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W3F1-2006-0007

February 15, 2006

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

SUBJECT: Supplement to Amendment Request NPF-38-262
Steam Generator Tube Inservice Inspection Program
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES: 1. Entergy letter dated July 21, 2005, *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* (W3F1-2005-0040)

Dear Sir or Madam:

By letter (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications (TSs) to replace the existing steam generator tube surveillance program with that being proposed by the Technical Specification Task Force in TSTF 449, Revision 4.

On January 3, 2006, Entergy received an NRC Staff request for additional information (RAI) to support the review of the proposed change. On January 19, 2006, Entergy and members of your staff held a call to clarify the additional information requested. During the conference call Entergy requested and received an extension to the Entergy response from 30 days to 45 days. Entergy's response is contained in Attachment 1.

Changes to the Analysis of proposed TS change, TS pages and TS Bases pages, which were originally submitted in Reference 1, are proposed. The revised mark-ups are included in Attachments 2, 3 and 4. Note that marked up TS Bases pages in Attachment 4 replace the pages provided in Attachment 3 of the original submittal (Reference 1) in their entirety.

The conclusions of the original no significant hazards consideration included in Reference 1 are not affected by any information contained in this supplemental letter. There are no new commitments contained in this letter.

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Entergy previously requested approval of the proposed amendment by February 1, 2006. However, this response to your Request for Additional Information and potential impact of the Waterford-3 proposed TS change regarding tubesheet inspection depth for SG tube inspection (reference 1 in the original proposed TS Request NPF-38-260) will prevent your staff from meeting this requested date. Therefore, Entergy requests NRC approval of the amendment by August 1, 2006 to satisfy our need to adopt TSTF-449, Revision 4 into Waterford 3's TS prior to the next refueling outage scheduled for November 2006.

If you have any questions or require additional information, please contact Ron Williams or Steve Bennett at (504) 739-6255 and 479-858-4626, respectively.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 15, 2006.

Sincerely,

T.E. Tankersley

TET/RLW/cbh

Attachments:
1. Response to Request for Additional Information
2. Analysis of Proposed TS Change Pages (Corrected Pages) Contained in Att.1 of Ref.1
3. Revised Markup of Corrected Technical Specification Pages
4. Revised Markup of Replacement Pages for All TS Bases Pages

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Attachment 1
To
W3F1-2006-0007

Response to Request for Additional Information

Response to Request for Additional Information

Question 1:

The licensee's Insert 1 (TS 3/4.4.4), corresponding to Technical Specification (TS) Task Force (TSTF) Section 3.4.18, does not have the NOTE stating "Separate Condition entry is allowed for each SG [steam generator] tube." This is under the limiting condition for operation (LCO) for steam generator tube integrity. This note is required for the LCO to be used as intended and described in the TSTF. Please discuss your plans to modify the TS accordingly, or provide justification for omitting it.

Response 1:

A similar NOTE to that referenced above is contained in Insert 2, first sentence. However, Waterford-3 (non-ITS plant) will rephrase the statement in Insert 2 to match the original TSTF 449 wording, with the exception that "Action" will be substituted for "Condition" as follows:

> Separate Action entry is allowed for each SG tube.

This change has been incorporated in the revised Insert 2 for TS 3/4.4.4 provided in Attachment 3.

Question 2:

In the proposed TS 6.5.9, "Steam Generator (SG) Program," paragraph 'f' refers to the installation of leak-tight sleeves for repairing defective tubes. Since the licensee's amendment request regarding the partial inspection of tubes within the hot-leg tubesheet (C*) did not address portions of sleeves extending outside the C* distance, the staff has asked for additional information about this issue as part of the C* review. The staff notes here that any changes made as a result of the C* review must be fully consistent with the TS changes proposed in this TSTF amendment.

Response 2:

Waterford 3 acknowledges that the NRC Staff has requested additional information via NRC letter dated October 25, 2005 related to proposed TS change regarding tubesheet inspection depth for SG tube inspections in which inspection of the leak-tight sleeve below the C* depth needs to be addressed. Waterford 3 will coordinate the proposed amendments to ensure consistency within the TSs.

Question 3:

In the proposed TS 6.5.9, "Steam Generator (SG) Program," Section 6.5.9.d does not include paragraph 1 from the corresponding TSTF-449 section (5 .5.9.d.1), which states, "Inspect 100% [percent] of the tubes in each SG during the first refueling outage following SG replacement." Please discuss your plans to include this statement in your TS.

Response 3:

Waterford-3 will incorporate the words "Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement" into the proposed TS section 6.5.9.d. This change has been incorporated in the revised Insert 7 for the new TS 6.5.9 provided in Attachment 3.

Question 4:

The second paragraph of the Insert B-1 Safety Analysis refers to the amount of leakage assumed from the faulted and unfaulted SGs. Please discuss your reason for assuming that leakage from unfaulted SGs will be greater than or equal to 75 gallons per day.

Response 4:

Certain events analyzed for offsite and control room dose which are documented in FSAR Chapter 15 involve releases via a secondary steaming pathway. The events analyzed using this pathway include Small Break LOCA (SBLOCA), Main Steam Line Break (MSLB), Feedwater Line Break (FWLB), and CEA Ejection. For such events, radiological releases occur from the release to the environment of secondary steam, due to plant cooldown and decay heat removal. The activity in the secondary steam is driven by the transport of activity from the Reactor Coolant System (RCS) via primary-to-secondary leakage, that is, via steam generator (SG) tube leakage.

In support of the 2005 Extended Power Uprate, Waterford-3 originally intended to assume a primary to secondary leak rate of 150 gallons per day (GPD) for the offsite and control room dose analyses consistent with NEI 97-06, *Steam Generator Program Guidelines*. During the course of the analyses, Waterford-3 determined that a 75 gpd leakage limit should be imposed to ensure acceptable control room dose results for the SBLOCA event due to the close proximity of one atmospheric dump valve to one control room air intake. Other event analyses were not revised to assume 75 gpd since acceptable dose consequences were achieved with the more conservative and higher 150 gpd leak rate. This is an added conservatism that increases the radioactivity release assumed in the analyses above what would actually occur due to the TS limit of 75 gpd. The results of these analyses were submitted as part of the Waterford-3 Alternative Source Term (AST) License Amendment Request and associated supplements (references 3 and 4 in the original proposed License Amendment Request NPF-38-262). AST License Amendment 198 was approved on March 29, 2005.

Furthermore, the 75 gpd limit is more restrictive than the 150 gpd per SG limit recommended in NEI 97-06. Thus, the Waterford-3 limit provides an acceptable defense in depth measure to limit the potential for tube rupture.

In summary, the use of a 150 gpd per SG assumption for SG tube leakage in most of the Waterford-3 radiological analyses is an added conservatism in the analysis. Adoption of such added conservatisms is an acceptable and common practice in safety analyses. The 75 gpd per SG Technical Specification limit is directly assumed for the SBLOCA event, which is the most limiting assumption of leakage with regard to the impact of tube leakage on radiological consequences.

Question 5:

Regarding Insert B-1, Section 3/4.4.4, STEAM GENERATOR TUBE INTEGRITY, the first paragraph of the Background section leaves out the last four sentences from the corresponding paragraph in the TSTF:

> The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB [reactor coolant pressure boundary], the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS [reactor coolant system] Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

Please provide your reason for omitting these sentences (with section numbers appropriate for Waterford 3 in your proposed bases.

Response 5:

Waterford-3 considered the four sentences extraneous information for a non-ITS plant. However, Waterford-3 will incorporate the four sentences into the proposed TS Bases section (replacement page 4 of 8 in Attachment 4) while substituting the applicable Waterford-3 specific non-ITS LCO sections for the applicable ITS sections. This change has been incorporated in the revised Insert B-1 for the revised TS Bases section 3/4.4.4 provided in Attachment 4.

Question 6:

In the section of Insert B-1 called "Limiting Condition for Operation," the final item on performance criteria addresses operational leakage. It does not include the following sentences in the TSTF (B.3.4.18) explaining the basis for an operational leakage criterion:

> This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR [steam generator tube rupture] under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Please discuss the reason for omitting these sentences.

Response 6:

Waterford-3 did not include these words in our original submittal (Ref. 1) since the above sentences referred to the 150 gpd recommendation from NEI 97-06, as opposed to the more restrictive Waterford-3 TS limit of 75 gpd per SG which was based upon dose consequence considerations. In response to the NRC's interest in retaining the two sentences from TSTF-449, Waterford-3 proposes to revise the third bullet to read:

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. This limit is based on assumptions in radiological analyses. This limit is less than the 150 gallons per day per SG limit of NEI 97-06, which assumes that a single crack leaking this amount would not propagate to a SGTR [under the stress conditions of a LOCA or a Main Steam Line Break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

This captures the intent of the TSTF wording while accurately characterizing the basis for the 75 gpd per SG limit in Waterford-3 TSs. This change has been incorporated in the revised Insert B-1 for TS Bases section 3/4.4.4.4 provided in Attachment 4.

Question 7:

In the proposed insert B-1, Action b (page 6 of Attachment 3) uses different language than the TSTF. This action refers to tubes that met the tube repair criteria but were not plugged or repaired according to the Steam Generator Program (new TS 6.5.9). The licensee's proposed wording is, "An allowed outage time of 7 days ...," while the TSTF states, "A completion time of 7 days ..." Please discuss the meaning of "allowed outage time" in this context, as well as why this wording is being used rather than the TSTF wording. Alternatively, please discuss your plans to revise the proposed specifications to adopt the TSTF language.

Response 7:

The term "allowed outage time" is defined in non-ITS plant TS Bases Specification 3.0.1 and is used to address an ACTION requirement that specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. The allowed outage time terminology is consistent with the definition of completion time as defined in ITS plant TS and maintains consistency with the Waterford-3 TS.

The use of "different language" proposed in Insert B-1 than the wording in the TSTF for Action "b" was to maintain consistency with the terminology used in Waterford-3 non-ITS TS and Bases. The reasons for the terminology use are as follows:

a. Terms such as "Required Actions" and "Completion Times" are not defined terms within the Waterford-3 TS and were replaced with "Actions" and "allowed outage time," respectively.
b. The use of the TS Bases phrase "... HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours" was to maintain consistency with TS Action "b" terminology as well as with the remaining Waterford-3 TS and Bases sections. The TSTF illustrates this consistency between the ITS plant TS and the Bases through the use of Modes (3 and 5) and Completion Times (6 hours and 36 hours - referenced to the situation time of discovery).

Based on the discussion above, the wording for TS Bases Action "b" should be retained.

Question 8:

The licensee's cover letter for the proposed TS changes discusses the inclusion of non-pressure loads into accident-induced leakage analyses. The staff notes that the Nuclear Energy Institute, in Enclosure 1 to a letter dated September 2, 2005, has now offered language that identifies this issue and NEI's position.

Response 8:

Waterford-3 initially developed a summarized version of the technical issue related to the effects of bending loads on SG tube leakage integrity and the position being discussed at the time by the industry and NRC with regard to it. This discussion was inserted into the Waterford-3 cover letter for reference 1 to address this on-going technical issue as a separate open question from the adoption and implementation of TSTF-449 prior to the development of the generic wording offered in the NEI letter dated September 2, 2005. With the issuance of agreed-upon language that identifies the issue and the industry position to support licensee amendment requests, Waterford-3 includes the following discussion as part of the license amendment request.

> The industry is currently evaluating a technical issue related to the Accident Induced Leakage Performance Criterion (AILPC) specified in Section 6.5.9.b.2 of our proposed technical specifications. The issue concerns the consideration of non-pressure (bending) loads on the accident induced leak rates of steam generator tubes (axial differential thermal loads are routinely considered in assessing accident induced leakage). The EPRI Steam Generator Management Program (SGMP) is conducting a study to determine if bending loads are significant, and if they are, to define how to account for the loads in steam generator tube integrity assessments. In the interim, as this study is being completed, EPRI has completed a preliminary impact assessment. The assessment (Preliminary Assessment of the Impact of Non-Pressure Loads on Leakage Integrity of Steam Generator Tubing) found that the effect of the loads in question may, in certain circumstances, initiate primary-to-secondary leakage, or increase pre-existing primary-to-secondary leakage during and after load application. The effort also assessed the effect of such loads in combination with the applicable design basis accident. The results indicate that these circumstances are expected to be limited to the presence of significant circumferential cracks located in high bending stress regions of tubing. As of this date, such degradation has not been observed in the Industry.
>
> The structural integrity impact of non-pressure loads on degraded steam generator tubes has been well-documented in a previous EPRI report (NRC accession number ML050760208) related to the revised Structural Integrity Performance Criterion (SIPC). Experimental results indicated that neither axial loads nor bending loads have a significant effect on the burst pressure of tubing with axial degradation. Similarly, these loads are considered inconsequential for axially oriented degradation with respect to localized pop-through conditions and corresponding accident leakage. As such, industry evidence indicates that the only meaningful impact of non-pressure loads with respect to leakage are due to the application of bending moments on circumferential cracking.
>
> The EPRI Preliminary Assessment found that high bending loads that could affect the leakage analysis are only present in the top span region in the original design of once-

through steam generators and in the U-bend region of large-radius tubes in some recirculating steam generators. The high bending loads in the OTSGs are a consequence of crossflow during a steam line break whereas the high bending loads in the recirculating steam generators are a result of a seismic event.

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

Based on the above, Entergy believes that the effect of bending loads is not safety significant for Waterford-3 with respect to leakage integrity given the expected effect and existing margins with respect to degradation type, susceptible location and allowable flaw size.

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

Question 9:

Please note that TS 6.5.9 .d contains a typographical error in the second sentence: "The number an portions of the tubes inspected ... ." The word "an" should be "and."

Response 9:

Waterford-3 will correct the typographical error identified in TS 6.5.9 (Insert 7-page 17 of 19 in Attachment 2 of the original submittal). In addition, Entergy discovered that TS 6.9.1.5 (Insert 8-page 19 of 19 in Attachment 2) contained an editorial error in the TS title *Steam Generator Tube Surveillance Report*. The word "Surveillance" should be "Inspection."

Also other minor editorial changes were identified in the original submittal. Although not technical in nature, Entergy is providing the following changes for clarification.

- Description Section 1.0 on page 1 of 25 in Attachment 1
  - Use of an incomplete title of Specification 3.4.5.2. Insert the word "RCS" between "Specification 3.4.5.2," and "Operational Leakage."
  - Use of the incorrect title of TS section 6.9.1.5 *Steam Generator Tube Surveillance Report*. The word "Surveillance" should be "Inspection."

- SG Performance Criteria, Section 9 on page 14 of 25 in Attachment 1

- In the description of the accident induced leakage performance criterion, delete "...[1 gpm] per SG, [except for specific types of degradation of specific locations as described in paragraph c of the Steam Generator Program" and replace with " 540 gpd per SG" to be consistent with the intent of TSFT-449 and the use of the gpd value in the TS Bases.
- In the description of the operational leakage performance criterion, delete "... per SG" to be consistent with TSFT-449.

The TS changes specified above have been incorporated in TS sections 6.5.9 .d and 6.9.1.5, respectively and revised marked up pages provided in Attachment 3.

The other editorial changes specified above have been incorporated in the applicable pages of the Analysis of Proposed Technical Specification Change and annotated with a revision bar in the right hand column. Revisions of the affected pages of Attachment 1 of the original submittal are provided in Attachment 2.

Attachment 2
To
W3F1-2006-0007

Analysis of Proposed Technical Specification Change (Corrected Pages)
Contained in Attachment 1 of Reference 1

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, *RCS Operational Leakage.* In addition, a new Specification 6.5.9, *Steam Generator Program,* and a new Specification 6.9.1.5, *Steam Generator Tube Inspection 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.

stress corrosion cracking for verifying a safety factor of three against burst. Additionally, the 3ΔP 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 3ΔP 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 540 gpd per SG.*

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

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.

Attachment 3
To
W3F1-2006-0007

Revised Markup of Corrected Technical Specification Pages

Note

TS 3/4.4.4 Insert 2 (page 10 of 19 in Att. 2 of Ref.1);
TS 6.5.9 Insert 7 (page 17 of 19 in Att. 2 of Ref.1);
and TS 6.9.1.5 Insert 8 (page 19 of 19 in Att. 2 of Ref.1)
are affected by this submittal. For all other affected
TSs, see original submittal.]

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)

Separate Action entry is allowed for each SG tube.

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.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, 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% [percent] of the tubes in each SG during the first refueling outage following SG replacement.

   2. 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.

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

e. *Provisions for monitoring operational primary to secondary leakage.*

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.

Insert 8

STEAM GENERATOR TUBE INSPECTION 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 4
To
W3F1-2006-0007

Revised Markup of Replacement Pages for All Technical Specification Bases Pages

For Information Only

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 (Continued) TUBE INTEGRITY

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.

• (DRN 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.

• (DRN 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. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.1.1, "RCS Loops - MODES 1 and 2," LCO 3.4.1.2, "RCS Loops - MODE 3**," LCO 3.4.1.3, "RCS Loops - MODE 4," and LCO 3.4.1.4, "RCS Loops - MODE 5 with reactor coolant loops filled**."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements. Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 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. This limit is based on assumptions in radiological analyses. This limit is less than the 150 gallons per day per SG limit of NEI 97-06, which assumes that a single crack leaking this amount would not propagate to a SGTR [under the stress conditions of a LOCA or a Main Steam Line Break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

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.*

Insert B-2

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