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2CAN090501

September 19, 2005

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

SUBJECT: License Amendment Request
Proposed Technical Specification Change to ANO-2 Steam Generator Tube Inservice Inspection Program Using Consolidated Line Item Improvement Process
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

CORRES. REFERENCE: 1 NRC letter dated May 20, 1997, *Issuance of Amendment 184 to Facility Operating License No. NPF-6 – Arkansas Nuclear One, Unit No. 2* (2CNA059703)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests an Operating License amendment for Arkansas Nuclear One, Unit 2 (ANO-2) to replace the existing steam generator tube surveillance program with that being proposed by the Technical Specification Task Force in TSTF 449, Revision 4. Specifically, TS 3/4.4.5, *Steam Generators*; TS 3.4.6.2, *Operational Leakage* and TS 6.5.9, *Steam Generator Tube Surveillance Program,* and TS 6.6.7, *Steam Generator Tube Surveillance Reports* are being revised to incorporate the new Steam Generator Program of TSTF 449, Revision 4. The basis for this request is discussed in Attachment 1. Attachments 2 and 3 provide markups of the Technical Specifications (TS) and associated Bases, respectively.

The proposed changes are consistent with the Consolidated Line Item Improvement Process (CLIIP) provided in the May 6, 2005 Federal Register Notice. TSTF-449, Revision 4 is formatted to the Improved Technical Specification (ITS) plants while the ANO-2 technical specifications (TSs) are based on the CE standard TSs. Therefore, the information contained in TSTF-449, Revision 4 has been modified to correspond with the ANO-2 TS format.

As discussed in Section 5.2 of Attachment 1, Entergy is aware of the NRC request to consider non-pressure (bending) loads on the accident induced leak rates of steam generator tube. Entergy believes that the effect of non-pressure loads from external events is not safety

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significant with respect to leakage integrity given the expected effect and existing margins for degradation type, susceptible location and allowable flaw size. If upon completion of EPRI's technical study, it is concluded that the effect of bending loads should be specifically accounted for in integrity assessments, the industry will revise the applicable steam generator program guideline documents to reflect the means developed to account for the loads.

The proposed change contained herein 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 a new commitment as identified in Attachment 4.

Entergy requests approval of the proposed amendment by August 1, 2006. Once approved, the amendment shall be implemented within 90 days.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 19, 2005.

Sincerely,

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 2 (ANO-2). The proposed changes modify the Technical Specifications (TS) and associated Bases for Specification 3/4.4.5, *Steam Generators* and Specification 3.4.6.2, *RCS Leakage*. In addition, Specification 6.5.9, *Steam Generator Tube Surveillance Program* and Specification 6.6.7, *Steam Generator Tube Surveillance Reports* are also being revised to be consistent with the new Steam Generator Program guidance of TSTF-449, Revision 4. Even though the TS Bases will be implemented under the ANO-2 Bases Control Program, they are being provided for NRC review and approval. The proposed TS and Bases changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, *Steam Generator Program Guidelines*, (Reference 1).

The ANO-2 TSs are formatted to the Standard Technical Specifications for Combustion Engineering PWRs (NUREG-0212). Even though the ANO-2 TSs have to be modified from that of the TSTF-449, Revision 4 format, the content of the changes proposed herein are consistent with the Consolidated Line Item Improvement Process contained in the May 6, 2005 Federal Register Notice.

2.0 PROPOSED CHANGE

Specifically, the proposed changes will:

- Revise Technical Specification 3/4.4.5, *Steam Generators*

  TS 3/4.4.5, *Steam Generators* is being revised and will be re-titled as *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* prior to entering HOT SHUTDOWN following a SG tube inspection. The remainder of the TS is being deleted. The title of this specification is revised to be Steam Generator Tube Integrity.

  The TS Bases are being modified to reflect the proposed changes to this Specification.

- Revise Technical Specification 3.4.6.2, *Reactor Coolant System Leakage*

  The proposed change incorporates the LCO of the current TS 3.4.6.2 by removing the 300 gallons per day (gpd) allowance for total primary to secondary leakage and retaining the 150 gpd through any one steam generator (SG). The title of this specification is being revised to *Reactor Coolant System Operational Leakage*.

  Action "a" is being modified to add primary to secondary leakage not within limit along with the PRESSURE BOUNDARY LEAKAGE. Action "b" is being revised to exclude primary to secondary leakage along with other leakage sources for this action statement.

  An exception has been added to SR 4.4.6.2.1 stating that the SR is not applicable to primary to secondary leakage for an RCS water inventory balance. SR 4.4.6.2.2 is added to require primary to secondary leakage be verified to be ≤ 150 gpd per SG.

The TS Bases are being modified to reflect the proposed changes to this Specification.

- Revise TS 6.5.9, *Steam Generator Tube Surveillance Program*

  This program will require provisions for condition monitoring, SG tube integrity performance criteria, SG tube repair criteria, SG tube inspections, primary to secondary leakage, and for SG tube repair methods. This program replaces the previous SG Inservice Inspection requirements from the current specifications. The title of this specification is being revised to *Steam Generator Program.*

  The revised TS requirements under TSTF 449, Revision 4, 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% throughwall tube plugging limit, will be performed for the ANO-2 SG licensing basis. These analyses will be performed prior to implementation of this license amendment under the requirements of 10 CFR 50.59.

- Revise TS 6.6.7, *Steam Generator Tube Surveillance Reports*

  The new reporting requirements under TS 6.6.7 will submit the results of the SG tube inspection within 180 days of entry into Hot Shutdown. This report will replace the 15 day and 12 month reports previously required per this specification. The title of this specification is being changed to *Steam Generator Tube Inspection Report.*

- Revise TS Definitions in 1.14 for IDENTIFIED LEAKAGE and 1.16 for PRESSURE BOUNDARY LEAKAGE

  An editorial change is made to the definition of IDENTIFIED LEAKAGE and PRESSURE BOUNDARY LEAKAGE. The definitions are being modified to clarify that steam generator tube leakage is more properly defined as primary to secondary leakage.

3.0 BACKGROUND

The SG tubes in pressurized water reactors 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 upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they act as a heat transfer surface between the primary and secondary systems to remove heat from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system.

Steam generator tube integrity is necessary in order to satisfy the tubing's safety functions. Maintaining tube integrity ensures that the tubes are capable of performing their intended safety functions consistent with the plant licensing basis, including applicable regulatory requirements.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced 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. When the degradation of the tube wall reaches a prescribed repair criterion, the tube is considered defective and corrective action is taken.

The criteria governing structural integrity of SG tubes were developed in the 1970s and assumed uniform tube wall thinning. This led to the establishment of a through wall SG tube repair criteria (e.g., 40 percent) that has historically been incorporated into most pressurized water reactor (PWR) Technical Specifications and has been applied, in the absence of other repair criteria, to all forms of SG tube degradation where sizing techniques are available. Since the basis of the through wall depth criterion was 360° wastage, it is generally considered to be conservative for other mechanisms of SG tube degradation. The repair criterion does not allow licensees the flexibility to manage different types of SG tube degradation. Licensees must either use the through wall criterion for all forms of degradation or obtain approval for use of more appropriate repair criteria that consider the structural integrity implications of the given mechanism.

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). Under this approach, different methods of inspection and different repair criteria may be developed for different types of degradation. A degradation specific approach to managing SG tube integrity has several important benefits. These include:

- improved scope and methods for SG inspection,
- industry incentive to continue to improve inspection methods, and
- development of plugging and repair criteria based on appropriate NDE parameters.

As a result, the assurance of SG tube integrity is improved and unnecessary conservatism is eliminated. Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG Program. These guidelines include:

- *Steam Generator Examination Guideline*" (Reference 2),
- *Steam Generator Integrity Assessment Guideline* (Reference 3),
- *Steam Generator In-situ Pressure Test Guideline* (Reference 4),
- *PWR Primary-to-Secondary Leak Guideline* (Reference 5),
- *Primary Water Chemistry Guideline* (Reference 6), and
- *Secondary Water Chemistry Guideline* (Reference 7).

These EPRI Guidelines, along with NEI 97-06 (Reference 1), tie the Steam Generator Program 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 provides 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). Since then the industry and the NRC have participated in a series of meetings to resolve the differences and develop the regulatory framework necessary to implement a comprehensive Steam Generator Program.

Revising the existing regulatory framework to accommodate degradation specific management is the most appropriate way to address the issues of regulatory stability, resource expenditure, use of state-of-the-art inservice inspection techniques, repair criteria, and enforceability. The NRC Staff has stated that an integrated approach for addressing SG tube integrity is essential and that materials, systems, and radiological issues that pertain to tube integrity need to be considered in the development of the new regulatory framework.

4.0 TECHNICAL ANALYSIS

The proposed changes do not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. 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 SG inspection requirements and provide additional assurance that the plant licensing basis will be maintained between SG inspections.

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. The analysis of a SGTR event assumes a bounding primary to secondary leakage rate equal to the operational LEAKAGE rate limits in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube.

For design basis accidents such as main steam line break (MSLB) and rod ejection, the SG tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). These analyses typically assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute or increases to 1 gallon per minute as a result of accident induced stresses. For accidents that do not involve fuel damage, the reactor coolant activity levels are at the technical specification values. 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.

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 proposed technical specification changes are generally a significant improvement over the current requirements. They replace an outdated prescriptive technical specification with one that references *Steam Generator Program* requirements that 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 required by current technical specifications. These proposed changes will result in added assurance of the function and integrity of SG tubes.

The table below and associated sections describe the requirements and provide the technical justification for the proposed changes.

Condition or Requirement	Current Licensing Basis	Location - Proposed Change
Operational primary to secondary leakage	RCS Operational Leakage TS ≤ 150 gallons per day per SG. 300 gpd for both SGs.	Retain the existing Operational Leakage limit of 150 gpd per SG. 300 gpd total SG leakage limit is being eliminatd.
RCS primary to secondary leakage through any one SG not within limits	Reduce leakage to within limits in 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours	RCS Operational Leakage TS - Be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours
RCS leakage determined by water inventory balance (SR 4.4.6.2.1) Verify primary to secondary leakage within limits once every 72 hours	Note states: Not required to be performed until 12 hours after establishment of steady state operation None	Modified SR 4.4.6.2.1 to state this SR is not applicable to primary to secondary leakage. Added new SR 4.4.6.2.2 to verify primary to secondary leakage every 72 hours.
SG Tube integrity verification (SR 3/4.4.5)	Verify in accordance with the SG Tube Surveillance Program	New Action statements and SRs are added to ensure that the SGs are maintained in accordance with the *Steam Generator Program*
Frequency of verification of tube integrity	At least every 40 months.	SG Tube Integrity TS – Requires Surveillance Frequency in accordance with TS 6.5.9, *Steam Generator Program*. The ANO-2 replacement SGs are Thermally Treated Alloy 690 allowing a longer inspection period. *Steam Generator Program* – establishes maximum inspection intervals
Tube sample selection	Based on SG Category, industry experience, random selection, existing indications, and results of the initial sample set - 3% times the number of SGs at the plant as a minimum	*Steam Generator Program* and implementing procedures - Dependent on a pre-outage evaluation of actual degradation locations and mechanisms, and operating experience – 20% of all tubes as a minimum.

Condition or Requirement	Current Licensing Basis	Location - Proposed Change
Inspection techniques	Not specified	*Steam Generator Tube Integrity* TS – SR 4.4.5.1 and 4.4.5.2 requires that tube integrity be verified in accordance with the *Steam Generator Program*. *Steam Generator Program* and implementing procedures – Establishes requirements for qualifying NDE techniques. Requires use of qualified techniques in SG inspections. Requires a pre-outage evaluation of potential tube degradation morphologies and locations and an identification of NDE techniques capable of finding the degradation.
Inspection Scope	Hot leg point of entry to (typically) the first support plate on the cold leg side of the U-bend	Steam Generator Program procedures – Inspection scope is defined by the degradation assessment that considers existing and potential degradation morphologies and locations. Explicitly requires consideration of entire length of tube from tube-sheet weld to tube-sheet weld.
Performance criteria	Operational leakage ≤ 150 gallons per day through any one SG. No criteria specified for structural integrity or accident induced leakage.	RCS Operational Leakage TS – Unchanged SG Tube Integrity TS – Requires that tube integrity be maintained. TS 6.5.9 – Defines structural integrity and accident induced leakage performance criteria which are dependent on design basis limits. Provides provisions for condition monitoring assessment to verify compliance.
Repair criteria	Plug tubes with imperfections extending ≥40% through wall and alternate criteria approved by NRC.	*Steam Generator Program* –Criteria unchanged

Condition or Requirement	Current Licensing Basis	Location - Proposed Change
Actions	Performance Criteria not defined. Primary to secondary leakage limit and actions included in the Tech Specs. Plug tubes exceeding repair criteria.	*RCS Operational Leakage* TS and *SG Tube Integrity* TS – Contains primary to secondary leakage limit, SG tube integrity requirements and Actions required upon failure to meet performance criteria. Plug tubes satisfying repair criteria.
Repair methods	Methods (except plugging) require previous approval by the NRC. ANO-2 does not have any tube repair methods being applied.	*Steam Generator Program* –Requirements unchanged
Reporting requirements	Plugging report required 15 days after each inservice inspection, 12 month report documenting inspection results, reports in accordance with §50.72 and when the inspection results fall into category C-3.	*Steam Generator Program* - 180 days after the initial entry into HOT SHUTDOWN after performing a SG inspection Serious SG tube degradation (i.e., tubing fails to meet the structural integrity or accident induced leakage criteria) requires reporting in accordance with §50.72 or §50.73.
Definitions SG Terminology	Normal TS definitions (i.e., Definitions Section) did not address SG Program issues. The Definitions Section uses the term "SG Leakage."	*Steam Generator Program*, TS Bases– Includes terminology applicable only to SGs. The Definitions Section is revised to use the term "primary to secondary leakage."

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Section 1: Operational LEAKAGE

The ANO-2 TS primary to secondary leakage limit had been previously reduced to ≤ 150 gallons per day per SG with a 300 gpd total primary to secondary leakage limit per Correspondence Reference 1. The total steam generator tube leakage limit of 300 gallons per day for all steam generators is being deleted. This limit had been established to ensure that the dose contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. This limit is still protected by 150 gpd per SG specification. 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. This together with the allowable accident induced leakage limit helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR 100 and GDC 19 dose limits or other NRC approved licensing basis for postulated faulted events.

This limit also contributes to meeting the GDC 14 requirement that the reactor coolant pressure boundary "have an extremely low probability of abnormal leakage, of rapidly propagating to failure, and of gross rupture." The proposed surveillance references the *Steam Generator Program*. The *Steam Generator Program* uses the EPRI *Primary-to-Secondary Leak Guideline* (Reference 5) to establish sampling requirements for determining primary to secondary leakage and plant shutdown requirements if leakage limits are exceeded. The guidelines ensure leakage is effectively monitored and timely action is taken before a leaking tube exceeds the performance criteria. The frequency for determining primary to secondary leakage is unchanged (i.e., 72 hours and within 12 hours after establishing stable operating conditions).

Section 2: Operational Leakage Actions

If primary to secondary leakage exceeds 150 gallons per day per SG, a plant shutdown must be commenced. Hot Standby must be achieved within 6 hours and Cold Shutdown achieved within the following 30 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.

The removal of the 4 hour period during which primary to secondary leakage can be reduced to avoid a plant shutdown results in a technical specification that is more conservative than the existing *RCS Operational Leakage* specification. This change is consistent with the *Steam Generator Program* that also does not allow 4 hours before commencing a plant shutdown.

Section 3: RCS Operational Leakage Determined by Water Inventory Balance

The proposed change modifies SR 4.4.6.2.1 that makes the water inventory balance method not applicable to determining primary to secondary leakage. This change is proposed because primary to secondary leakage as low as 150 gallons per day through any one SG cannot be measured accurately by an RCS water inventory balance. This change is necessary to make the surveillance requirement appropriate for the proposed LCO.

Section 4: SG Tube Integrity Verification

The existing Action statements for TS 3.4.5 were to ensure that with one or more inoperable SGs restore the inoperable SG(s) to Operable prior to increasing above a T_{avg} above 200°F. The new Action statement will require that if one or more SG tubes not plugged in accordance with the *Steam Generator Program*, tube integrity will be verified within 7 days, and plug the affected tube(s) in accordance with the *Steam Generator Program* prior to entering Hot Shutdown following the next refueling outage or SG tube inspection. In addition, if Action 'a' cannot be met, then be in HOT STANDBY within the next 6 hours and in Cold Shutdown with the following 30 hours.

The current SR 4.4.5 is based on the original CE STS format and provides details for performing SG inservice inspection. The new SR 4.4.5.1 requires verification of SG tube integrity in accordance with the *Steam Generator Program*. In addition, the new SR 4.4.5.2 requires verification that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering Hot Shutdown following a SG tube inspection.

The *Steam Generator Program* and the EPRI *Pressurized Water Reactor Primary-to-Secondary Leak Guidelines* (Reference 5) provide guidance on leak rate monitoring. During normal operation the program depends upon continuous process radiation monitors and/or radiochemical grab sampling in accordance with the EPRI guidelines. The monitoring and sampling frequency increases as the amount of detected leakage increases or if there are no continuous radiation monitors available.

Section 5: Frequency of Verification of SG 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 tube condition is classified into one of three categories based on the number of tubes found degraded and defective. 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* 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 is found. The interval is limited by existing and potential degradation mechanisms and their anticipated growth rate. In addition, a maximum inspection interval is established in Specification 6.5.9.

The maximum inspection interval for Alloy 690 thermally treated tubing is "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." Even though the maximum interval for Alloy 600 thermally treated tubing and Allow 690 thermally treated tubing is slightly longer than allowed

by current technical specifications, it is only applicable to SGs with advanced materials and only if the SGs are free from active degradation. In addition, the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection. Taken in total, the proposed inspection intervals provide a larger 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* 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.

In summary, the proposed change is an improvement over the current technical specification. The current technical specification bases inspection intervals on the results of previous inspections; it does not require an evaluation of expected performance. The proposed technical specification uses information from previous plant inspections as well as industry experience to evaluate the length of time that the SGs can be operated and still provide reasonable assurance that the performance criteria will be met at the next inspection. The actual interval is the shorter of the evaluation results and the requirements in Reference 3. Allowing plants to use the proposed inspection intervals maximizes the potential that plants will use improved techniques and knowledge since better knowledge of SG conditions supports longer intervals.

<u>Section 6: SG Tube Sample Selection</u>

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* degradation assessment guidance for sampling requirements. The minimum sample size is 20% of the tubes inspected.

The *Steam Generator Program* requires the preparation of a degradation assessment. The degradation assessment is the key document used for planning a SG inspection, where inspection plans and related actions are determined, documented, and communicated. 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* (Reference 2) and EPRI *Steam Generator Integrity Assessment Guidelines* (Reference 3) 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, but 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.

Section 7: SG Inspection Techniques

The Surveillance Requirements proposed in the *Steam Generator Tube Integrity* specification require that tube integrity be verified in accordance with the requirements of the *Steam Generator Program*. The *Steam Generator Program* uses the EPRI *Steam Generator Examination Guidelines* (Reference 2) to establish requirements for qualifying NDE techniques and maintains a list of qualified techniques and their capabilities.

The *Steam Generator Program* requires the performance of a degradation assessment and refers utilities to EPRI *Steam Generator Examination Guidelines* (Reference 2) and EPRI *Steam Generator Integrity Assessment Guidelines* (Reference 3) for guidance on its performance. The degradation assessment will identify current and potential new degradation locations and mechanisms and 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.

Section 8: SG Inspection Scope

The current technical specifications include a definition of inspection that specifies the end points of the eddy current examination of each tube. Typically an inspection is required from the point of entry of the tube on the hot leg side to some point on the cold leg side of the tube, usually at the first tube support plate after the U-bend. This definition is overly prescriptive and simplistic and has led to interpretation questions in the past.

The *Steam Generator Program* 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 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.*

The *Steam Generator Program* provides extensive guidance and a defined process, 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 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.

Section 9: SG Performance Criteria

The proposed change adds a performance-based *Steam Generator Program* to the ANO-2 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 an effective measure for minimizing the frequency of tube ruptures at normal operating and faulted conditions. The performance criteria are specified in Specification 6.5.9. The proposed structural integrity and accident induced leakage performance criteria are new requirements. The requirements and methodologies 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:

> *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.*

The structural integrity performance criterion is based on providing reasonable assurance that a 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 (Power Operation) at the maximum steady state reactor power as defined in the design or equipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T_{hot} should be assessed and included if significant. 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. Additionally, the $3\Delta P$ criterion is measurable through the condition monitoring process.

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 Guidelines* (Reference 3).

The accident induced leakage performance criterion is:

> *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 through any one SG.*

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 100 guidelines, or the radiological limits to control room personnel imposed by GDC-19, or other NRC approved licensing basis.

The operational leakage performance criterion is:

> *150 gallons per day primary to secondary leakage through any one steam generator (SG)*

Plant shutdown will commence if primary to secondary leakage exceeds 150 gallons per day per SG. The operational leakage performance criterion is documented in the *Steam Generator Program* and implemented in Specification 3.4.6.2, *RCS Operational Leakage.*

Proposed Specification 6.5.9 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. In addition, the primary to secondary leakage limit ($\leq$ 150 gallons per day per SG) included in Technical Specification 3.4.6.2, *RCS Operational Leakage*, is consistent with the primary to secondary leakage limit in the current RCS operational leakage specification.

Section 10: SG REPAIR CRITERIA

Repair criteria are those NDE measured parameters at or beyond which the tube must be 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 (Reference 12). A SG degradation-specific management strategy is followed to develop and implement an ARC.

The surveillance requirements of the proposed *Steam Generator tube Integrity* specification require that tubes that satisfy the tube repair criteria be plugged in accordance with approved methods. SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "plugged-on-detection" and their integrity should be assessed. It cannot be guaranteed that every flaw will be detected with a given eddy current technique and, therefore, it is possible that some flaws will not be detected during an inspection. If a flaw is discovered and it is determined that this flaw would have satisfied the repair criteria at the time of the last inspection of the affected tube, this does not mean that the *Steam Generator Program* was violated. However, it may be an indication of a shortcoming in the inspection program.

Section 11: 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 Technical Specification 3.4.6.2. 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 during a previous SG inspection. Under these circumstances, the licensee is expected to take the actions required by Action 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* 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 and therefore do not include the proposed requirement.

During COLD SHUTDOWN and REFUELING, 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 proposed technical specification's change to the Actions required upon exceeding the operational leakage criterion is conservative with respect to the current technical specifications as explained in Section 2 above.

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 6.6.7. The changes in the required reports are discussed in Section 13 below.

Section 12: SG Repair Methods

Repair methods are those means used to re-establish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a SG tube is not a repair. Currently ANO-2 does not use any repair processes. If the repair criteria are reached, the subject tubes will be removed from service.

Section 13: Reporting Requirements

The current technical specifications require the following reports:

- A report listing the number of tubes plugged in each SG submitted within 15 days of the end of the inspection.
- A SG inspection results report submitted within 12 months after the inspection.
- Reports required pursuant to 10 CFR 50.73.

The proposed change to Technical Specification 6.6.7 replaces the 15 day and the SG inspection reports with one report required within 180 days. The proposed report also contains more information than the current SG inspection report. This provision expands the report to provide more substantive information and will be sent earlier (180 days versus 12 months). This allows the NRC to focus its attention on the more significant conditions.

The guidance in NUREG-1022, Rev. 2, *Event Reporting Guidelines 10 CFR 50.72 and 50.73*, identifies serious SG tube degradation as an example of an event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. Steam generator tube degradation is considered serious if the tubing fails to meet the structural integrity or accident induced leakage performance criteria. Serious SG tube degradation would be reportable in accordance with 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) requiring NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence.

The proposed reporting requirements are an improvement as compared to those required by the current technical specifications. The proposed reporting requirements are more useful in identifying the degradation mechanisms and determining their effects. In the unlikely event that a performance criterion is not met, NEI 97-06 (Reference 1) directs the licensee to submit additional information on the root cause of the condition and the basis for the next operating cycle.

The changes to the reporting requirements are performance based. The new requirements remove the burden of unnecessary reports from both the NRC and the licensee, while ensuring that critical information related to problems and significant tube degradation is reported more completely and, when required, more expeditiously than under the current technical specifications.

Section 14: SG Terminology

The proposed *Steam Generator Tube Integrity* specification Bases explain a number of terms that are important to the function of a *Steam Generator Program*. The Technical Specification Bases are controlled by the Technical Specification Bases Control Program contained in TS 6.5.14.

The terms are described below.

1. "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 Part 100 guidelines for offsite doses, or the GDC-19 requirements for control room personnel, or other NRC approved licensing basis.

2. The LCO section of the *Steam Generator Tube Integrity* Bases define the term "burst" 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 (Reference 9), and the historical guidance of draft Regulatory Guide 1.121 (Reference 10). 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 (Reference 11)." Burst, or

gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

The above definition of burst was chosen for a number of reasons:

- The burst definition supports field application of the condition monitoring process. For example, verification of structural integrity during condition monitoring may be accomplished via in situ testing. Since these tests do not have the capability to provide an unlimited water supply, or the capability to maintain pressure under certain leakage scenarios, opening area may be more a function of fluid reservoir rather than tube strength. Additionally, in situ designs with bladders may not be reinforced. In certain cases, the bladder may rupture when tearing or extension of the defect has not occurred. This condition may simply mean the opening of the flanks of the defect was sufficient to permit extrusion of the bladder, and that the actual, or true, burst pressure was not achieved during the test. The burst definition addresses this issue.

- The definition does not characterize local instability or "ligament pop-through", as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. For example, an axial crack about 0.5" long with a uniform depth at 98% of the tube wall would be expected to fail the remaining ligament, (i.e., extend the crack tip in the radial direction) due to deformation during pressurization at a pressure below that required to cause extension at the tips in the axial direction. Thus, this would represent a leakage situation as opposed to a burst situation and a factor of safety of three against crack extension in the axial direction may still be demonstrated. Similar conditions have been observed for localized deep wear indications.

3. The LCO section of *Steam Generator Tube Integrity* Bases defines a "SG tube" 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.*

   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 *Steam Generator Program* 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.

4. 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.

5. 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."

6. The LCO section of *Steam Generator Tube Integrity* Bases describes how to determine whether thermal loads are primary or secondary loads. For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as 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.

Conclusion

The proposed changes will provide greater assurance of SG tube integrity than that offered by the current technical specifications. The proposed requirements are performance based and provide the flexibility to adopt new technology as it matures. These changes are consistent with the guidance in NEI 97-06, *Steam Generator Program Guidelines*, (Reference 1). Adopting the proposed changes will provide added assurance that SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Section 3/4.4.5, *Steam Generator Tube Integrity*, Section 3.4.6.2, *RCS Operational Leakage*, Section 6.5.9, *Steam Generator Tube Surveillance Program*, and Section 6.6.7, *Steam Generator Tube Inspection Report are* being revised to reflect the Steam Generator Program. The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, *Steam Generator Program Guidelines*. The Technical Specification Task Force (TSTF) 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 operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational leakage.

   The structural integrity performance criterion is:

   > *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.*

   The accident induced leakage performance criterion is:

   > *The primary to secondary accident induced leakage rate for any design basis accidents, 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 1gpm through any one SG.*

The operational leakage performance criterion is:

> *The RCS operational primary to secondary leakage through any one SG shall be limited to ≤ 150 gallons per day per SG.*

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. In the analysis of a SGTR event, a bounding primary to secondary leakage rate equal to the leakage rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB) and control element assembly (CEA) ejection, the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed change identify the standards against which tube integrity 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 integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the *Steam Generator Program* required by the proposed change. The program, defined by NEI 97-06, *Steam Generator Program Guidelines*, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I-131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, 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 typical analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gallon per minute with no more than 720 gallons per day in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I-131 are at the technical specification values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current technical specifications and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current technical specifications.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of other design basis events.

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 result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary to secondary leakage that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The 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

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

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, 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.

5.2 Applicable Regulatory Requirements/Criteria

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

10 CFR 50.55a, Codes and Standards –

> *Section (b), ASME Code - c)* Reactor coolant pressure boundary. *(1) 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).

10 CFR 50.65 Maintenance Rule –

> *Each holder of a license to operate a nuclear power plant under §§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. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in §50.82(a)(1), this section only shall apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended functions.*

Under the Maintenance Rule, licensees classify 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 SG "is being effectively controlled through the performance of appropriate preventive maintenance" (Maintenance Rule §(a)(2)). If the performance criteria are not met, a root cause determination of appropriate depth is done and the results evaluated to determine if goals should be established per §(a)(1) of the Maintenance Rule.

NEI 97-06, *Steam Generator Program Guidelines*, and its referenced EPRI guidelines define a SG program that provides the appropriate preventive maintenance that meets the intent of the Maintenance Rule. NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, (Reference 13) offers guidance for implementing the Maintenance Rule should a licensee elect to incorporate additional monitoring goals beyond the scope of those documented in NEI 97-06.

10 CFR 50, Appendix A, 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.*

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

10 CFR 50, Appendix A, 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 criteria. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

10 CFR 50, Appendix A, GDC 32 –Inspection of reactor coolant pressure boundary.

> *Components which are part of the reactor coolant pressure boundary shall be designed to (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.*

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

General Design Criteria (GDC) 14, 30, and 32 of 10 CFR Part 50, Appendix A,

> *Defines requirements for the reactor coolant pressure boundary with respect to structural and leakage integrity. Steam generator tubing and tube repairs constitute a major fraction of the reactor coolant pressure boundary surface area. Steam generator tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure. The Steam Generator Program required by the proposed technical specification establishes performance criteria, repair criteria, repair methods, inspection intervals and the methods necessary to meet them. These requirements provide reasonable assurance that tube integrity will be met in the interval between SG inspections.*

Industry Review of Non-Pressure (Bending) Loads

The industry is currently evaluating a technical issue related to the Accident Induced Leakage Performance Criterion (AILPC) specified in Section 5.5.9.b.2 of our proposed technical specifications. The issue concerns the consideration of non-pressure (bending) loads on the accident induced leak rates of steam generator tubes. The EPRI Steam Generator Management Program (SGMP) is conducting a study to determine if these loads are significant, and if they are, to define how to account for the loads in steam generator tube integrity assessments. In the interim, as this study is being completed, EPRI has completed a preliminary impact assessment. The assessment entitled *Preliminary Assessment of the Impact of Non-Pressure Loads on*

Leakage Integrity of Steam Generator Tubing found that the effect of the loads in question may, in certain circumstances, initiate primary-to-secondary leakage, or increase pre-existing primary-to-secondary leakage during and after load application. The effort also assessed the effect of such loads in combination with the applicable design basis accident. The results indicate that these circumstances are expected to be limited to the presence of significant circumferential cracks located in high bending stress regions of tubing. As of this date, such degradation has not been observed in the Industry. .

The structural integrity impact of non-pressure loads on degraded steam generator tubes has been well-documented in a previous EPRI report (NRC accession number ML050760208.) related to the revised Structural Integrity Performance Criterion (SIPC). Experimental results indicated that neither axial loads nor bending loads have a significant effect on the burst pressure of tubing with axial degradation. Similarly, these loads are considered inconsequential for axially oriented degradation with respect to localized pop-through conditions and corresponding accident leakage. As such, industry evidence indicates that the only meaningful impact of non-pressure loads with respect to leakage are due to the application of bending moments on circumferential cracking.

The EPRI Preliminary Assessment found that for Recirculating Steam Generators (RSGs) bending loads are trivial for circumferential degradation within tubesheet crevices and straight sections of tubing and do not need to be considered. However, specific local regions can be subjected to high bending loads due to either accident-induced crossflow or seismic body forces. Crossflow produces relatively high transient bending loads in the top span region during a steam line break (SLB) event in the original design of once-through steam generators (OTSGs), and some (RSG) designs show transient high bending loads in the U-bend region during seismic events.

After review of available analysis and experimental data, the EPRI Assessment concluded that the effect of high bending loads is only noteworthy for large 100% or near through-wall circumferential degradation. From a degradation assessment perspective, the EPRI study also reported that current industry experience indicates that there have been no observed stress corrosion circumferential cracks that are both capable of leaking and located in high bending stress regions. The industry's preliminary impact assessment and the plans for the further technical study and experimental testing were presented to the Staff in a meeting on August 12, 2005. The NRC Staff did not have any significant comments on the results presented.

Based on the above, Entergy believes that the effect of non-pressure loads from external events is not safety significant for ANO-2 with respect to leakage integrity given the expected effect and existing margins with respect to degradation type, susceptible location and allowable flaw size.

If upon completion of EPRI's technical study, it is concluded that the effect of bending loads should be specifically accounted for in integrity assessments, the industry will revise the applicable steam generator program guideline documents to reflect the means developed to account for the loads.

The ANO-2 proposed TS changes provide requirements that are more effective in detecting SG degradation and prescribing corrective actions. The proposed changes result in added assurance of the function and integrity of SG tubes. Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;
2) Such activities will be conducted in compliance with the Commission's regulations; and
3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed change would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change 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 change 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 change.

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. EPRI Report R-5515-00-2, *Experience of US and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves*, June 5, 2002.
9. Testimony of James Knight Before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.
10. Draft Regulatory Guide 1.121, *Bases for Plugging Degraded Steam Generator Tubes*, August 1976.
11. ASME B&PV Code, Section III, Rules for Construction of Nuclear Facility Components.

12. Generic Letter 95-05, *Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking*, August 3, 1995.

13. NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Revision 3.

14. S. C. Collins memo to W. D. Travers, *Steam Generator Action Plan Revision to Address Differing Professional Opinion on Steam Generator Tube Integrity*, May 11, 2001.

Attachment 2

2CAN090501

Proposed Technical Specification Changes (mark-up)

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels – The injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels – The injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital computer channels – The exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except controlled leakage) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system (primary to secondary leakage).

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or controlled leakage.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except ~~steam generator tube~~primary to secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

AZIMUTHAL POWER TILT – T_q

1.17 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

1.18 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.19 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

REACTOR COOLANT SYSTEM

STEAM GENERATORS (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 ~~Each steam generator shall be OPERABLE.~~

a. SG tube integrity shall be maintained.

b. 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.

ACTION:

~~With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing Tavg above 200°F.~~
The Actions may be entered separately for each SG tube.

a. With one or more SG tubes satisfying the tube repair criteria and are not plugged in accordance with the Steam Generator Program,

   1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next inspection, and
   2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering Hot Shutdown following the next refueling outage or SG tube inspection

b. If the required Action and Allowed Outage Time of Action a. above cannot be met or the SG tube integrity cannot be maintained, be in Hot Standby within the next 6 hours and in Cold Shutdown with the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 ~~Each steam generator shall be demonstrated OPERABLE in accordance with the Steam Generator Tube Surveillance Program.~~ Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering Hot Shutdown following a SG tube inspection.

REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. ~~300 gallons per day total primary-to-secondary leakage through both steam generators and~~ 150 gallons per day primary to secondary leakage through any one steam generator (SG),

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for those Reactor Coolant System Pressure Isolation Valves identified in Table 3.4.6.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or any primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and primary to secondary leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two valves* in each high pressure line having a non-functional valve and be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* These valves may include check valves for which the leakage rate has been verified, manual valves or automatic valves. Manual and automatic valves shall be tagged as closed to preclude inadvertent valve opening.

REACTOR COOLANT SYSTEM LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by:

a. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode.

b. Monitoring the reactor head flange leakoff temperature at least once per 24 hours.

4.4.6.2.2 Primary to secondary leakage shall be verified to be ≤ 150 gallons per day per SG at least once per 72 hours.

4.4.6.2.~~2~~3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4.6-1 shall be demonstrated OPERABLE by individually verifying leakage to be within its limit:

a. Prior to entering MODE 2 after each refueling outage,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months, and

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

6.5.9 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 tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, 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 through any one SG.

   3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "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.

~~6.5.9.1 Steam Generator Sample Selection and Inspection~~

~~Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 6.5.9-1.~~

~~6.5.9.2 Steam Generator Tube Sample Selection and Inspection~~

~~The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 6.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 6.5.9.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 6.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

~~a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.~~

~~b. The first sample of tubes selected for each inservice inspection (subsequent to the pre-service inspection) of each steam generator shall include:~~

~~1. All non-plugged tubes that previously had detectable wall penetrations (> 20%).~~

~~2. Tubes in those areas where experience has indicated potential problems.~~

~~3. A tube inspection (pursuant to Specification 6.5.9.4.a.9) 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.~~

~~c. The tubes selected as the second and third samples (if required by Table 6.5.9-2) during each inservice inspection may be subjected to a partial inspection provided:~~

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

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

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. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1 and d.2 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 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.]

2. 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.

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

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

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

~~6.5.9.3 Inspection Frequencies~~

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

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

~~A one-time inspection interval of a maximum of once per 40 months is allowed for the inspection performed immediately following the 2R15 outage. This is an exception to 6.5.9.3.a in that the interval extension is based on all of the results of one inspection falling into the C-1 category.~~

~~b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 6.5.9-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 6.5.9.3.a; the interval may then be extended to a maximum of once per 40 months.~~

~~c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 6.5.9-2 during the shutdown subsequent to any of the following conditions:~~

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

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

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

~~4. A main steam line or feedwater line break.~~

~~6.5.9.4 Acceptance Criteria~~

~~a. As used in this Specification~~

~~1. Tubing or Tube means that portion of the tube which forms the primary system to secondary system pressure boundary.~~

~~2. 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.~~

~~3. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.~~

~~4. Degraded Tube means a tube containing imperfections ≥20% of nominal wall thickness caused by degradation.~~

~~5. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

~~6. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.~~

~~7. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging because it may become unserviceable prior to the next inspection. The plugging limit is equal to 40% of the nominal tube wall thickness.~~

~~8. 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 6.5.9.3.c, above.~~

~~9. Tube Inspection means an inspection of the steam generator tube from tube end (cold leg side) to tube end (hot leg side).~~

~~10. Pre-service Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the hydrostatic test and prior to POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.~~

~~b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 6.5.9-2.~~

~~TABLE 6.5.9-1~~

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

~~Pre-service Inspection~~	~~Yes~~
~~No. of Steam Generators per Unit~~	~~Two~~
~~First Inservice Inspection~~	~~One~~
~~Second & Subsequent Inservice Inspections~~	~~One[1]~~

~~Table Notation:~~

1. ~~The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.~~

~~TABLE 6.5.9-2~~

~~STEAM GENERATOR TUBE INSPECTION~~

~~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.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Plug defective tubes and inspect additional 2S tubes in this S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Plug defective tubes and inspect additional 4S tubes in this S.G.~~	~~C-1~~	~~None~~
					~~C-2~~	~~Plug 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 defective tubes and inspect 2S tubes in the other S.G.~~ ~~Special Report to NRC per Specification 6.6.7~~	~~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 the other S.G. and plug defective tubes.~~ ~~Special Report to NRC per Spec. 6.6.7~~	~~N/A~~	~~N/A~~

~~S = 3 (2/n) % Where n is the number of steam generators inspected during an inspection.~~

6.6.6 not used

6.6.7 Steam Generator Tube ~~Surveillance~~ Inspection Reports

A report shall be submitted within 180 days after the initial entry into Hot Shutdown following completion of an inspection performed in accordance with the Specification 6.5.9, *Steam Generator (SG) Program.* The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

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

h. The effective plugging percentage for all plugging in each SG

~~a. Following each inservice inspection of steam generator tubes the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.~~

~~b. The complete results of the steam generator tube inservice inspection shall be reported within 12 months following the completion of the inservice inspection. This report shall include:~~

~~1. Number and extent of tubes inspected.~~

~~2. Location and percent of wall-thickness penetration for each indication of an imperfection.~~

~~3. Identification of tubes plugged.~~

~~c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission as denoted by Table 6.5.9-2. Notification of the Commission will be made prior to resumption of plant operation (i.e., prior to entering Mode 4). The written report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

6.6.8. Specific Activity

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

(2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded the results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

Attachment 3

2CAN090501

Proposed Technical Specification Bases Changes (mark-up)

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The steam bubble functions to relieve RCS pressure during all design transients.

The requirement that 150 KW of pressurizer proportional heaters per bank and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss-of-offsite power condition to maintain natural circulation at HOT STANDBY. Action (b) is applicable to conditions when a single proportional heater bank is inoperable for any reason, whether loss of an emergency power supply or the loss of capability (less than 150 KW output per bank as required by Surveillance 4.4.4.2(b)). Action (b) requires that the AOT associated with a single inoperable EDG (TS 3.8.1.1 Action b.3) be entered when a heater bank has an inoperable emergency power supply. This action allows 14 days to restore an inoperable EDG to OPERABLE status provided the AACDG is available. If the AACDG is not available, the EDG must be restored to OPERABLE status within 72 hours. In the event the circuit between the AACDG and the proportional heater bank is unavailable (480 V Load Center breaker supplying the heater bank is open or other similar break in the circuit), the AACDG cannot be considered available to the heater bank. Likewise, if a proportional heater bank output capability is known to be < 150 KW, the availability of the AACDG provides no additional capability. Both of these cases require the proportional heater bank to be restored to OPERABLE status within 72 hours in accordance with Note 1 of EDG TS 3.8.1.1 Action b.3 (Reference Licensing Memo LIC 04-045 and -046).

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY

~~The Steam Generator Tube Surveillance Program ensures that the structural integrity of this portion of the RCS will be maintained. The program is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.~~

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. 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.

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 6.5.9, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.5.9, 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 6.5.9. 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 NEI 97-06, *Steam Generator Program Guidelines* (Reference 1).

Safety Analysis

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2, "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 3.4.8, "RCS Specific Activity," limits. 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).

Limiting Condition for Operation

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 a 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 6.5.9, *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.

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 significantly 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 tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as 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.

  Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB.

- 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 through any one SG. 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 3.4.6.2, *RCS Operational Leakage*, and limits primary to secondary leakage through any one SG to 150 gallons per day per SG. 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.

<u>Actions</u>

The Actions may be entered separately for each SG tube. This is acceptable because the Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Actions may allow for continued operations, and subsequent affected SG tubes are governed by subsequent application of associated Actions.

Action "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 4.4.5.2. 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 refueling outage or 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, Action "b." applies.

An allowed outage 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, 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 HOT SHUTDOWN following the next refueling outage or SG inspection. This time period is acceptable since operation until the next inspection is supported by the operational assessment.

Action "b" applies if the actions and associated allowed outage time of Action "a." are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. The allowed outage time 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

During shutdown periods the SGs are inspected as required by SR 4.4.5.1 and the Steam Generator Program. NEI 97-06, *Steam Generator Program Guidelines* (Reference 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 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 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 6.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

As required by SR 4.4.5.2, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.5.9 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.

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, "Pressurized Water Reactor Steam Generator Examination Guidelines."

3/4.4.6.2 REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

~~The total steam generator tube leakage limit of 300 gallons per day for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 150 gallon per day leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~
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 which 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.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduce the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a

Attachment 4

2CAN090501

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	
The revised TS requirements under TSTF 449, Revision 4 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 ANO-2 SG licensing basis. These analyses will be performed and documented under the requirements of 10 CFR 50.59.	X		**Prior to implementation of the license amendment**