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Entergy

Tom Tankersley
Acting Director, Nuclear Safety Assurance
Waterford 3

W3F1-2006-0016

May 3, 2006

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

SUBJECT: Supplement 2 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 March 15, 2005, *License Amendment Request NPF-38-260 Proposed Technical Specification Change Regarding Tubesheet Inspection Depth for Steam Generator Tube Inspections* (W3F1-2005-0009)
2. 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)
3. Entergy letter dated February 15, 2006, *Supplement to Amendment Request NPF-38-262 Steam Generator Tube Inservice Inspection Program* (W3F1-2006-0007)
4. Entergy letter dated March 22, 2006, *Tubesheet Inspection Depth for Steam Generator Tube Inspections Waterford Steam Electric Station, Unit 3* (W3F1-2006-0008)

Dear Sir or Madam:

By letter (Reference 2), 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 and discuss an extension to the Entergy RAI response from 30 days to 45 days. On February 15, 2006, Entergy provided a response to the RAI (Reference 3).

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On April 17, 2006, Entergy received a second NRC Staff RAI dated March 31, 2006 to support the review of the proposed change. On April 25, 2006, Entergy discussed with members of your staff our desire to have the proposed TSTF-449 modeled TS change (Reference 2 supplemented by Reference 3) approved prior to the tubesheet inspection depth proposed TS change (C*) to simplify and expedite the review process. This decision will necessitate the removal of references to C* from the proposed specification. RAI questions related to the C* will be addressed in the proposed C* TS change (Reference 1 supplemented by Reference 4). Additionally, Entergy had included the already approved welded sleeve alternate repair method from the existing TSs to the proposed TSs in accordance with TSTF-449. However, due to subsequent NRC staff questions related to the use of and inspection techniques for the sleeving repair methodology and with this repair method not being applied at Waterford-3, Entergy will remove this method from this proposed TS change. Entergy's response to this RAI is contained in Attachment 1.

Changes to the TS pages and TS Bases pages, which were originally submitted in Reference 2 and supplemented by References 3 and 4, are proposed. The revised mark-ups are included in Attachments 2 and 3. Note that marked up TS pages in Attachment 2 replace the pages provided in Attachment 2 of Reference 2 and supplemented by Attachment 3 of Reference 3 and Attachment 6 of Reference 4 in their entirety. Note that marked up TS Bases pages in Attachment 3 replace the pages provided in Attachment 4 of Reference 3 in their entirety.

The conclusions of the original no significant hazards consideration included in Reference 2 are not affected by any information contained in this supplemental letter. There are no new commitments contained in this letter.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 3, 2006.

Sincerely,

T.E. Tankersley

TET/RLW

Attachments:
1. Response to Request for Additional Information
2. Revised Markup of Replacement Pages for All TS Pages
3. Revised Markup of Replacement Pages for All TS Bases Pages

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Attachment 1
To
W3F1-2006-0016

Response to Request for Additional Information

Response to Request for Additional Information

Question 1:

Currently, no sleeves are installed in the Waterford Steam Electric Station, Unit 3 (Waterford-3) steam generators (SGs); however, proposed Technical Specification (TS) 6.5.9.f allows the use of sleeving (CENS Report CEN-605-P, "Steam Generator Tube Repair Using Leak Tight Sleeves"). It is the staff's understanding that the tubesheet sleeves, as described in CEN-605-P, have a nickel band in the area of the rolled joint. Based on interactions with other plants, it is not clear whether techniques currently exist to inspect the parent tube located behind (adjacent to) the nickel band for crack-like indications. If this is the case, it is not clear how you will implement proposed TS 6.5.9 .d, which requires that the method of inspection should be capable of detecting flaws of any type that may be present along the length of the tube and that may satisfy the applicable tube repair criteria. In light of the above, either (a) discuss your plans for removing this sleeving method from your TSs, (b) provide information supporting the ability of an inspection technique to detect the forms of degradation that could occur in the parent tube adjacent to the nickel band and that may satisfy the applicable tube repair criteria, or (c) provide analysis and/or testing results which indicate that inspection of this region (i.e., behind the nickel band) is not needed.

Response 1:

Currently no sleeves are installed in the Waterford-3 SGs and plans are not to install any using this tube repair methodology. Therefore, this tube repair method will be removed from these proposed TS changes.

Question 2:

Proposed TS 6.5.9.d excludes from inspection the portion of each tube from the top support of the cold leg to the cold-leg tube end. This is inconsistent with the corresponding section of the Technical Specification Task Force (TSTF)-449 (5.5.9.d), which states the objective of tube inspection is to detect flaws of any type, "from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet." Please discuss your plans to modify the proposed TS to make them consistent with TSTF-449.

Response 2:

As discussed in the cover letter regarding the agreement to have the TSTF-449 format change approved prior to the tubesheet inspection depth change, Entergy agrees to modify the proposed TS to make them consistent with the wording in TSTF-449, 5.5.9.d. The revised TS pages for this proposed license amendment are contained in Attachment 2.

Question 3:

Proposed TS 6.5.9.d, states, "In addition to meeting the requirements of d.1 and d.2 below, ..." To be consistent with TSTF-449, this should read "... requirements of d.1, d.2, **and d.3** below," since your February 15, 2006 response to Request for Additional Information (RAI) question 3 added a paragraph that was missing from the original submittal. Please discuss your plans to modify the proposed TSs to make them consistent with TSTF-449. (Emphasis added by the staff.)

Response 3:

This was an editorial oversight in the last RAI response. A corrected page is being provided in Attachment 2.

Question 4:

Proposed TS 6.5.9.c addresses SG tube repair criteria. Since a tube is defined as the entire length of the tube, including the tube wall and any repairs to it, it could be construed that the 40% plugging limit is applicable to the sleeves. Please discuss your plans to incorporate the repair criteria for the sleeves into the specification. For example,

> In the region of a tube repaired in accordance with TS 6.5.9.f, the tube shall be plugged upon detection of any service-induced flaw in (a) the sleeve or (b) the pressure boundary portion of the original tube wall in the sleeve-to-tube joint.

Response 4:

In Entergy's response to RAI 9 received on the SG tubesheet depth submittal (Reference 4), Entergy made the following commitment:

> *If sleeves are installed, Entergy plans to inspect inservice sleeves over their full length plus 5 inches beyond the sleeve-to-tube rolled joint in the tube sheet in accordance with the requirements of the EPRI Guidelines using appropriate examination methodology. The tube shall be plugged upon detection of any service induced imperfection, degradation or defect in the sleeve or pressure boundary portion of the original tube wall in the sleeve-to-tube rolled joint. Entergy will periodically inspect sleeves as a minimum in accordance with the existing TS requirements*

As discussed in the response to RAI question 1 above regarding the removal of the sleeving tube repair method from the TSs, this commitment no longer applies and therefore will be withdrawn.

Question 5:

Proposed TS Bases Insert B-2 includes only the first sentence of a paragraph from the corresponding TSTF-449 insert (B 3.4.13B). Missing from the proposed Waterford insert is the following:

> The Steam Generator Program operational LEAKAGE performance criterion in NEI [Nuclear Energy Institute] 97-06 states, "The RCS [Reactor Coolant System] operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

In place of this paragraph you have a plant-specific discussion of operational leakage limits. The staff recognizes that the 75 gallons per day (gpd) operational leakage limit at Waterford-3 ensures the radiological consequences will be limited to the appropriate regulatory limits. However, this limit also reflects operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion (since it is less than 150 gpd through any one SG) in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of SG tube ruptures. Please discuss your plans for modifying your Bases to include the other reason for the operational leakage limit. The staff notes that from the Bases as currently proposed, one may incorrectly conclude 540 gpd is an appropriate operational leakage limit for a "faulted steam generator."

Response 5:

Entergy did not believe that the quote from NEI 97-06 added substantial value in light of the reduced operational leakage limit of 75 gpd. However, for completeness, Entergy will include the remainder of the TSTF-449 proposed TS Bases 3.4-13B into the Waterford-3 Bases insert for 3/4.4.5.2. The proposed Insert B-2 into Waterford-3 Bases will now read:

> The primary to secondary leakage limit of 75 gallons per day through any one SG is based on the operational leakage performance criterion in NEI 97-06. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The NEI 97-06 limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

Regarding the existing TS Bases 3/4.4.5.2, OPERATIONAL LEAKAGE, Entergy agrees that this discussion could lead one to believe that the assumptions in the accident analysis could be applicable to operational leakage based on the heading of the section. The same accident analysis assumptions are contained in Insert B-1 which is being included in the Bases for TS 3/4.4.4. Therefore, Entergy will remove the existing discussion in TS Bases 3/4.4.5.2.

A revised TS Bases and Insert B-2 for 3/4.4.5.2 is contained in Attachment 3.

Question 6:

In the Limiting Condition for Operation section of your BASES Section 3/4.4.4, "STEAM GENERATOR TUBE INTEGRITY", the reference to Regulatory Guide 1.121 is omitted from the bullet dealing with the structural integrity performance criterion (i.e., where Subsection NB of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code is referenced). Since Regulatory Guide 1.121 was used in the development of the structural integrity performance criterion, it is not clear why it is not referenced. Please discuss your plans to modify your proposal to address this comment.

Response 6:

The new structural integrity analysis that is being performed for Waterford-3 supersedes the typical analysis performed per draft RG 1.121 and therefore was not initially included. However, since the structural integrity analysis incorporates approaches and methodologies from RG 1.121, the reference to draft RG 1.121 will be added to the Bases. A revised TS Bases Insert B-1 is contained in Attachment 3.

Question 7:

You included a commitment in Attachment 4 indicating all loads that can significantly affect burst or collapse will be determined and assessed. In this commitment, there is a statement that indicates: "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."

The NRC staff is aware of the industry's efforts to assess the effects of non-pressure loads on tube integrity (structural and leakage integrity). These efforts include an assessment of whether changes are needed to the industry guidelines to ensure these loads are appropriately accounted for in tube integrity evaluations (i.e., in the methods used to determine whether the performance criteria have been exceeded).

However, your statements seem to imply that the on-going industry efforts may affect the 40% tube plugging limit. The reason for this is not clear since the 40% plugging limit was developed with consideration of non-pressure loads (consistent with the guidance in Regulatory Guide 1.121). Please clarify the meaning of your commitment which should include a determination of whether it is needed.

Response 7:

Entergy believes the intent of this commitment has been misinterpreted by the NRC. At the time of the submittal, Entergy had not performed the new structural integrity analysis to comply with NEI 97-06. The only intent of this commitment is to state that the structural integrity and plugging limit calculation would be completed prior to implementation of the TS amendment. It is believed that the analysis results can be incorporated into the Waterford-3 licensing basis under 10CFR50.59 and should not require NRC review and approval.

Question 8:

A separate license amendment request to apply the C* inspection criterion at Waterford-3 was submitted on March 15, 2005 and is still under NRC staff review. This would require tube inspection to a depth of 10.4 inches below the top of the hot-leg tubesheet or hot-leg expansion transition, whichever is lower. If your C* amendment is approved before the TSTF amendment, it may be necessary to amend the specifications in your TSTF amendment. Similarly, if you desire approval of the TSTF amendment before approval of the C* amendment, it will be necessary to remove references to C* from the specifications.

The following question was included in RAI question 9 about your C* amendment proposal. The staff notes that this will need to be addressed before the C* criterion can be incorporated into your proposed TSs modeled after TSTF-449.

> The Waterford[-3] technical specifications (4.4.4.4.b) currently allow installation of leak-tight sleeves according to CENS Report CEN-605-P. Since sleeves could extend into the tubesheet below the C* distance, the proposed technical specifications would not require an inspection of this portion of the sleeve (including the lower sleeve joint.) Sleeves were not addressed in the testing and analysis used to justify excluding part of the tube from inspection (WCAP-16208-P, Rev. 1). What plans do you have to modify the technical specifications to ensure the lower ends of sleeves (i.e., those within the tubesheet below the C* distance) will be inspected?

Response 8:

As discussed in the cover letter and responses to RAI questions 1 and 4 above, the sleeving tube repair method will be removed from these proposed TS changes. Therefore, the need for inspection of these sleeves no longer applies.

Question 9:

In your proposed TS 3.4.5.2.c under OPERATIONAL LEAKAGE for the RCS, the primary-to-secondary limit is 75 gpd <u>**per SG**</u>. The wording in TSTF-449 and in your proposed accident-induced leakage performance criterion (TS 6.5.9.b.2) is "<u>**through any one**</u>" SG. Please discuss your plans for modifying your proposed TS to make the wording of your leakage limits fully consistent with your performance criteria and the TSTF. (Emphasis added by the staff.)

Response 9:

The term "per SG" was used in several locations in the existing TSs and its usage was carried over to the proposed TSs. However, for consistency Entergy has made changes, where appropriate, to use the term "through any one SG". The appropriately revised TS pages are contained in Attachment 2 and the appropriately revised TS Bases pages are contained in Attachment 3.

Question 10:

Proposed TS 6.5.9.b.3, the operational leakage performance criterion, refers to Limiting Condition for Operation 3.4.5.2 as "Operational Leakage." The wording used in your proposed TS 3.4.5.2 is "Reactor Coolant System Operational Leakage," and the TSTF-449 wording is "**RCS** Operational Leakage." Please discuss how you will modify your proposed TS to make them consistent with either your existing wording or the TSTF wording. (Emphasis added by the staff.)

Response 10:

"Reactor Coolant System" is used in Waterford-3 TS LCO 3.4.5.2 when referring to operational leakage. Therefore, Entergy will correct references of "RCS operational leakage" or "operational leakage" to "Reactor Coolant System operational leakage". The appropriately revised TS pages are contained in Attachment 2 and the appropriately revised TS Bases pages are contained in Attachment 3.

Question 11:

In your February 15, 2006, response to RAI question1, you proposed changes to the ACTION section of TS 3/4.4.4, "Steam Generator (SG) Tube Integrity." Paragraph a.1 of the proposed insert states:

> Within 7 days verify tube integrity of the affected tube(s) is maintained until the next **inspection**, (Emphasis added by the staff.)

The corresponding section of the TSTF states:

> Within 7 days verify tube integrity of the affected tube(s) is maintained until the next **refueling outage or SG tube inspection**. (Emphasis added by the staff.)

The TSTF wording could eliminate the need to shut down the facility in the event that tube integrity is only maintained until a refueling outage and not until the next SG tube inspection. Please discuss your plans to revise your proposed TS to make them consistent with the TSTF

Response 11:

Action a.1 of Insert 2 for TS 3/4.4.4 will be revised to:

> Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.

The appropriately revised TS pages are contained in Attachment 2.

Question 12:

On page 6 of 8 of Attachment 4 in your February 15, 2006 RAI response, the final bullet under "Limiting Condition for Operation" discusses operational leakage. The staff notes there appears to be an unnecessary bracket in the next-to-last sentence between "SGTR" [steam generator tube rupture] and "under." Please delete this bracket, or provide the missing information and closing bracket you intended to include.

Response 12:

The bracket has been removed and the revised Insert B-1 for TS Bases 3/4.4.4 is contained in Attachment 3.

Attachment 2
To
W3F1-2006-0016

Revised Markup of Replacement Pages for All TS Pages

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

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary ~~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 GENERATORS~~ (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 selected for these inspections shall be selected on a random basis except:~~

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

~~b. The first sample of tubes selected for each inservice inspection (subsequent to the 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 ... bjected 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

SURVEILLANCE REQUIREMENTS (Continued)

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.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

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.

$S = \frac{3}{n}\%$ Where n is the number of steam generators inspected during an inspection

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 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 in accordance with the Steam Generator Program,

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

b. If the required Action and Allowed Outage Time of Action a. above cannot be met or the SG tube integrity cannot be maintained, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN with the following 30 hours.

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 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

Operational

3.4.5.2 Reactor Coolant System 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,~~ through any one SG

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.

Operational

b. With any Reactor Coolant System leakage ~~greater~~ than any one of the limits, excluding PRESSURE BOUNDARY 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. primary to secondary leakage

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)

4.4.5.2.~~2~~ 3 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.

4.4.5.2.~~3~~ 4 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 through any one 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 of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

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

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

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

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

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet 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.

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.

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 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 during the inspection outage for each active degradation mechanism,

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

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

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

Attachment 3
To
W3F1-2006-0016

Revised Markup of Replacement Pages for All Technical Specification Bases Pages

For Information Only

REACTOR COOLANT SYSTEM

BASES

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

TUBE INTEGRITY

STEAM GENERATORS (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.

• (ORNL 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.

• (ORNL 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)~~

INSERT B-2

Insert B-1

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 in accordance with the *Steam Generator Program*. During a SG inspection, any inspected tube that satisfies the *Steam Generator Program* repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity. In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.5.9, *Steam Generator Program*, and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

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

  Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and Draft Regulatory Guide 1.121.

- 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, *Reactor Coolant System operational leakage*, and limits primary to secondary leakage through any one SG to ≤ 75 gallons per day. This limit is based on assumptions in radiological analyses. This limit is less than the 150 gallons per day through any one 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.

<u>Actions</u>

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 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 has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Action "b" applies.

An allowed outage time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity. If the evaluation determines that the affected tube(s) have tube integrity, Action a.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This time period is acceptable since operation until the next inspection is supported by the operational assessment.

Action "b" applies if the actions and associated allowed outage time of Action "a" are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. The allowed outage time are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements

During shutdown periods the SGs are inspected as required by SR 4.4.4.1 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 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.

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 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 of 75 gallons per day through any one SG is based on the operational leakage performance criterion in NEI 97-06. The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The NEI 97-06 limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion (since it is less than 150 gpd through any one SG) in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures