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W3F1-2005-0040

July 21, 2005

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

SUBJECT: License Amendment Request NPF-38-262
Proposed Technical Specification Change to Waterford-3 Steam Generator Tube Inservice Inspection Program Using Consolidated Line Item Improvement Process
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

CORRES. REFERENCES:

1 Entergy letter dated March 15, 2005, *Proposed Technical Specification Change Regarding Tubesheet Inspection Depth for Steam Generator Tube Inspections* (W3F1-2005-0009)

2 Entergy letter dated July 14, 2004, *Supplement to Amendment Request NPF-38-249 Extended Power Uprate* (W3F1-2004-0052)

3 Entergy letter dated July 15, 2004, *License Amendment Request NPF-38-256 Alternate Source Term* (W3F1-2004-0053)

4 Entergy letter dated October 19,2004, *Supplement 4 to Amendment Request NPF-38-256, Alternate Source Term* (W3F1-2004-0101)

5 NRC letter dated April 15, 2005, *Waterford Steam Electric Station, Unit 3 (Waterford 3) -Issuance of Amendment Re: Extended Power Uprate*

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the Operating License amendment for Waterford Steam Electric Station, Unit 3 (Waterford-3) to replace the existing steam generator tube surveillance program with that being proposed by the Technical Specification Task Force in TSTF 449, Revision 4 as discussed in Attachment 1. The proposed change in Attachments 2 and 3 revises the Technical Specifications (TS) and associated Bases for Specification 3/4.4.4, *Steam Generators* and Specification 3.4.5.2, *Operational Leakage.* In addition, Specification 6.5.9, *Steam Generator Program,* and Specification 6.9.1.5, *Steam*

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Generator Tube Surveillance Reports, are being added to the Waterford-3 technical specifications. Both the technical specifications (TSs) and Bases are being provided for NRC review and approval.

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 Waterford-3 TSs is based on the CE standard technical specifications. Therefore, the information contained in TSTF-449, Revision 4 has been modified to the Waterford-3 TS format.

Entergy is aware of the current request by the NRC to include non-pressure loads (i.e. GDC-2 loads) into the industry analysis for accident induced leakage to comply with TSTF-449, Revision 4. These loads are not believed to be safety significant and are also not believed to contribute significantly to the pressure loads already considered in the analysis. However, these loads are not being considered in this application until resolution through the continuing discussion with the NRC Staff on this matter and the safety assessment being performed by the industry to address these loads.

In letter dated March 15, 2005 (Reference 1 above), Entergy submitted a change to the Waterford-3 TSs in accordance with Generic Letter 2004-01 for a specific limitation for tubesheet inspection depth. This amendment is still under review by the NRC. The TS pages contained in a pending license amendment request also impacts one or more of the pages being submitted in this request. Therefore, the approval pages for the subsequent approval will need to be modified to reflect the current change.

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 Entergy has determined this change involves no significant hazards consideration. The basis for this determination is included in Attachment 1. The proposed change includes new commitments as identified in Attachment 4.

Entergy requests approval of the proposed amendment by February 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 July 21, 2005.

Sincerely,

for J Venable

JEV/SAB/RLW

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

W3F1-2005-0040

Analysis of Proposed Technical Specification Change

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

The proposed changes revise the Standard Technical Specification (STS), for Waterford Steam Electric Station, Unit 3 (Waterford-3), Docket No. 50-382, License No. NPF-38. The proposed changes modify the Technical Specifications (TS) and associated Bases for Specification 3/4.4.4, *Steam Generators* and Specification 3.4.5.2, *Operational Leakage.* In addition, a new Specification 6.5.9, *Steam Generator Program,* and a new Specification 6.9.1.5, *Steam Generator Tube Surveillance Reports*, are being incorporated into the Waterford-3 Technical Specifications (TSs). Both the TSs and Bases are being provided for NRC review and approval. The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, *Steam Generator Program Guidelines*, (Reference 1). The proposed changes and additions to the Waterford-3 TSs and Bases are provided in Attachments 2 and 3, respectively.

The Waterford-3 TSs are formatted to the Standard Technical Specifications for Combustion Engineering PWRs (NUREG-0212). Even though the Waterford-3 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

The proposed change will:

- Revise Technical Specification 3/4.4.4, *Steam Generators*

  TS 3/4.4.4, *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 or repaired 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.

- Revise Technical Specification 3.4.5.2, *Reactor Coolant System Operational Leakage*

  The proposed change incorporates the LCO of the current TS 3.4.5.2 which had already reduced the allowable leakage to ≤ 75 gallons per day (gpd) per steam generator (SG). This value is Waterford-3 specific as proposed in Correspondence References 2, 3, and 4 and approved by the NRC in Correspondence References 5. This additional conservatism was established in order that more dose assessment margin would be retained.

  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.

SR 4.4.5.2.2 is added to require primary to secondary leakage be verified to be ≤ 75 gpd per SG. An exception has been added to SR 4.4.5.2.1 stating that the SR is not applicable to primary to secondary leakage for an RCS water inventory balance.

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

- Add a new program requirement to TS 6.5.9 entitled *Steam Generator 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. Two areas, not specific to TSTF-449, Revision 4, are being incorporated from the current TSs to the Steam Generator Program.

  These are (1) the allowance to not inspect, plug, or repair 10.4 inches below the bottom of the expansion transition or top of the tubesheet, whichever is lower, to the tube end, and (2) the alternate repair criteria for defective tubes which may be repaired in accordance with CENS Report CEN-605-P, *Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves*, Revision 00-P, dated December 1992.

  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 Waterford-3 SG licensing basis. These analyses will be performed prior to implementation of this license amendment under the requirements of 10 CFR 50.59.

- Add a new TS 6.9.1.5, *Steam Generator Tube Inspection Report*

  A new reporting requirement is being added to TS 6.9.1.5, *Steam Generator Tube Inspection Report* to 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 in TS 3/4.4.4.

- Revise TS Definitions in 1.14 for IDENTIFIED LEAKAGE and 1.21 for PRESSURE BOUNDARY LEAKAGE

  An editorial change is made to the definition of IDENTIFIED LEAKAGE and PRESSURE BOUNDARY LEAKAGE. The definition is being modified to clarify 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 entire 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 (Reference 1) 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.

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 is based on the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes a Loss of Offsite Power with subsequent releases to the atmosphere via Main Steam Safety Valves and Atmospheric Dump Valves.

For other design basis accidents such as main steam line break (MSLB), control element assembly (CEA) ejection, and reactor coolant pump seized rotor/sheared shaft, the SG tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). The TS primary to secondary limitation of ≤ 75 gpd bounds the other accident analyses. 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 Specific Activity (including DOSE EQUIVALENT I-131) in primary coolant to ensure the plant is operated within its analyzed condition.

The current technical specification limit of ≤ 75 gallons per day per SG of primary to secondary leakage through any one SG is based on the assumptions discussed in Correspondence Reference 2. 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.

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. As a result, 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	Section
Operational primary to secondary leakage	RCS Operational Leakage TS ≤ 75 gallons per day per SG	Retain the existing Operational Leakage limit	1
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	2
RCS leakage determined by water inventory balance (SR 4.4.5.2.1) 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 4.4.5.2.1 to state this SR is not applicable to primary to secondary leakage. Added new SR 4.4.5.2.2 to verify primary to secondary leakage every 72 hours.	3
SG Tube integrity verification (SR 3/4.4.4)	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*	4
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*. Frequency is dependent on tubing material and the previous inspection results and the anticipated defect growth rate. *Steam Generator Program* – establishes maximum inspection intervals	5
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.	6

Condition or Requirement	Current Licensing Basis	Location - Proposed Change	Section
Inspection techniques	Not specified	*Steam Generator Tube Integrity* TS – SR 4.4.4.1 and 4.4.4.2 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.	7
Inspection Scope	Currently proposed TS change in inspection scope from 10.4 inches below the bottom of the hot leg expansion transition or top of the tubesheet, whichever is lower, completely around the U-bend to the top support of the cold leg.	*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 except for the portion of the tube 10.4 inches from the tubesheet.	8
Performance criteria	For those analyzed events that result in a faulted steam generator (e.g., MSLB), 540 gpd primary to secondary leakage is assumed through the faulted steam generator while greater than or equal to 75 gpd primary-to-secondary leakage is assumed through the intact steam generator. No criteria specified for structural integrity or accident induced leakage.	Retain *Steam Generator Tube Integrity* TS – Requires that tube integrity be maintained. *Steam Generator Program* – 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.	9

Condition or Requirement	**Current Licensing Basis**	**Location - Proposed Change**	**Section**
Repair criteria	Plug or repair tubes with imperfections extending ≥40% through wall and alternate criteria approved by NRC. Approved alternate repair criteria listed in the Technical Specification.	*Steam Generator Program* –Criteria unchanged	10
Actions	Performance Criteria not defined. Primary to secondary leakage limit and actions included in the Tech Specs. Plug or repair 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 or repair tubes satisfying repair criteria.	11
Repair methods	Methods (except plugging) require previous approval by the NRC. Approved methods listed in Technical Specification.	*Steam Generator Program* –Requirements unchanged	12
Reporting requirements	Plugging and repair report required 15 days after each inservice inspection, 12 month report documenting inspection results, and reports in accordance with §50.72 when the inspection results fall into category C-3.	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. *Steam Generator Program* - 180 days after the initial entry into HOT SHUTDOWN after performing a SG inspection	13
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."	14

Section 1: Operational LEAKAGE

The primary to secondary leakage limit had been previously reduced to ≤ 75 gallons per day per SG in the current TSs. 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 50.67 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 75 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 4.4.5.2.1 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 75 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 3.4.4 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 or repaired in accordance with the *Steam Generator Program*, tube integrity will be verified within 7 days, and plug or repair 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 4.4.4.0 is based on the original CE STS format and provides details for performing SG inservice inspection. The new SRs require verification of SG tube integrity in accordance with the *Steam Generator Program*. In addition, the SRs require verification that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired 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.

The Surveillance frequency is unchanged. Determination of the primary to secondary leakage is required every 72 hours. The SR is modified by a Note stating the SR is not applicable to primary to secondary leakage. The SR is not required to be performed until 12 hours after establishment of stable operating conditions. As stated above, additional monitoring of primary to secondary leakage is also required by the *Steam Generator Program* based upon guidance provided in Reference 5.

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 requirement for Alloy 600 mill annealed tubing (600MA) is "Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected." This frequency is at least as conservative as the current technical specification requirement. The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. Reference 8 discusses the comparison of Alloy 600MA, 600 thermally treated (TT) and 690TT tubing.

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.

Section 6: SG Tube Sample Selection

The current technical specifications base tube selection on SG conditions and industry and plant experience. The minimum sample size is 3% of the tubes times the number of SGs in the plant. The proposed change refers to the *Steam Generator Program* 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.

A currently proposed TS change (Correspondence Reference 1) also establishes an inspection scope from 10.4 inches below the bottom of the hot leg expansion transition or top of the tubesheet, whichever is lower, completely around the U-bend to the top support of the cold leg.

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 Waterford-3 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 proposed structural integrity and accident induced leakage performance criteria are new requirements. The performance criteria are specified in Specification 6.5.9. 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:

> *The primary to secondary accident induced leakage rate for all design basis accidents, other than a steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed [1 gpm] per SG, [except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program].*

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 50.67 guidelines, or the radiological limits to control room personnel imposed by GDC-19, or other NRC approved licensing basis (i.e. 10 CFR 50.67).

The limit for accident induced leakage is 540 gpd in any one SG. Use of an increased accident induced leakage limit approved in conjunction with alternate repair criteria (ARC) is limited to the specific criteria and type of degradation for which it was granted and is described in the SG Program.

The operational leakage performance criterion is:

> *The RCS operational primary to secondary leakage through any one steam generator shall be limited to ≤ 75 gallons per day per SG.*

Plant shutdown will commence if primary to secondary leakage exceeds 75 gallons per day per SG from any one SG. The operational leakage performance criterion is documented in the *Steam Generator Program* and implemented in Specification 3.4.5.2, *RCS Operational Leakage*.

Proposed Administrative 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 (≤ 75 gallons per day per SG) included in Technical Specification 3.4.5.2, *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 repaired or removed from service by plugging.

Tube repair criteria are established for each active degradation mechanism. Tube repair criteria are either the standard through-wall depth-based criterion (e.g., 40% through-wall for most plants) or through-wall depth based criteria for repair techniques approved by the NRC, or other ARC approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05 (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 or repaired in accordance with approved methods. SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/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.

Waterford-3 defective tubes may be repaired in accordance with CENS Report CEN-605-P, "Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves," Revision 00-P, dated December 1992. This is the same criteria that are listed in the existing Technical Specifications. In addition, Technical Specification 6.5.9 lists any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

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 3.4.5.2. 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 or repair 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 repaired or plugged. A report will be submitted to the NRC in accordance with Technical Specification 6.9.1.5. The changes in the required reports are discussed in Section 13 below.

<u>Section 12: SG Repair Methods</u>

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.

The purpose of a repair is typically to reestablish or replace the RCPB. The proposed Steam Generator Tube Integrity surveillance requirements requires that tubes that satisfy the tube repair criteria be plugged or repaired in accordance with the *Steam Generator Program*. Repair methods established in accordance with the *Steam Generator Program* are listed in Technical Specification 6.5.9 as in the current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube repair criteria.

Steam generator tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired/plugged-on-detection" and their integrity is assessed. This requirement is unchanged by the proposed technical specifications. Note that SG plug designs do not require NRC review and therefore plugging is not considered a repair in the context of this requirement.

The proposed approach is not a change to the technical specifications.

Section 13: Reporting Requirements

The current technical specifications require the following reports:

- A report listing the number of tubes plugged or repaired 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 6.9.1.5 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 6.16.

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 50.67 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 and any repairs to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.*

   This definition ensures that all portions of SG tubes that are part of the RCPB, with the exception of the tube-to-tubesheet weld, are subject to *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.

<u>Conclusion</u>

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 3/4.4.4, *Steam Generator Tube Integrity*, Section 3.4.5.2, *RCS Operational Leakage*, are being revised and Section 6.5.9, *Steam Generator Tube Surveillance Program*, and Section 6.9.1.5, *Steam Generator Tube Inspection Report are* being added to the Waterford-3 Technical Specifications. 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 540 gallons per day through any one SG, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program.*

The operational leakage performance criterion is:

> *The RCS operational primary to secondary leakage through any one SG shall be limited to ≤ 75 gallons per day per SG.*

A steam generator tube rupture (SGTR) event is one of the design basis accidents that is 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), control element assembly (CEA) ejection, and reactor coolant pump seized rotor/sheared shaft 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 account 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 Specific Activity 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 Specific Activity in primary coolant to ensure the plant is operated within its analyzed condition. For those analyzed events that do not result in faulted steam generators, greater than or equal to 75 gpd primary to secondary leakage per steam generator is assumed in the analysis. For those analyzed events that result in a faulted steam generator (e.g., MSLB), 540 gpd primary to secondary leakage is assumed through the faulted steam generator while greater than or equal to 75 gpd primary to secondary leakage is assumed through the intact steam generator.

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, repair, and plugging. The requirements established by the *Steam Generator Program* are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current technical specifications.

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.

<u>*10 CFR 50, Appendix A, GDC 14* – Reactor Coolant Pressure Boundary.</u>

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.

<u>*10 CFR 50, Appendix A, GDC 30* – Quality of reactor coolant pressure boundary.</u>

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.

<u>*10 CFR 50, Appendix A, GDC 32 –Inspection of reactor coolant pressure boundary.*</u>

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.

<u>*General Design Criteria (GDC) 14, 30, and 32 of 10 CFR Part 50, Appendix A,*</u>

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.

The proposed change provides requirements that are more effective in detecting SG degradation and prescribing corrective actions. The proposed change results 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

W3F1-2005-0040

Proposed Technical Specification Changes (mark-up)

DEFINITIONS

IDENTIFIED LEAKAGE (Continued)

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)

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.16 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.1.7 and 6.9.1.8.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

DEFINITIONS

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.20 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

primary to secondary

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

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

REACTOR COOLANT SYSTEM

3/4.4.4 STEAM GENERATOR~~S~~ (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

~~3.4.4 Each steam generator shall be OPERABLE.~~ INSERT 1

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 T_{avg} above 200°F.~~ INSERT 2

SURVEILLANCE REQUIREMENTS

INSERT 3

~~4.4.4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.~~

~~4.4.4.1 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 4.4-1.~~

~~4.4.4.2 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 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.4.4 The tubes selected for each inservice inpection shall include at least 3% of the total number of tubes in all steam generators; the tubes sèlected 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 preservice inspection) of each steam generator shall include:~~

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

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

3. A tube inspection (pursuant to Specification 4.4.4.4a.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 ... subjected to

... tubes from ... ith

Pages 3/4 4-12 through 3/4 4-16 have been deleted.

... es where

The resul ... of the following

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

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

Next Page is 3/4 4-17

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

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.4.3a.; the interval may then be extended to a maximum of once per 40 months.

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

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.5.2.

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 main feedwater line break.

4.4.4.4 Acceptance Criteria

a. As used in this Specification

1. Tubing or tube means that portion of the tube or sleeve 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 greater than or equal to 20% of the 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 or repair limit. A tube containing a defect is defective.

7. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and 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 4.4.4.3c., above.

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

10. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection was performed prior to field hydrostatic test and prior to initial 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 or repair all tubes exceeding the plugging or repair limit and all tubes containing through-wall cracks) required by Table 4.4-2. Defective tubes may be repaired in accordance with CENS Report CEN-605-P, "Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves," Revision 00-P, dated December 1992.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special 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 or sleeved.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 6% of the tubes 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 4.4-2

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 or sleeve defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N. A.	N. A.
			C-2	Plug or sleeve defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug or sleeve defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N. A.	N. A.
	C-3	Inspect all tubes in this S. G. plug or sleeve defective tubes and inspect 2S tubes in each other S. G. Notification to NRC pursuant to 50.72(b)(2) to 10CFR Part 50	All other S. G.s are C-1	None	N. A.	N. A.
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2	N. A.	N. A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug or sleeve defective tubes. Notification to NRC pursuant to 50.72(b)(2) of 10CFR Part 50	N. A.	N. A.

$S = \frac{6}{n}\%$ Where n is the number of steam generators inspected during an inspection

~~WATERFORD~~

Insert 1 (TS 3/4.4.4)

3.4.4

a. SG tube integrity shall be maintained.

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

Insert 2 (TS 3/4.4.4)

The Actions may be entered separately.

a. With one or more SG tubes satisfying the tube repair criteria and are not plugged or repaired 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 or repair 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.

Insert 3 (TS 3/4.4.4)

4.4.4.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.4.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired 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

3.4.5.2 Reactor Coolant System operational leakage shall be ~~limited~~ to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 75 gallons per day primary-to-secondary leakage per steam generator,

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

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

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

ACTION:

INSERT 4

a. With any PRESSURE BOUNDARY LEAKAGE, ~~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 limits, excluding PRESSURE BOUNDARY LEAKAGE, primary to secondary leakage and leakage from Reactor Coolant System pressure isolation valves, 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 one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

NOTE: Not required to be performed until 12 hours after establishment of steady state operation.

INSERT 5

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

INSERT 6

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

3

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1, Section A and Section B, shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

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

d. Following valve actuation for valves in Section B due to automatic or manual action or flow through the valve:

1. Within 24 hours by verifying valve closure, and

2. Within 31 days by verifying leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

4

4.4.5.2.3 Each Reactor Coolant System pressure isolation valve power-operated valve specified in Table 3.4-1, Section C, shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months, and

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

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

Insert 4 (TS 3.4.5.2)

or any primary to secondary leakage not within limit,

Insert 5 (Note to SR 4.4.5.2)

except for primary to secondary leakage,

Insert 6 (TS 4.4.5.2.2)

4.4.5.2.2 Primary to secondary leakage shall be verified to be ≤ 75 gallons per day per SG at least once per 72 hours.

6.5.8 INSERVICE TESTING PROGRAM

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

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

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities.

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

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

INSERT 7

Pages 6-9
through
page 6-13
not used

EDITORS NOTE:
PAGE NUMBERS WILL BE
ADJUSTED ON FINAL
REVISED COPY

Insert 7 (New SG Program)

6.5.9, STEAM GENERATOR (SG) 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 or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired 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. Primary to secondary leakage is not to exceed 540 gpd through any one SG.

   3. The operational leakage performance criterion is specified in LCO 3.4.5.2, "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 or repaired. Plugging or Repair is not applicable in the portion of the tube that is greater than 10.4 inches below the bottom of the expansion transition or top of the tubesheet, whichever is lower, to the tube end. Degradation detected between 10.4 inches below the bottom of the expansion transition or top of the tubesheet, whichever is lower, and the bottom of the expansion transition or top of the tubesheet, whichever is higher, shall be plugged on detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number an 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, 10.4 inches below the bottom of the hot leg expansion transition or top of the tubesheet, whichever is lower, completely around the U-bend to the top support of the cold leg 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 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 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (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.

f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. The following tube repair method is applicable:

   Defective tubes may be repaired in accordance with CENS Report CEN-605-P, "Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves," Revision 00-P, dated December 1992.

ADMINISTRATIVE CONTROLS

ANNUAL REPORTS (Continued)

(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 and 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 steady-state level; and

(5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

6.9.1.5 ~~DELETED~~

INSERT 8

Insert 8

STEAM GENERATOR TUBE SURVEILLANCE REPORT

6.9.1.5 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 or repaired during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged or repaired 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 and tube repairs in each SG, and

i. Repair method utilized and the number of tubes repaired by each repair method.

Attachment 3

W3F1-2005-0040

Proposed Technical Specification Bases Changes (mark-up)

SAFETY VALVES (Continued)

valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the overpressure protection system provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during reactor shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized while uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an SIAS test signal the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

The auxiliary pressurizer spray is used to depressurize the RCS by cooling the pressurizer steam space. The auxiliary pressurizer spray is used during those periods when normal pressurizer spray is not available, such as the later stages of a normal RCS cooldown. The auxiliary pressurizer spray also distributes boron to the pressurizer when normal pressurizer spray is not available.

The auxiliary pressurizer spray is used, in conjunction with the throttling of the HPSI pumps, during the recovery from a steam generator tube rupture accident. The auxiliary pressurizer spray is also used during a natural circulation cooldown as a safety related means of RCS depressurization to achieve shutdown cooling system initiation conditions and subsequent COLD SHUTDOWN per the requirements of Branch Technical Position (RSB) 5-1.

3/4.4.4 STEAM GENERATOR~~S~~ TUBE INTEGRITY

INSERT B-1

~~The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is~~

REACTOR COOLANT SYSTEM

BASES

~~STEAM GENERATORS~~ TUBE INTEGRITY (Continued)

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.

• (ORN 04-1243, Ch 38)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 75 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 75 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of the 75 gallon per day limit in Specification 3.4.5.2 will require plant shutdown and an unscheduled inspection, during which the leakage tubes will be located and plugged or repaired.

• (CR 04-1243, Ch 38)

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or sleeving will be required for all tubes with imperfections exceeding the plugging or repair limit as defined in Surveillance Requirement 4.4.4.4. Defective tubes may be repaired by sleeving in accordance with CENS Report CEN-605-P, "Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves," Revision 00-P, dated December 1992. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Sleeved tubes will be included in the periodic tube inspections for the inservice inspection program.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

REACTOR COOLANT SYSTEM.

BASES (continued)

Monitoring Containment Sump In-Leakage Flow

During automatic operation of the containment sump pumps (after a containment sump pump has operated), the flow calculation performed by the plant monitoring computer based on a level change will no longer be accurate since the level in the sump will be lowering. A 20 minute time period has been conservatively determined based on engineering calculations for this equipment operation. In addition, upon reboot of the plant monitoring computer, a period of 10 minutes is required for the leak rate calculation to become available. It has been determined these time periods (independent or combined) of calculation sump in-leakage flow inaccuracies, the instrumentation remains adequate to detect a leakage rate, or its equivalent, of one gpm in less than one hour; therefore, the containment sump level instrumentation and the corresponding flow calculation is considered to remain operable.

References

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45, Revision 0, dated May 1973.
3. UFSAR, Sections 5.2.5 and 12.3.

• •(DRN 04-1223, Ch. 33)

3/4.4.5.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from 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 allowable limit.

• (DRN 04-1243, Ch. 38)

The 75 gallon per day (gpd) per steam generator tube leakage limit ensures that the radiological consequences, including that from tube leakage, will be limited to the 10CFR50.67 limits for offsite dose and within the limits of General Design Criterion 19 for control room dose. For those analyzed events that do not result in faulted steam generators, greater than or equal to 75 gpd primary-to-secondary leakage per steam generator is assumed in the analysis. For those analyzed events that result in a faulted steam generator (e.g., MSLB), 540 gpd primary-to-secondary leakage is assumed through the faulted steam generator while greater than or equal to 75 gpd primary-to-secondary leakage is assumed through the intact steam generator.

• (DRN 04-1243, Ch. 38)

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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 is based on the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes a Loss of Offsite Power with subsequent releases to the atmosphere via Main Steam Safety Valves and Atmospheric Dump Valves.

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.) For those analyzed events that do not result in faulted steam generators, greater than or equal to 75 gpd primary to secondary leakage per steam generator is assumed in the analysis. For those analyzed events that result in a faulted steam generator (e.g., MSLB), 540 gpd primary to secondary leakage is assumed through the faulted steam generator while greater than or equal to 75 gpd primary to secondary leakage is assumed through the intact steam generator.

For accidents that do not involve fuel damage, the primary coolant activity level is assumed to be equal to the LCO 3.4.7 *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 and 10 CFR 50.67. 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 or repaired in accordance with the *Steam Generator Program.* During a SG inspection, any inspected tube that satisfies the *Steam Generator Program* repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged [or repaired], 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 and any repairs made to it, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

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 540 gpd 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 3.4.5.2, *RCS Operational leakage*, and limits primary to secondary leakage through any one SG to ≤ 75 gallons per day per SG.

Actions

The Actions are modified by a Note clarifying that the Actions may be entered independently 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 or repaired in accordance with the *Steam Generator Program* as required by SR 4.4.4.2. 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 or repaired 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 or repaired 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 4.4.4.1 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 4.4.4.1. 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 4.4.4.2 any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or 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.

Steam generator tube repairs are only performed using approved repair methods as described in the *Steam Generator Program*. Defective tubes may be repaired by sleeving in accordance with CENS Report CEN-605-P, *Waterford 3 Steam Generator Tube Repair Using Leak Tight Sleeves*, Revision 00-P, dated December 1992. The frequency of prior to entering HOT SHUTDOWN following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged or repaired prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

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

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67.
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."

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The primary to secondary leakage limit is based on the operational leakage performance criterion in NEI 97-06, *Steam Generator Program Guidelines*.

Attachment 4

W3F1-2005-0040

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) ONE-TIME ACTION	TYPE (Check one) CONT COMPL	SCHEDULED COMPLETION DATE (If Required)
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 Waterford-3 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**