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CP-201301065  
Log # TXX-13132

Ref. # 10 CFR 50.90

August 29, 2013

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

**SUBJECT:** COMANCHE PEAK NUCLEAR POWER PLANT (CPNPP)  
DOCKET NOS. 50-445 AND 50-446  
LICENSE AMENDMENT REQUEST 13-002  
REVISION TO APPLICATION TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT  
TSTF-510, "REVISION TO STEAM GENERATOR PROGRAM INSPECTION  
FREQUENCIES AND TUBE SAMPLE SELECTION," USING THE CONSOLIDATED  
LINE ITEM IMPROVEMENT PROCESS

**REFERENCE:** 1. Letter logged TXX-130197, dated July 10, 2013, regarding License Amendment  
Request 13-002, from Luminant Power to the NRC (ML13205A172)  
2. Letter logged TSTF-12-09, dated March 28, 2012, regarding correction to TSTF-510-A,  
Revision 2, from Technical Specifications Task Force to the NRC (ML12088A082)

Dear Sir or Madam:

Pursuant to 10CFR50.90, Luminant Generation Company LLC (Luminant Power) hereby requests an amendment to the CPNPP Unit 1 Operating License (NPF-87) and CPNPP Unit 2 Operating License (NPF-89) by incorporating the attached changes into the CPNPP Unit 1 and 2 Technical Specifications (TSs). This change request applies to both Units.

The proposed amendment would modify TS requirements regarding steam generator tube inspections and reporting as described in TSTF-510-A, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

This submittal revises Reference 1 by incorporating the changes in Reference 2 consistent with TSTF-510, Revision 2.

Attachment 1 provides a description and assessment of the proposed changes, the requested confirmation of applicability, and plant-specific verifications. Attachment 2 provides the affected Unit 1 and Unit 2 Technical Specification (TS) pages marked-up to reflect the proposed changes. Attachment 3 provides retyped TS pages which incorporate the requested changes, including pages that were renumbered due to the insertion of additional text. Attachment 4 provides existing TS Bases pages marked up to show the proposed changes. Attachment 5 provides retyped TS Bases pages which incorporate the proposed changes. The TS Bases pages are provided to the NRC for information only and do not require NRC approval.

A member of the STARS Alliance

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*received  
9/11/2013*

*Designated Original  
per Balwant Singal*

*ADDI  
NRR*

The proposed changes are consistent with NRC approved Revision 2 to Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection." The availability of this TS improvement was announced in the Federal Register on October 27, 2011, (76 FR 66763) as part of the consolidated line item improvement process (CLIP).

Luminant Power requests approval of the proposed license amendment by January 10, 2014, to support implementation during the CPNPP Unit 2 Spring 2014 (2RF14) refueling outage. Once approved, the amendment shall be implemented within 90 days for CPNPP.

This communication contains no new licensing basis commitments regarding CPNPP Units 1 and 2.

In accordance with 10 CFR 50.91(b), Luminant Power is providing the State of Texas with a copy of the proposed license amendment.

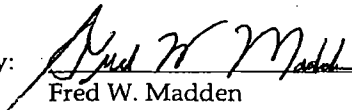
Should you have any questions, please contact Mr. Jack Hicks at (254)897-6725.

I state under penalty of perjury that the foregoing is true and correct. Executed on the 29<sup>th</sup> of August, 2013.

Sincerely,

Luminant Generation Company LLC

Rafael Flores

By:   
Fred W. Madden  
Director, Oversight & Regulatory Affairs

- Attachments -
1. Description and Assessment
  2. Proposed Technical Specification Changes (Markup)
  3. Retyped Technical Specification Pages
  4. Proposed Technical Specification Bases Changes (Markup For Information Only)
  5. Retyped Technical Specification Bases Changes (For Information Only)

c - S. A. Reynolds, Region IV  
B. K. Singal, NRR  
Resident Inspectors, CPNPP

Robert Free  
Texas Department of State Health Services  
Austin, TX 78714-9347

## **DESCRIPTION AND ASSESSMENT**

### **1.0 DESCRIPTION**

### **2.0 ASSESSMENT**

2.1 Applicability of Published Safety Evaluation

2.2 Optional Changes and Variations

### **3.0 REGULATORY ANALYSIS**

3.1 No Significant Hazards Consideration Determination

### **4.0 ENVIRONMENTAL EVALUATION**

## **1.0 DESCRIPTION**

Pursuant to 10 CFR 50.90, Luminant Generation Company, LLC (Luminant Power) hereby requests an amendment to the Comanche Peak Nuclear Power Plant (CPNPP) Unit 1 and Unit 2 Technical Specifications. The proposed change revises Technical Specifications (TS) 3.4.17, "Steam Generator (SG) Tube Integrity," TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program", and TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report." The proposed changes are needed to address implementation issues associated with inspection periods, and address other administrative changes and clarifications.

The proposed amendment is consistent with TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

Approval of this amendment application is requested by January 10, 2014, to support CPNPP Unit 2 Refueling Outage 14 (Spring 2014).

## **2.0 ASSESSMENT**

### **2.1 Applicability of Published Safety Evaluation**

Luminant Generation Company LLC (Luminant Power) has reviewed TSTF-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection.," (ADAMS Accession No. ML110610350) and the Model Safety Evaluation dated October 19, 2011 (ADAMS Accession No. ML112101513) as identified in the Federal Register Notice of Availability, dated October 27, 2011 (76 FR 66763). As described in the subsequent paragraphs, Luminant Power has concluded that the justifications presented in TSTF-510 and the Model Safety Evaluation prepared by the Nuclear Regulatory Commission (NRC) staff is applicable to CPNPP Units 1 and 2 and justify this amendment for incorporation of the changes to the CPNPP TS.

### **2.2 Optional Changes and Variations**

Luminant Power is proposing the following variations from the TS changes described in TSTF-510, Revision 2, or the applicable parts of the NRC staff's model safety evaluation dated October 27, 2011.

The CPNPP TS utilize different numbering and titles than the Standard Technical Specifications on which TSTF-510 was based. Specifically, the CPNPP TS title for Section 5.5.9 is "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program," to denote that the CPNPP SGs are different models. TSTF-510 TS Section 5.5.9.d.2 for inspections after the first refueling outage following SG installation has been divided in the CPNPP TS as Section 5.5.9.d.2 for the CPNPP Unit 2 model D5 steam generators and Section 5.5.9.d.3 for the CPNPP Unit 1 model Delta-76 steam generators. TSTF-510 TS Section 5.5.9.d.3 is CPNPP TS Section 5.5.9.d.4 for crack indications. These differences are administrative and do not affect the applicability of TSTF-510 to the CPNPP TS.

The proposed change corrects an administrative inconsistency in TSTF-510, Paragraph d.2 of the Steam Generator Tube Inspection Program. In Section 2.0, "Proposing Change," TSTF-510 states that references to "tube repair criteria" in Paragraph d is revised to "tube plugging [or repair] criteria." However, in the following sentence in Paragraph d.2, this change was

inadvertently omitted, "If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at the location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated" (Emphasis added).

Luminant Power does not have an approved tube repair criteria. Therefore the sentence is revised to state "tube plugging criteria", consistent with TSTF-510, Revision 2.

### 3.0 REGULATORY ANALYSIS

#### 3.1 **No Significant Hazards Consideration Determination**

Luminant Power requests adoption of an approved change to the standard technical specifications (STS) into the plant specific Technical Specifications (TS) for CPNPP, Units 1 and 2, to revise TS 3.4.17, "Steam Generator (SG) Tube Integrity," TS 5.5.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program", and TS 5.6.9, "Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report," to address inspection periods and other administrative changes and clarifications.

As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

**1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No

The proposed change revises the Steam Generator (SG) Program to modify the frequency of verification of SG tube integrity and SG tube sample selection. A steam generator tube rupture (SGTR) event is one of the design basis accidents that are analyzed as part of a plant's licensing basis. The proposed SG tube inspection frequency and sample selection criteria will continue to ensure that the SG tubes are inspected such that the probability of a SGTR is not increased. The consequences of a SGTR are bounded by the conservative assumptions in the design basis accident analysis. The proposed change will not cause the consequences of a SGTR to exceed those assumptions.

Therefore, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

**2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No

The proposed changes to the Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The proposed change does not affect the design of the SGs or their method of operation. In addition, the proposed change does not impact any other plant system or component.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

**3. Does the proposed change involve a significant reduction in a margin of safety?**

Response: No

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes. Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change will continue to require monitoring of the physical condition of the SG tubes such that there will not be a reduction in the margin of safety compared to the current requirements.

Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

Based on the above evaluations, Luminant Power concludes that the proposed amendment presents no significant hazards under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

**4.0 ENVIRONMENTAL CONSIDERATIONS**

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

**ATTACHMENT 2 TO TXX-13132**

**PROPOSED TECHNICAL SPECIFICATION CHANGES (MARKUP)**

**Page 3.4-44**

**Page 3.4-45**

**Page 5.5-5**

**Page 5.5-6**

**Page 5.5-7**

**Page 5.6-5**

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

plugging

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

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

plugging

NOTES

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube <del>repair</del> criteria and not plugged <del>or repaired</del> in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.  <u>AND</u> A.2 Plug <del>or repair</del> the affected tube(s) in accordance with the Steam Generator Program.	7 days  Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.  <u>OR</u>  SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>  B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube <del>repair</del> criteria is plugged <del>or repaired</del> in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

plugging

5.5 Programs and Manuals (continued)

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5.5.9 Unit 1 Model D76 and Unit 2 Model D5 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. Leakage is not to exceed 1 gpm per SG.
  3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

5.5 Programs and Manuals

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

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.

plugging

1. The following alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

a. For Unit 2 only, tubes with service-induced flaws located greater than 14.01 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 14.01 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. For Unit 1, 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. For Unit 2, 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 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side 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 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.

plugging

A

assessment

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

installation

2. For the Unit 2 model D5 steam generators (Alloy 600 thermally treated) inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling

INSERT 1

5.5 Programs and Manuals

5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

~~outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.~~

3. For the Unit 1 model Delta-76 steam generators (Alloy 690 thermally treated) inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

INSERT 2

affected and potentially affected

4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indications shall not exceed 24 effective full power months or one refueling outage (whichever is less). For Unit 2, if crack indications are found in any SG tube from 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side, then the next inspection for each SG for the degradation mechanism that caused the crack indications 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.

results in more frequent inspections

affected and potentially affected

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;

## INSERT 1

after the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

- a. After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;
- b. During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and
- c. During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

## INSERT 2

after the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

- a. After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
- b. During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
- c. During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
- d. During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

5.6 Reporting Requirements

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5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

1. WCAP-14040-NP-A; "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used

5.6.8 PAM Report

When a report is required by the required actions of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report

D A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The 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, and
- g. The results of condition monitoring, including the results of tube pulls and in-situ testing,
- h. For Unit 2, the primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to

The

the effective plugging percentage in each steam generator,

**ATTACHMENT 3 TO TXX-13132**

**RETYPE TECHNICAL SPECIFICATION PAGES**

**Page 3.4-44**

**Page 3.4-45**

**Page 5.5-5**

**Page 5.5-6**

**Page 5.5-7**

**Page 5.5-8**

**Page 5.5-9 through Page 5.5-18 (For Information Only)**

**Page 5.6-5**

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

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

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

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Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	<u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.  <u>OR</u>  SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>  B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals (continued)

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5.5.9 Unit 1 Model D76 and Unit 2 Model D5 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:

- 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), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
    2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.
    3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."
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## 5.5 Programs and Manuals

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### 5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

- c. Provisions for SG tube plugging 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.
1. The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:
    - a. For Unit 2 only, tubes with service-induced flaws located greater than 14.01 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 14.01 inches below the top of the tubesheet shall be plugged upon detection.
- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. For Unit 1, 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 plugging criteria. For Unit 2, 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 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements 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. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.
  2. For the Unit 2 model D5 steam generators (Alloy 600 thermally treated) after the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each

## 5.5 Programs and Manuals

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### 5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging [or repair] criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

- a. After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;
  - b. During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and
  - c. During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.
3. For the Unit 1 model Delta-76 steam generators (Alloy 690 thermally treated) after the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging [or repair] criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the

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5.5.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator (SG) Program (continued)

inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

- a. After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
  - b. During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
  - c. During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
  - d. During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.
4. For Unit 1, if crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indications shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). For Unit 2, if crack indications are found in any SG tube from 14.01 inches below the top of the tubesheet on the hot leg side to 14.01 inches below the top of the tubesheet on the cold leg side, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indications shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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.

5.5 Programs and Manuals (continued)

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5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 and in accordance with Regulatory Guide 1.52, Revision 2, ANSI/ASME N509-1980, ANSI/ASME N510-1980, and ASTM D3803-1989.

-----NOTE-----

ANSI/ASME N510-1980, ANSI/ASME N509-1980, and ASTM D3803-1989 shall be used in place of ANSI 510-1975, ANSI/ASME N509-1976, and ASTM D3803-1979 respectively in complying with Regulatory Guide 1.52, Revision 2.

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- a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% for Primary Plant Ventilation System - ESF Filtration units and < 0.05% for all other units when tested in accordance with Regulatory

5.5 Programs and Manuals

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5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below  $\pm 10\%$ .

ESF Ventilation System	Flowrate
Control Room Emergency filtration unit	8,000 CFM
Control Room Emergency pressurization unit	800 CFM
Primary Plant Ventilation System – ESF filtration unit	15,000 CFM

- b. Demonstrate for each of the ESF systems that an in-place test of the charcoal adsorber shows a penetration and system bypass  $< 1.0\%$  for Primary Plant Ventilation System - ESF Filtration units and  $< 0.05\%$  for all other units when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below  $\pm 10\%$ .

ESF Ventilation System	Flowrate
Control Room Emergency filtration unit	8,000 CFM
Control Room Emergency pressurization unit	800 CFM
Primary Plant Ventilation System - ESF filtration unit	15,000 CFM

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of  $\leq 30^{\circ}\text{C}$  and greater than or equal to the relative humidity specified below.

ESF Ventilation Systems	Penetration	RH
Control Room Emergency filtration unit	0.5%	70%
Control Room Emergency pressurization unit	0.5%	70%
Primary Plant Ventilation System – ESF filtration unit	2.5%	70%

- d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in

5.5 Programs and Manuals

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5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

accordance with Regulatory Guide 1.52, Revision 2, and ANSI/ASME N510-1980 at the system flowrate specified below  $\pm 10\%$

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency filtration unit	8.0 in WG	8000 CFM
Control Room Emergency pressurization unit	9.5 in WG	800 CFM
Primary Plant Ventilation System – ESF filtration unit.	8.5 in WG	15000 CFM

- e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI/ASME N510-1980.

ESF Ventilation System	Wattage
Control Room Emergency pressurization unit	10 $\pm$ 1 kW
Primary Plant Ventilation System - ESF filtration unit	100 $\pm$ 5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Waste Processing System, the quantity of radioactivity contained in each Gas Decay Tank, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure," Revision 0, July 1981. The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures," Revision 2, July 1981.

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the Gaseous Waste Processing System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

## 5.5 Programs and Manuals

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### 5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

- b. A surveillance program to ensure that the quantity of radioactivity contained in each Gas Decay Tank is less than the amount that would result in a whole body exposure of  $\geq 0.5$  rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and
- c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2 to 10 CFR 20.1001 - 20.2402, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.
- d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

### 5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
  - 1. an API gravity or an absolute specific gravity within limits,
  - 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
  - 3. a clear and bright appearance with proper color or a water and sediment content within limits.
- b. Within 31 days following addition of the new fuel oil to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and
- c. Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program.

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5.5 Programs and Manuals (continued)

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5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
  1. a change in the TS incorporated in the license; or
  2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e) and exemptions thereto.

5.5.15 Safety Function Determination Program (SFDP)

- a. This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:
  1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
  2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
  3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
  4. Other appropriate limitations and remedial or compensatory actions.

## 5.5 Programs and Manuals

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### 5.5.15 Safety Function Determination Program (SFDP) (continued)

- b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:
  - 1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
  - 2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
  - 3. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.
- c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

### 5.5.16 Containment Leakage Rate Testing Program

- a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September, 1995" as modified by the following exceptions:
  - 1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.
  - 2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.
  - 3. NEI 94-01 – 1995, Section 9.2.3: The first Type A Test performed after the December 7, 1993 Type A Test (Unit 1) and the December 1, 1997 Type A Test (Unit 2) shall be performed no later than December 15, 2008 (Unit 1) and December 9, 2012 (Unit 2)."

5.5 Programs and Manuals (continued)

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5.5.16 Containment Leakage Rate Testing Program

- b. The peak calculated containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is 48.3 psig.
- c. The maximum allowable containment leakage rate,  $L_a$ , at  $P_a$ , shall be 0.10% of containment air weight per day.
- d. Leakage rate acceptance criteria are:
  - 1. Containment leakage rate acceptance criteria is  $\leq 1.0 L_a$ . During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are  $< 0.60 L_a$  for the Type B and Type C tests and  $\leq 0.75 L_a$  for Type A tests;
  - 2. Air lock testing acceptance criteria are:
    - i. Overall air lock leakage rate is  $\leq 0.05 L_a$  when tested at  $\geq P_a$ .
    - ii. For each door, leakage rate is  $\leq 0.01 L_a$  when pressurized to  $\geq P_a$ .
- e. The provision of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program, with the exception of the containment ventilation isolation valves.
- f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Technical Requirements Manual (TRM)

The TRM contains selected requirements which do not meet the criteria for inclusion in the Technical Specification but are important to the operation of CPNPP. Much of the information in the TRM was relocated from the TS.

Changes to the TRM shall be made under appropriate administrative controls and reviews. Changes may be made to the TRM without prior NRC approval provided the changes do not require either a change to the TS or NRC approval pursuant to 10 CFR 50.59. TRM changes require approval of the Plant Manager.

5.5.18 Configuration Risk Management Program (CRMP)

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to technical specification structures, systems, or components for

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## 5.5 Programs and Manuals

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### 5.5.18 Configuration Risk Management Program (CRMP) (continued)

which a risk-informed Completion Time has been granted. The program shall include the following elements:

- a. Provisions for the control and implementation of a Level 1, at-power, internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.
- b. Provisions for performing an assessment prior to entering the LCO Action for preplanned activities.
- c. Provisions for performing an assessment after entering the LCO Action for unplanned entry into the LCO Action.
- d. Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action.
- e. Provisions for considering other applicable risk significant contributors such as Level 2 issues, and external events, qualitatively or quantitatively.

### 5.5.19 Battery Monitoring and Maintenance Program

This Program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," or of the battery manufacturer for the following:

- a. Actions to restore battery cells with float voltage < 2.13 V, and
- b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates.

### 5.5.20 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Filtration System (CREFS), CRE occupants can control the reactor safety under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem whole body or its equivalent to any part of the body for the duration of the accident. The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.

5.5 Programs and Manuals

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5.5.20 Control Room Envelope Habitability Program (continued)

- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.
- c. Requirements for (i) determining the unfiltered air leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

- 1. C. - Section 4.3.2 "Periodic CRH Assessment" from NEI 99-03 Revision 1 will be used as input to a site specific Self Assessment procedure.
  - 2. C.1.2 - No peer reviews are required to be performed.
- d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREFS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.
  - e. The quantitative limits on unfiltered air leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air leakage measured by the testing described in paragraph c. The unfiltered air leakage limit for radiological challenges is the leakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air leakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.
  - f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered leakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.5 Programs and Manuals (continued)

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5.5.21 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
  - b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI-04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
  - c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.
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## 5.6 Reporting Requirements

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### 5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

1. WCAP-14040-NP-A; "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

### 5.6.7 Not used

### 5.6.8 PAM Report

When a report is required by the required actions of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

### 5.6.9 Unit 1 Model D76 and Unit 2 Model D5 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG,
  - b. 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 degradation mechanism,
  - f. The number and percentage of tubes plugged [or repaired] to date, and the effective plugging percentage in each steam generator,
  - g. The results of condition monitoring, including the results of tube pulls and in-situ testing,
  - h. For Unit 2, the primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to
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**ATTACHMENT 4 TO TXX-13132**

**PROPOSED TECHNICAL SPECIFICATION BASES CHANGES  
(MARKUP FOR INFORMATION ONLY)**

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BASES (continued)

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APPLICABLE  
SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser. However, the radiological dose consequence analysis for SGTR assumes the condenser is not available, and that the Atmospheric Relief Valve on the affected (ruptured) SG opens following the reactor trip / turbine trip and fails to close, thereby releasing the radioactivity directly to the atmosphere.

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

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

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LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the ~~repair~~ criteria be plugged (or repaired for Unit 1 D4 SGs only) in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~ criteria is repaired (Unit 1 D4 SGs only) or removed from service by plugging. If a tube was determined to satisfy the ~~repair~~ criteria but was not plugged (~~or repaired for Unit 1 D4 SGs only~~), 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

(continued)

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BASES

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LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage (i.e., Specification 5.5.9.1; "Unit 1 model D4 Steam Generator (SG) Program"). The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident. The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

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APPLICABILITY

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

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

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ACTIONS

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

A.1 and A.2

plugging

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube ~~repair~~ criteria but were not plugged (or repaired for Unit 1 D4 SGs only) in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of

(continued)

BASES

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ACTIONS

A.1 and A.2 (continued)

plugging

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 for Unit 1 D4 SGs only~~) 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, Condition B applies.

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

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged (or repaired for Unit 1 D4 SGs only) prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the

(continued)

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BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.1 (continued)

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.

plugging

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

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.9 until subsequent inspections support extending the inspection interval.

SR 3.4.17.2

plugging

plugging

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired (Unit 1 D4 SGs only) or removed from service by plugging. The tube repair criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

plugging

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.2 (continued)

plugging →

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged ~~(or repaired for Unit 1 D4 SGs only)~~ prior to subjecting the SG tubes to significant primary to secondary pressure differential.

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REFERENCES

1. NEI-97-06, "Steam Generator Program Guidelines."
  2. 10 CFR 50 Appendix A, GDC 19.
  3. 10 CFR 100.
  4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
  5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
  6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
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**ATTACHMENT 5 TO TXX-13132**

**RETYPE TECHNICAL SPECIFICATION BASES CHANGES  
(FOR INFORMATION ONLY)**

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**Page B 3.4-93**

BASES (continued)

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APPLICABLE  
SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser. However, the radiological dose consequence analysis for SGTR assumes the condenser is not available, and that the Atmospheric Relief Valve on the affected (ruptured) SG opens following the reactor trip / turbine trip and fails to close, thereby releasing the radioactivity directly to the atmosphere.

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

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

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LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the plugging criteria be plugged (or repaired for Unit 1 D4 SGs only) in accordance with the Steam Generator Program. |

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is repaired (Unit 1 D4 SGs only) or removed from service by plugging. If a tube was determined to satisfy the plugging 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-tube |

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BASES

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LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage (i.e., Specification 5.5.9.1; "Unit 1 model D4 Steam Generator (SG) Program"). The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident. The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

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APPLICABILITY

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

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

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ACTIONS

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

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must

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BASES

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ACTIONS

A.1 and A.2 (continued)

be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG plugging 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, Condition B applies.

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

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged (or repaired for Unit 1 D4 SGs only) prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.1 (continued)

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

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections. If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected SGs is restricted by Specification 5.5.9 until subsequent inspections support extending the inspection interval.

SR 3.4.17.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program plugging criteria is removed from service by plugging. The tube plugging criteria delineated in Specification 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.2 (continued)

criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

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

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REFERENCES

1. NEI-97-06, "Steam Generator Program Guidelines."
  2. 10 CFR 50 Appendix A, GDC 19.
  3. 10 CFR 100.
  4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
  5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
  6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
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