO-1 safety analysis assumptions. Therefore, the existing ANO-1 TS LCO for primary to secondary LEAKAGE is consistent with TSTF 449 and no change is being proposed.

The other technical specification changes proposed are, in general, an improvement over current requirements. They replace the current prescriptive technical specification with one that references the *Steam Generator Program* requirements which incorporate the latest knowledge of SG tube degradation morphologies and the techniques developed to manage them.

The requirements being proposed are more effective in detecting SG degradation and prescribing corrective actions than those of the current technical specifications. As a result, these proposed changes will result in added assurance of the function and integrity of SG tubes. The sections below provide the technical justification for the proposed changes.

Operational LEAKAGE Actions

If primary to secondary LEAKAGE exceeds 150 gallons per day through any one SG, a plant shutdown is to be commenced. MODE 3 must be achieved in 6 hours and MODE 5 in 36 hours. The existing technical specifications allow 4 hours to reduce primary to secondary LEAKAGE to less than the limit. The proposed technical specification removes this allowance. This proposed change is consistent with TSTF 449.

RCS Operational LEAKAGE Determined by Water Inventory Balance

The proposed change adds a second Note to SR 3.4.13.1 that makes the water inventory balance method not applicable to determining primary to secondary LEAKAGE. RCS water inventory balance is not the most appropriate means for determining primary to secondary LEAKAGE as low as 150 gallons per day. This change is necessary to make the surveillance requirement appropriate for the proposed LCO.

Steam Generator Tube Integrity Verification

The current SR 3.4.13.2 requires verification of tube integrity in accordance with the *SG Tube Surveillance Program*. This surveillance is no longer appropriate since tube integrity is addressed through the addition of a new *SG Tube Integrity* specification per TS 3.4.16. Surveillance Requirement 3.4.13.2 has been changed to verify the LCO requirement on primary to secondary LEAKAGE only. Steam generator tube integrity is verified in accordance with an SR in the SG Tube Integrity Specification.

Determination of the primary to secondary LEAKAGE is required every 72 hours. The SR is modified by a Note stating the SR is not required to be performed until 12 hours after establishment of stable operating conditions. Monitoring of primary to secondary LEAKAGE is also required by the *Steam Generator Program* in TS 5.5.9 based upon guidance provided in Reference 5.

Frequency of Verification of Steam Generator Tube Integrity

The current technical specifications contain prescriptive inspection intervals which depend on the condition of the tubes as determined by the last SG inspection. The minimum inspection interval is no less than 12 and no more than 24 months unless the results of two consecutive inspections are in the best category (no additional degradation), and then the interval can be extended to 40 months.

The surveillance Frequency in the proposed *Steam Generator Tube Integrity* specification is governed by the requirements in the *Steam Generator Program* in TS 5.5.9 and specifically by References 2 and 3. The proposed Frequency is also prescriptive, but has a stronger engineering basis than the existing technical specification requirements. The interval is dependent on tubing material and whether any active degradation has been identified in previous inspections. The interval is limited by existing and potential degradation mechanisms and their anticipated growth rate.

Since ANO-1 will be replacing the SGs with thermally treated Alloy 690 tubing in the next refueling outage, a longer surveillance Frequency between inspections is appropriate. The maximum inspection interval for Alloy 690 thermally treated tubing is 100% of the tubes at

sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected. The new inspection intervals are established in Specification 5.5.9.

Taken in total, the proposed inspection intervals provide a greater margin of safety than the current requirements because they are based on an engineering evaluation of the tubing condition and potential degradation mechanisms and growth rates, not only on the previous inspection results. As an added safety measure, the *Steam Generator Program* in TS 5.5.9 requires a minimum sample size at each inspection that is significantly larger than that required by current technical specifications (20 percent versus 3 percent times the number of SGs in the plant); thus providing added assurance that any degradation within the SGs will be detected and accounted for in establishing the inspection interval. The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. There are no known instances of cracking in thermally treated 690 tubes in either the U.S. or international SGs.

Steam Generator Tube Sample Selection

The current technical specifications base tube selection on SG conditions and industry and plant experience. The minimum sample size is 3% of the tubes times the number of SGs in the plant. The proposed change refers to the *Steam Generator Program* in TS 5.5.9 degradation assessment guidance for sampling requirements. The minimum sample size per the *Steam Generator Program* is 20% of the number of available tubes.

The *Steam Generator Program* requires the preparation of a degradation assessment before every SG inspection. The degradation assessment is the key document used for planning a SG inspection, where inspection plans and related actions are determined, documented, and communicated prior to the outage. The degradation assessment addresses the various reactor coolant pressure boundary components within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary.) In a degradation assessment, tube sample selection is performance based and is dependent upon actual SG conditions and plant operational experience and of the industry in general. Existing and potential degradation mechanisms and their locations are evaluated to determine which tubes will be inspected. Tube sample selection is adjusted to minimize the possibility that tube integrity might degrade during an operating cycle beyond the limits defined by the performance criteria. The EPRI *Steam Generator Examination Guidelines* and EPRI *Steam Generator Integrity Assessment Guideline* provide guidance on degradation assessment.

In general, the sample selection considerations required by the current technical specifications and the requirements in the *Steam Generator Program* as proposed by this change are consistent; however, the *Steam Generator Program* provides more guidance on selection methodologies and incorporation of industry experience and requires more extensive documentation of the results. Therefore, the sample selection method proposed by this change is more conservative than the current technical specification requirements. In addition, the minimum sample size in the proposed requirements is larger.

Steam Generator Inspection Techniques

The *Steam Generator Program* in TS 5.5.9 requires the performance of a degradation assessment before every SG inspection and refers utilities to EPRI *Steam Generator Examination Guideline* and EPRI *Steam Generator Integrity Assessment Guideline* for guidance on its performance. The degradation assessment will identify current and potential new degradation locations and mechanisms and non destructive examination (NDE) techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that might progress during an operating cycle beyond the limits defined by the performance criteria. Since ANO-1 will be beginning Cycle 20 with new SGs and thermally treated Alloy 690 tubing, there will be no active degradation mechanisms present and special inspection techniques are not planned.

Steam Generator Inspection Scope

The current technical specifications include a definition of Tube Inspection that requires an inspection of the steam generator tube from the point of entry completely to the point of exit. This definition is overly prescriptive and simplistic and has led to interpretation questions in the past.

The *Steam Generator Program* in TS 5.5.9 states:

> *The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that OTSG tube integrity is maintained until the next OTSG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.*

The *Steam Generator Program* provides guidance and a defined process as part of the degradation assessment for determining the extent of a tube inspection. This guidance takes into account industry and plant specific history to determine potential degradation mechanisms and the location that they might occur within the SG. This information is used to define a performance based inspection scope targeted on plant specific conditions and SG design.

The proposed change is an improvement over the current ANO technical specifications because it focuses the inspection effort on the areas of concern, thereby minimizing the unnecessary data that the NDE analyst must review to identify indication of tube degradation.

Steam Generator Performance Criteria

The proposed change in TS 5.5.9 adds the performance-based *Steam Generator Program* to the Technical Specifications. A performance-based approach has the following attributes:

- measurable parameters,
- objective criteria to assess performance based on risk-insights,
- deterministic analysis and/or performance history, and
- licensee flexibility to determine how to meet established performance criteria.

The performance criteria used for SGs are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. The structural integrity and accident induced leakage criteria were developed deterministically and are consistent with the plant's licensing basis. The operational LEAKAGE criterion was based on providing added assurance against tube rupture at normal operating and faulted conditions. The proposed structural integrity and accident induced leakage performance criteria are new requirements. These performance criteria are provided in Specification 5.5.9. The requirements and methodologies will be established to meet the performance criteria are documented in the *Steam Generator Program*. The current technical specifications contain only the operational LEAKAGE criterion; therefore the proposed change is more conservative than the current requirements.

The SG performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary (RCPB) integrity throughout each operating cycle. The structural integrity performance criterion is based on providing reasonable assurance that an SG tube will not burst during normal operation or postulated accident conditions. Adjustments to include contributing loads are addressed in the applicable EPRI guidelines.

Normal steady state full power operation is defined as the conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. The definition of normal steady state full power operation is important as it relates to application of the safety factor of three in the structural integrity performance criterion. The criterion requires "...retaining a safety factor of 3.0 under normal steady state full power operation primary to secondary pressure differential...". The application of the safety factor of three to normal steady state full power operation is founded on past NRC positions, accepted industry practice, and the intent of the ASME Code for original design and evaluation of inservice components. The assumption of normal steady state full power operating pressure differential has been consistently used in the analysis, testing and verification of tubes with stress corrosion cracking for verifying a safety factor of three against burst. The revised TS requirements under TSTF 449, Draft Revision 2 require those loads that significantly affect burst or collapse be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads. These loads, as well as the other analyses to support a 40% plugging limit, will be analyzed for the RSGs under the existing ANO-1 licensing basis. These analyses will be performed and documented under the requirements of 10 CFR 50.59 prior to startup from the 1R19 refueling outage.

The actual operational parameters may differ between cycles. As a result of changes to these parameters, reaching the differential pressure in the equipment specification may not be possible during plant operations. Evaluating to the pressure in the design or equipment specification in these cases would be an unnecessary conservatism. Therefore, the definition allows adjustment of the $3\Delta P$ limit for changes in these parameters when necessary. Further guidance on this adjustment is provided in Appendix M of the EPRI *Steam Generator Integrity Assessment Guideline*.

Primary to secondary LEAKAGE is a factor in the activity releases outside containment resulting from a limiting design basis accident. The potential dose consequences from primary to secondary LEAKAGE during postulated design basis accidents must not exceed the radiological limits imposed by 10 CFR Part 100 guidelines, or the radiological limits to control

room personnel imposed by General Design Criterion (GDC)-19, or other NRC approved licensing basis. The operational LEAKAGE performance criterion is:

> *The RCS operational primary to secondary LEAKAGE through any one steam generator shall be limited to 150 gallons per day.*

Plant shutdown will commence if primary to secondary LEAKAGE exceeds 150 gallons per day at room temperature conditions from any one SG. The operational LEAKAGE performance criterion is documented in the *Steam Generator Program* of TS 5.5.9 and implemented in Specification 3.4.13, *RCS Operational LEAKAGE.*

Specification 5.5.9 contains the performance criteria and is more conservative than the current technical specifications. The current technical specifications do not address the structural integrity and accident induced leakage criteria.

Steam Generator Repair Criteria

Repair criteria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging. Tube repair criteria are established for each active degradation mechanism. Tube repair criteria are either the standard through-wall depth-based criterion (e.g., 40% through-wall for most plants) or through-wall depth based criteria for repair techniques approved by the NRC, or other ARC approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05 (Ref. 11). Entergy is not proposing any ARC or sleeving repairs for the replacement SGs. The existing SG ARC and sleeving allowance are being eliminated.

Actions

The *RCS Operational LEAKAGE* and *Steam Generator Tube Integrity* specifications require the licensee to monitor SG performance against performance criteria in accordance with the *Steam Generator Program.* During plant operation, monitoring is performed using the operational LEAKAGE criterion. Exceeding that criterion will lead to a plant shutdown in accordance with TS 3.4.13. Once shutdown, the *Steam Generator Program* will ensure that the cause of the operational LEAKAGE is determined and corrective actions are taken to prevent recurrence. Operation may resume when the requirements of the *Steam Generator Program* have been met. This requirement is unchanged from the current technical specifications.

Also during plant operation the licensee may discover an error or omission that indicates a failure to implement a required plugging or repair during a previous SG inspection. Under these circumstances, the licensee is expected to take the actions required by Condition A in the *Steam Generator Tube Integrity* specification. If a performance criterion has been exceeded, a principal safety barrier has been challenged and 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) require NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence. The *Steam Generator Program* in TS 5.5.9 additionally requires that the report contain information on the performance criteria exceeded and the basis for the planned operating cycle. The current technical specifications only address operational LEAKAGE during operations.

During MODES 5 and 6, the operational LEAKAGE criterion is not applicable, and the SGs will be inspected as required by the surveillance in the *Steam Generator Tube Integrity*

specification. A condition monitoring assessment of the "as found" condition of the SG tubes will be performed to determine the condition of the SGs with respect to the structural integrity and accident leakage performance criteria. If the performance criteria are not met, the *Steam Generator Program* requires ascertaining the cause and determining corrective actions to prevent recurrence. Operation may resume when the requirements of the *Steam Generator Program* have been met.

The current technical specifications do not address ACTIONS required while operating if it is discovered that the structural integrity or accident induced leakage performance criteria or a repair criterion are exceeded, so the proposed change is conservative with respect to the current technical specifications. If performance or repair criteria are exceeded while shutdown, the affected tubes must be plugged. A report will be submitted to the NRC in accordance with Technical Specification 5.6.7.

Reporting Requirements

The current technical specifications require that a steam generator tube inservice inspection report be submitted to the NRC within 90 days of inspection completion, and if the results of steam generator tube inspections fall into Category C-3, another report is to be submitted to the NRC prior to resumption of plant operation. The proposed change to TS 5.6.7 replaces these reports with one report required within 180 days. The proposed report will contain comprehensive inspection results which have similar information to that provided in the 90 day report.

Steam Generator Terminology

The proposed TS 3.4.16 *Steam Generator Tube Integrity* Bases explain a number of terms that are important to the function of the *Steam Generator Program*. The terms are described below.

Accident Induced Leakage Rate means:

> *The primary to secondary LEAKAGE rate occurring during postulated accidents other than a steam generator tube rupture.*

This includes the primary to secondary LEAKAGE rate existing immediately prior to the accident plus additional primary to secondary LEAKAGE induced during the accident. *Primary to Secondary LEAKAGE* is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary to secondary leak rate during postulated design basis accidents must not cause radiological dose consequences in excess of the 10 CFR 100 guidelines for offsite doses, or the GDC-19 requirements for control room personnel, or other NRC approved licensing basis.

The term *Burst* is defined as:

> *The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.*

Since a burst definition is required for condition monitoring, a definition that can be analytically defined and is capable of being assessed via in situ and laboratory testing is necessary. Furthermore, the definition must be consistent with ASME Code requirements, and apply to most forms of tube degradation. The definition developed for tube burst is consistent with the

testimony of James Knight (Ref. 8), and the historical guidance of draft Regulatory Guide 1.121 (Ref. 9). The definition of burst per these documents is in relation to gross failure of the pressure boundary; e.g., "the degree of loading required to burst or collapse a tube wall is consistent with the design margins in Section III of the ASME B&PV Code" (Ref. 10).

The term *Collapse* is defined as

> *The load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero.*

In dealing with pure pressure loadings, burst is the only failure mechanism of interest. If bending loads are introduced in combination with pressure loading, the definition of failure must be broadened to encompass both burst and bending collapse. Which failure mode applies depends on the relative magnitude of the pressure and bending loads and also on the nature of any flaws that may be present in the tube.

SG *Tube* is defined as,

> *The entire length of the tube, including the tube wall and any repairs to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.*

This definition ensures that all portions of SG tubes that are part of the RCPB, with the exception of the tube-to-tubesheet weld, are subject to *SG* tube integrity inspection requirements. The definition is also intended to exclude tube ends that can not be NDE inspected by eddy current. If there are concerns in the area of the tube end, they will be addressed by NDE techniques if possible or by using other methods if necessary. For the purposes of SG tube integrity inspection, any weld metal in the area of the tube end is not considered part of the tube. This is necessary since the acceptance requirements for tubing and weld metals are different.

The LCO section of *Steam Generator Tube Integrity* Bases defines the term "collapse" as "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." In dealing with pure pressure loadings, burst is the only failure mechanism of interest. If bending loads are introduced in combination with pressure loading, the definition of failure must be broadened to encompass both burst and bending collapse. Which failure mode applies depends on the relative magnitude of the pressure and bending loads and also on the nature of any flaws that may be present in the tube. Guidance on assessing applicable failure modes is provided in the EPRI steam generator guidelines.

The LCO section of Steam Generator Tube Integrity Bases define the term "significant" as used in the structural integrity performance criterion as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established."

The LCO section of Steam Generator Tube Integrity Bases describes how to determine whether thermal loads are primary or secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The regulatory requirements applicable to SG tube integrity are the following:

5.1.1 10 CFR 50.55a, Codes and Standards - Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section. The proposed change and the *Steam Generator Program* requirements which underlie it are in full compliance with the ASME Code. The proposed technical specifications are more effective at ensuring tube integrity and, therefore, compliance with the ASME Code, than the current technical specifications as described in Section 4.0 (Technical Analysis).

5.1.2 10 CFR 50.65 Maintenance Rule – Each holder of a license to operate a nuclear power plant under 10 CFR 50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions.

Under the Maintenance Rule, ANO-1 classifies the SGs as risk significant components because they are relied on to remain functional during and after design basis events. The performance criteria included in the proposed technical specifications are used to demonstrate that the condition of the SGs is being effectively controlled through the performance of appropriate preventive maintenance (Maintenance Rule §(a)(2)). If the performance criteria are not met, a determination of appropriate depth is done and the results evaluated to determine if goals should be established per 10 CFR 50.65(a)(1) of the Maintenance Rule.

5.1.3 10 CFR 50, Appendix A General Design Criteria,

GDC 14 – Reactor Coolant Pressure Boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture. The discussion provided in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

GDC 30 - Quality of Reactor Coolant Pressure Boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. There are no changes to the SG design that impact this general design criterion. The discussion provided in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

GDC 32 – Inspection of Reactor Coolant Pressure Boundary. Components which are part of the reactor coolant pressure boundary shall be designed to (1) allow periodic inspection and testing of important areas and features to assess their structural and leak tight integrity, and (2) include an appropriate material surveillance program for the reactor pressure vessel. There are no changes to the SG design that impact this general design criterion. The discussion provided in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

5.1.4 ANO-1 Safety Analysis Report (SAR) – The ANO-1 SAR was reviewed to determine whether a change in the ANO-1 TSs would impact the licensing basis. The following ANO-1 SAR sections were noted:

§4.3.4.3 Steam Generator Tube Surveillance states:

The steam generator tubes are examined periodically under a systematic surveillance program in accordance with Regulatory Guide 1.83, Revision 1. This program is fully described in Technical Specification 5.5.9.

The revision to the ANO-1 Technical Specifications impacts this commitment in that Entergy has established our SG Integrity Program in accordance with NEI 97-06. Entergy will revise Section 4.3.4.3 of the ANO-1 SAR to reference NEI 97-06.

§16.2.20 Steam Generator Integrity states:

Steam Generator Integrity Program ensures the steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The Steam Generator Integrity Program applies to the steam generator internals, tubing, and associated repair techniques and components, such as plugs and sleeves. The aging effects addressed by the Steam Generator Integrity Program are loss of material, cracking, and fouling.

A revision to the ANO-1 SAR will be made to reference the Steam Generator Program TS. Even though the ANO-1 replacement SGs will not have active degradation mechanisms, the ANO-1 Steam Generator Program will still address repair techniques, if realized in the future.

The replacement of the existing SGs with new SGs requires certain parameters in the ANO-1 SAR to change. These changes will be controlled under 10 CFR 50.59 as part of the SG replacement and are not being addressed herein.

5.2 No Significant Hazards Consideration

The proposed change revises the improved Arkansas Nuclear One, Unit 1 (ANO-1) Technical Specifications (TS) Section 3.4.13, *RCS Operational LEAKAGE*, Section 5.5.9, *Steam Generator Program*, and Section 5.6.7, *Steam Generator Tube Inspection Report*. The proposed change also adds a new TS 3.4.16 for *Steam Generator Tube Integrity*. The proposed changes apply draft Revision 2 of the Technical Specification Task Force (TSTF) 449 format. Entergy has evaluated whether or not a significant hazards

consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, 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

The proposed change requires a *Steam Generator Program* that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of design basis operating conditions (including startup, power operation, hot standby, cooldown, anticipated transients and postulated accidents). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. These criteria assure that the probability of an accident will not be increased.

The primary to secondary accident induced leakage rate for any design basis accidents, other than an SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. The operational LEAKAGE performance criterion meets current NRC regulations and NEI 97-06 criteria for reactor coolant system (RCS) operational primary to secondary LEAKAGE through any one SG of 150 gallons per day. These criteria assure that accident doses will stay within regulatory and licensing basis limits.

Therefore, the proposed change does not affect the probability or consequences of any ANO-1 analyzed accidents.

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 performance based requirements are an improvement over the requirements imposed by the current technical specifications. Implementation of the proposed *Steam Generator Program* will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. The proposed change enhances SG inspection requirements.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident 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 the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the *Steam Generator Program* to manage SG tube inspection, assessment, repair, and plugging. The requirements established by

the *Steam Generator Program* are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current technical specifications.

Therefore, the margin of safety is not changed by the proposed change to the ANO-1 TSs.

5.3 Environmental Consideration

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

The proposed change to the ANO-1 Steam Generator Integrity Program is consistent with that submitted by Duke Power for the Catawba Nuclear Station, Units 1 and 2 on July 30, 2003 including later supplements.

The proposed change is also consistent with the OL Amendment request by Southern Company for the Farley Nuclear Plant submitted on June 28, 2004 and supplemented on August 5, 2004.

7.0 REFERENCES

1. NEI 97-06, *Steam Generator Program Guidelines.*
2. EPRI *Steam Generator Examination Guideline.*
3. EPRI *Steam Generator Integrity Assessment Guideline.*
4. EPRI *Steam Generator In-situ Pressure Test Guideline.*
5. EPRI *PWR Primary-to-Secondary Leak Guideline.*
6. EPRI *Primary Water Chemistry Guideline.*
7. EPRI *Secondary Water Chemistry Guideline.*
8. Testimony of James Knight Before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.
9. Draft Regulatory Guide 1.121, *Bases for Plugging Degraded Steam Generator Tubes*, August 1976.
10. ASME B&PV Code, Section III, *Rules for Construction of Nuclear Facility Components.*
11. Generic Letter 95-05, *Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking*, August 3, 1995.

Attachment 2

1CAN090401

Proposed Technical Specification Changes (mark-up)

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. RCS primary to secondary LEAKAGE not within limits.~~	~~A.1 Reduce LEAKAGE to within limits.~~	~~4 hours~~
A~~B~~. RCS unidentified or identified LEAKAGE not within limits, except for primary to secondary LEAKAGE.	A~~B~~.1 Reduce LEAKAGE to within limits.	18 hours
B~~C~~. Required Action and associated Completion Time of Condition A ~~or B~~ not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B~~C~~.1 Be in MODE 3. AND B~~C~~.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	----------NOTES---------- 1. Not required to be performed until 12 hours after establishment of steady state operation at or near operating pressure. 2. Not applicable to primary to secondary LEAKAGE. ---------- Verify RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance.	72 hours
SR 3.4.13.2	----------NOTE---------- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG~~steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program~~.	72 hours~~In accordance with the Steam Generator Tube Surveillance Program~~

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 Steam Generator (SG) Tube Integrity

LCO 3.4.16 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.16.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.0 ADMINSTRATIVE CONTROLS

5.5 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Code terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Monthly	At least once per 31 days
Every 6 weeks	At least once per 42 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) ~~Tube Surveillance~~ Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

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~~This program provides controls to ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.~~

~~The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the SG Tube Surveillance Program inspection frequencies.~~

~~a. The first steam generator tubing inspection performed in accordance with 5.5.9.b and 5.5.9.c.1 shall be considered as constituting the baseline condition for subsequent inspections.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

~~b. Examination Methods:~~

~~1. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end.~~

~~2. For examination of the sleeved steam generator tubing at the lower sleeve end, the indications will be compared to those obtained during the baseline sleeved tube inspection. Significant deviations between these indications will be considered sufficient evidence to warrant designation as a degraded tube. Direct quantification of the 40 percent through-wall plugging limit is available with eddy current testing.~~

~~c. Selection and Testing:~~

~~The steam generator sample size is specified in Table 5.5.9-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies as specified in 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of 5.5.9.e. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

~~1. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:~~

~~i. All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and~~

~~ii. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per 5.5.9.c.1.iii.~~

~~A tube inspection (pursuant to 5.5.9.e.1.ix) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

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~~iii. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.~~

~~(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.~~

~~(2) Group A-2: Unplugged tubes with sleeves installed.~~

~~(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure 5.5.9-1.~~

~~iv. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to 5.5.9.d. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category of the OTSG.~~

~~v. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to 5.5.9.d. Tubes with ODIGA identified during previous inspections, which meet the criteria to remain in service, will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with ANO Engineering Report No. 00-R-1005-01.~~

~~2. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

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~~3. The second and third sample inspections during each inservice inspection as required by Table 5.5.9-2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.~~

~~4. 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.~~

~~NOTES:~~

~~(1) In all inspections, previously degraded tubes whose degradations have not been spanned by a sleeve must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.~~

~~(2) Where special inspections are performed pursuant to 5.5.9.c.1.iii, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.~~

~~(3) Where special inspections are performed pursuant to 5.5.9.c.2, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

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

~~1. The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group[*] of tubes following service under all volatile treatment (AVT) conditions fall 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 for that group may be extended to a maximum of 40 months.~~

~~2. If the results of the inservice inspection of a steam generator performed in accordance with Table 5.5.9-2 at 40-month intervals for a given group[*] of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 5.5.9.d.1 and the interval can be extended to 40 months.~~

~~3. 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:~~

~~i. Primary-to-secondary leakage in excess of the limits of Specification 3.4.13 (inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification 5.5.9.c.1.iii, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table 5.5.9-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.~~

~~[*]A group of tubes means: (a) All tubes inspected pursuant to 5.5.9.c.1.iii, or (b) All tubes in a steam generator less those inspected pursuant to 5.5.9.c.1.iii.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

~~If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table 5.5.9-2.~~

~~ii. A seismic occurrence greater than the Operating Basis Earthquake,~~

~~iii. A loss-of-coolant accident requiring actuation of the engineered safeguards, or~~

~~iv. A main steam line or feedwater line break.~~

~~e. Acceptance Criteria:~~

~~1. Terms as used in this program:~~

~~i. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.~~

~~ii. 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 tube wall thickness, if detectable, may be considered as imperfections.~~

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

~~iv. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve or repaired by a rerolled joint.~~

~~The reroll repair process will be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.~~

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

~~5.0 ADMINSTRATIVE CONTROLS~~

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~~vi. Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.~~

~~vii. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall besleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with ANO Engineering Report No. 00-R-1005-01, Rev. 1.~~

~~Axially oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.~~

~~viii. 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.d.3.~~

~~ix. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the tubesheets, that portion of the tube outboard of the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.~~

~~2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table 5.5.9-2.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

~~TABLE 5.5.9-1~~

~~MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION~~

~~Preservice Inspection~~	~~No~~
~~No. of Steam Generators per Unit~~	~~Two~~
~~First Inservice Inspection~~	~~Two~~
~~Second & Subsequent Inservice Inspection~~	~~One[1]~~

~~Table Notation:~~

~~[1] The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3N% 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 sample sequence shall be modified to inspect the most severe conditions.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

~~TABLE 5.5.9-2~~

~~STEAM GENERATOR TUBE INSPECTION~~ [2,3]

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.[1]	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug, reroll, or sleeve Defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug, reroll, or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes.	N/A	N/A

~~NOTES:~~

~~1 $S = \frac{3N}{n}$% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.~~

~~2 For tubes inspected pursuant to 5.5.9.c.1.iii: No action is required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a report to NRC pursuant to 5.6.7.~~

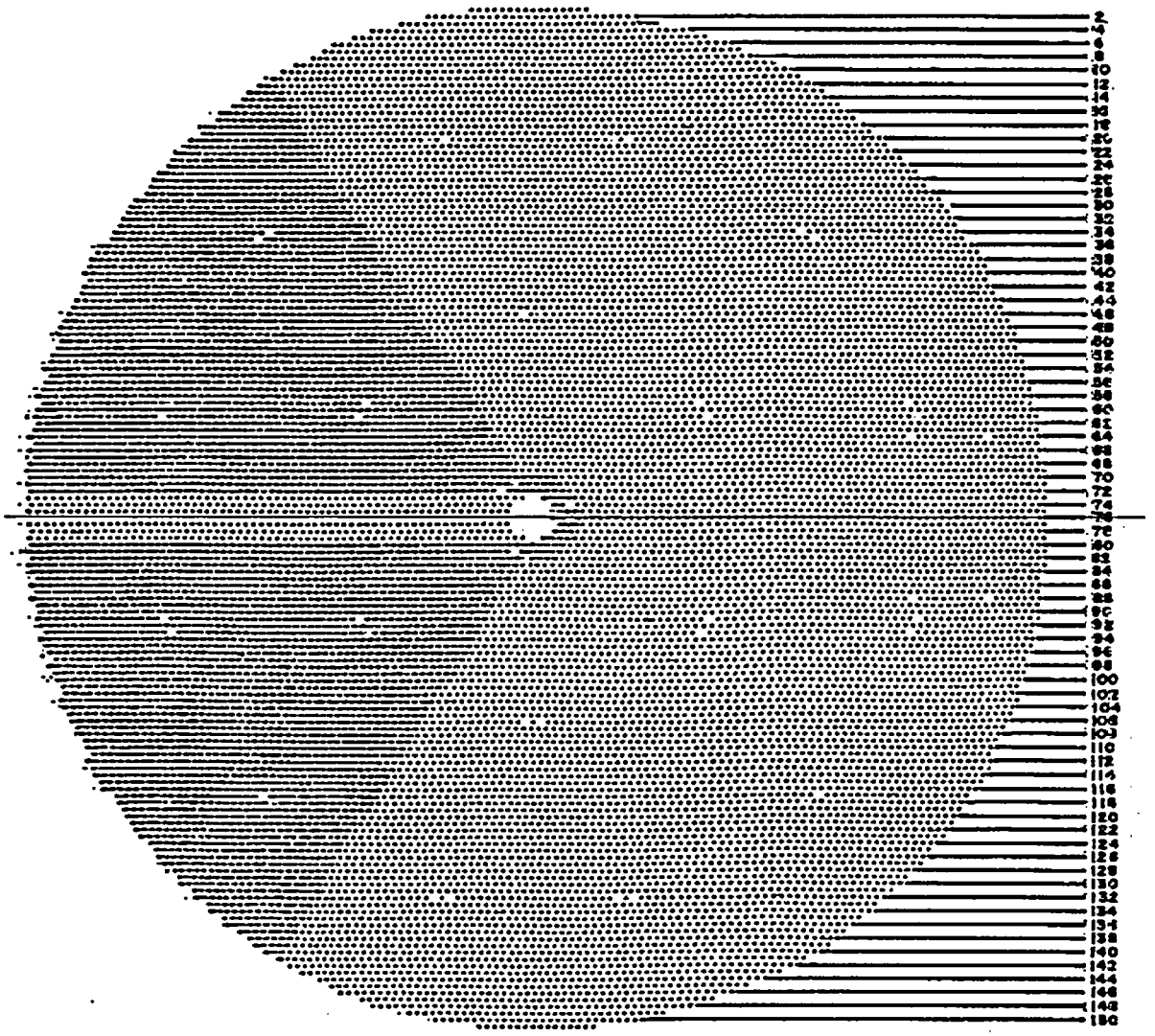
~~3 No more than ten thousand (10,000) sleeves may be installed in both ANO-1 steam generators combined.~~

~~5.0 ADMINSTRATIVE CONTROLS~~

~~5.5 Programs and Manuals~~

~~FIGURE 5.5.9-1~~

~~Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per 5.5.9.c.1.iii~~



~~DESCRIPTION~~	~~TUBE COUNT~~
~~Group A-1: Lane region tubes as defined in 5.5.9.c.1.iii(1)~~	~~382~~
~~Group A-3: Wedge shaped group depicted by darkened region of figure~~	~~4880~~

5.0 ADMINSTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.6 Reactor Building Inspection Report

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

5.6.7 Steam Generator Tube Inspection~~Surveillance~~ Reports

~~a. Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC~~A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The~~is~~ report~~, to be submitted within 90 days of inspection completion,~~ shall include:

a~~1~~. The scope of inspections performed on each SG~~Number and extent of tubes inspected;~~,

b~~2~~. Active degradation mechanisms found~~Location and percent of wall-thickness penetration for each indication of an imperfection;~~,

c~~3~~. Nondestructive examination techniques utilized for each degradation mechanism~~Identification of tubes plugged and tubes sleeved;~~,

d~~4~~. Location, orientation (if linear), and measured sizes (if available) of service induced indications~~Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;~~,

e~~5~~. Number of tubes plugged during the inspection outage for each active degradation mechanism~~Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and~~,

f~~6~~. Total number and percentage of tubes plugged to date~~Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.~~

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

~~b. In addition, the Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 5.5.9-2 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

Attachment 3

1CAN090401

Proposed Technical Specification Bases Changes (mark-up)

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit LEAKAGE from these sources to amounts that do not compromise safe operation. This LCO specifies the types and amounts of allowable LEAKAGE.

SAR Section 1.4, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable criteria for selecting Leakage Detection Systems. Reference 3 provides a comparison of the ANO-1 RCS leak detection systems to Regulatory Guide 1.45 (Ref. 2).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the reactor building is~~are~~ necessary.

A limited amount of leakage inside the reactor building is expected from auxiliary systems that cannot be made leaktight. Leakage from these systems should be detected, located, and isolated from the reactor building atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation. The consequences of violating this LCO include increasing the probability of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions except for the SGTR. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day -assures that the conditions assumed in the safety analysis are bounding.~~1 gpm primary to secondary LEAKAGE as the initial condition.~~

APPLICABLE SAFETY ANALYSES (continued)

Primary to secondary LEAKAGE is a factor in the radioactivity releases resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The SAR (Ref. 4) analysis for SGTR assumes the contaminated secondary fluid is released via turbine bypass valves to the condenser and briefly through the MSSVs to the atmosphere. The 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential compared to the tube rupture leakage.

The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100.

RCS leakage detection capabilities and methods are identified and discussed in SAR Section 4.2.3.8 (Ref. 5) and in the Bases for LCO 3.4.15, "RCS Leakage Detection Instrumentation."

In MODES 1 and 2, RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6). In MODES 3 and 4, RCS operational LEAKAGE satisfies Criterion 4 of 10 CFR 50.36.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the reactor building air monitoring and reactor building sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary. Controlled reactor coolant pump (RCP) seal leakoff is a normal function and is not considered as LEAKAGE.

LCO (continued)

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the reactor building from specifically known and located sources and LEAKAGE through a SG to the secondary system, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 7). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures and to assure the safety analysis is bounding.~~The 150 gallon per day (0.104 gpm) limit on one SG is intended to assure timely shutdown of the plant for appropriate corrective action before rupture of the steam generator tube(s) occurs under normal operating or postulated accident conditions. These limits also serve to provide added assurance that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR 100 (Ref. 7) limits for a design basis steam generator tube rupture or main steam line break. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

APPLICABILITY

In MODES 1, 2, 3, and 4, the LEAKAGE limits are required because the RCS is pressurized and the potential for RCPB LEAKAGE is greatest.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through RCS pressure isolation valves (PIVs) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves in series leak and result in a loss of coolant mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

~~A.1~~

~~If primary to secondary LEAKAGE is in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the primary to secondary RCPB.~~

AB.1

If unidentified LEAKAGE, or identified LEAKAGE, or both, are in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 18 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

BC.1 and BC.2

If any pressure boundary LEAKAGE exists, or if primary to secondary LEAKAGE is not within limit, or if the Required Action and associated Completion Time of Condition A is~~or B are~~ not met, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and may be positively identified by inspection. Total LEAKAGE is determined by performance of an RCS water inventory balance.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.1 (continued)

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. The Surveillance is modified by two notes. ~~Therefore, a~~ Note 1 states~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation at or near operating pressure (i.e., at or near 2155 psig). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the reactor building atmosphere radioactivity and the reactor building sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.16, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 8. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. During normal operation the primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling. ~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.~~

REFERENCES

1. SAR, Section 1.4, GDC 30.

2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.

3. Information Submittal - Comparison of ANO-1 RCS Leak Detection Systems to Regulatory Guide 1.45 (1CAN108607), dated October 14, 1986.

4. SAR, Chapter 14.

5. SAR, Section 4.2.3.8.

6. 10 CFR 50.36.

~~7. 10 CFR 100.~~

7. NEI 97-06, "Steam Generator Program Guidelines."

8. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding unidentified operational RCS LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits or that associated with 1% failed fuel. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that significant affect burst or collapse. In that context, the term "significantly" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.16.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.16.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.16.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the

flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Attachment 4

1CAN090401

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONT COMPL	
Entergy will revise the ANO-1 SAR to reference NEI 97-06 and the Steam Generator Program TSs.	X		**Prior to next scheduled SAR submittal after operating license Amendment approval**
The revised TS requirements under TSTF 449, Draft Revision 2 require those loads that significantly affect burst or collapse be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads. These loads, as well as the other analyses to support a 40% plugging limit, will be analyzed for the RSGs under the existing ANO-1 licensing basis. These analyses will be performed and documented under the requirements of 10 CFR 50.59.	X		**Prior to startup from 1R19**