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Ref: 10CFR50.90

CPSES-200601056
Log # TXX-06092
File # 00236

June 7, 2006

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
RELATED TO LICENSE AMENDMENT REQUEST (LAR) 2004-010
APPLICATION FOR TECHNICAL SPECIFICATION
IMPROVEMENT REGARDING STEAM GENERATOR TUBE
INTEGRITY

REF: 1) TXU Power letter, logged TXX-05182, from Mike Blevins to the U. S. Nuclear Regulatory Commission, dated December 16, 2005.

Dear Sir or Madam:

In Reference 1, TXU Generating Company LP (TXU Power) submitted a proposed change to the Technical Specification (TS) requirements related to steam generator (SG) tube integrity (LAR 04-010). The proposed change was generally consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity."

Based on questions provided by Mr. Mohan Thadani of the NRC in an email dated April 19, 2006, TXU Power provides the following additional information regarding LAR 04-010. Attachment 1 to this letter contains the NRC questions and TXU Power's response immediately following each question.

Additionally, one clarification has been made to the Bases of the new SG Tube Integrity Technical Specification to clarify the dose consequence analysis assumptions as listed in the APPLICABLE SAFETY ANALYSES section of Bases 3.4.17.

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Attachment 2 provides the updated Technical Specification (TS) pages marked-up to reflect the proposed changes. Attachment 3 provides updated changes to the Technical Specification Bases. Attachment 4 provides the new retyped Technical Specification pages which incorporate the requested changes. Attachment 5 provides the new retyped Technical Specification Bases pages which incorporate the proposed changes. The pages provided in Attachments 2, 3, 4, and 5 are intended to fully replace those pages previously provided in Reference 1.

The additional information provided in this letter does not affect the safety analysis of the proposed Technical Specification changes, nor the determination that the proposed changes do not involve a significant hazard consideration (provided as Attachment 1 of Reference 1).

In accordance with 10CFR50.91(b), TXU Power is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new or revised licensing basis commitments.

Should you have any questions, please contact Mr. Bob Kidwell at (254) 897-5310.

I state under penalty of perjury that the foregoing is true and correct.

Executed on 7 June, 2006

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

Mike Blevins

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

RJK

Attachments
1. TXU Power Response to Request for Additional Information
2. Replacement Proposed Technical Specification Changes
3. Replacement Proposed Technical Specifications Bases Changes
4. Replacement Retyped Technical Specification Pages
5. Replacement Retyped Technical Specification Bases Pages

c - B. S. Mallett, Region IV
M. C. Thadani, NRR
Resident Inspectors, CPSES

Ms. Alice Rogers
Environmental & Consumer Safety Section
Texas Department of State Health Services
1100 West 49th Street
Austin, Texas 78756-3189

ATTACHMENT 1 to TXX-06092

COMANCHE PEAK UNITS 1 AND 2 STEAM GENERATOR TUBE INTEGRITY TECHNICAL SPECIFICATION AMENDMENT
TAC NOS. MC9498 AND MC9499
DOCKET NOS. 50-445 AND 50-446

TXU POWER RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

NRC Question:

1. In your proposed Surveillance Requirement (SR) 3.4.13.1, it states to "Perform RCS water inventory balance." In order for your proposed Technical Specifications (TSs) to be fully consistent with the Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler (TSTF-449), this SR should state: "Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance." Please discuss your plans to modify your TSs in order to be fully consistent with the TSTF-449.

TXU Power Response:

This was the present SR wording and was a pre-existing difference in the CPSES TS from the Standard Technical Specifications (STS). As such, this SR statement was not one of the markups provided in TSTF-449 and was not changed in the original LAR submittal. In order to be consistent with TSTF-449 to the maximum extent possible, this statement has been revised and the new page markups (consistent with TSTF-449) and new clean pages are included in Attachments 2 and 4 of this letter.

NRC Question:

2. Your LCO 3.4.17 states: "SG tube integrity shall be maintained AND All steam generator (SG) tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program." In this LCO and associated actions and surveillance, no distinction is made between the Unit 1 model D4 and D76 SGs and the Unit 2 model D5 SGs. SG tube integrity has not been defined for the current Unit 1 model D4 SGs. As a result, it is not clear that this LCO is applicable to these SGs.

TXU Power Response:

As stated in Section 2.0; "Proposed Change" of Attachment 1 of the original LAR (Reference 1);

> *The current Comanche Peak Steam Electric Station (CPSES) Unit 1 SGs are scheduled to be replaced at the end of the current operating cycle in the Spring of 2007. For this limited duration, TXU Generation Company LP (TXU Power) is proposing to leave the existing TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program" (administratively renamed TS 5.5.9.1, "Steam Generator (SG) Program") in place for the remainder of the current Unit 1 operating cycle to allow use of the approved alternate repair criteria and tube repair methods contained therein. The new TS 5.5.9.2 is fully consistent with the proposed TS 5.5.9 as provided by the TSTF and is clearly marked for its applicability only to the current Unit 2 SGs and the replacement Unit 1 SGs. In addition, TS 5.6.10 (marked as applicable only to the current Unit 1 D-4 SGs) has also been retained to ensure the current approved Unit 1 specific reporting criteria are retained until the Unit 1 SGs are replaced in 2007. This modification of TSs 5.5.9 and 5.6, as provided by the TSTF, is judged the best method of incorporating the existing approved alternate repair criteria and tube repair methods for the limited duration of their remaining applicability.*

SG Tube integrity was previously defined for the Unit 1 D4 SGs in the existing SR 3.4.13.2 which was replaced by the TSTF with SR 3.4.17.1. This method retains intact the currently approved SG Tube Surveillance Program TS for the Unit 1 D4 SGs (renamed Section 5.5.9.1; "Unit 1 model D4 Steam Generator (SG) Program") for the very limited duration the D4 SGs will remain in service.

The LCO statement, CONDITION statement, and both SRs each use the phrase ". . . in accordance with the Steam Generator Program" so that implementation is accomplished by turning to one of the two applicable subsections of Section 5.5.9; "Steam Generator (SG) Program" which are specific in their title as to their applicability.

NRC Question:

In addition, there have been no repair methods approved for the Unit 1 model D76 replacement SGs and the Unit 2 D5 SGs. As a result, references to repair methods in this LCO give the impression that repair is an approved approach to address SG tube degradation. References to repairs are also made in Bases Section B 3.4.17, pages B 3.4-108, 110 and 112. Please discuss your plans to modify LCO 3.4.17 and the BASES in order to address the above (i.e., the inconsistencies between the old technical specifications and their associated LCO and BASES and the proposed new technical specifications and their associated LCO and BASES).

TXU Power Response:

As stated in the cover letter for the original LAR submittal (Reference 1);

> *"Modification has been made to the Technical Specification 5.5.9: "Steam Generator (SG) Program" section as provided by the TSTF to allow continued use of the existing Unit 1 Alternate Repair Criteria (ARC) and repair methods for the remainder of its current operating cycle. The Unit 1 steam generators (SGs) are scheduled to be replaced at the end of the current operating cycle in the Spring of 2007, and a separate license amendment request will be submitted to remove the ARCs and repair methods which are not required for the replacement SGs, thereby making the resulting TSs fully consistent with the TSTF."*

By creating two totally separate Steam Generator Program subsections (5.5.9.1; "Unit 1 model D4 Steam Generator (SG) Program" and 5.5.9.2; "Unit 1 model D76 and Unit 2 model D5 Steam Generator (SG) Program"), it is easy to tell which SGs have an approved ARC or repair method and which do not. Since the new LCO 3.4.17 uses the words "*. . . in accordance with the Steam Generator Program*" in both the LCO and SR sections, anyone implementing this TS will turn to the appropriate Steam Generator Program subsection of 5.5.9.

In order to provide further clarification, the following parenthetical statement "*(or repaired for Unit 1 Model D4 SGs only)*" has been inserted each time a repair method is mentioned in Bases Section B 3.4.17, pages B 3.4-108, 110, 111 and 113. New page markups and clean pages are included in Attachments 3 and 5 of this letter.

NRC Question:

3. In your proposed TS Section 5.6.9.e, you indicated that the number of tubes plugged [or repaired] during the inspection outage for each active degradation mechanism would be reported. Given that no repair methods have been approved for the Unit 1 model D76 and Unit 2 model D5 SGs, please discuss your plans to remove "[or repaired]" from your proposal.

TXU Power Response:

The bracketed *[or repaired]* should have been deleted from the TSTF-provided insert in our original submittal since this new section 5.6.9 is only applicable to the D5 and D76 SGs. This statement has been revised and the new page markups and clean pages are included in Attachments 2 and 4 of this letter.

NRC Question:

4. In your proposed TS Bases, pages 3.4-21, 25, 31, and 36, for all modes of operation, you proposed to eliminate the following statement: "...in accordance with the Steam Generator Tube Surveillance Program." The elimination of this statement is consistent with TSTF-449; however, it is only appropriate for the Unit 1 model D76 replacement SGs and the Unit 2 D5 SGs. This statement is still applicable to the current Unit 1 model D4 SGs. Please discuss your plans to add a paragraph in which the distinctions between the current Unit 1model D4 SGs and the Unit 1 model D76 replacement SGs and the Unit 2 D5 SGs are made with respect to the above statement.

TXU Power Response:

The current section 5.5.9 "Steam Generator (SG) Tube Surveillance Program" is being renumbered and renamed as Section 5.5.9.1 "*Unit 1 model D4 Steam Generator (SG) Program*" by this proposed LAR. This administrative renumbering and retitling is the only change being made to this section and all current content remains unchanged. This retains the appropriate information for the D4 SGs while preventing having to use an applicability disclaimer every time there is a difference between units in the Bases. By referring to Section 5.5.9 "Steam Generator (SG) Program" in all cases, we are able to have 2 separate subsections; 5.5.9.1 (clearly labeled for the D4 SGs only) and 5.5.9.2 (clearly labeled for the D5 and D76 SGs only).

NRC Question:

5. In your proposed TS Bases, page 3.4-36 of your TSs, the last paragraph of this page states; "An OPERABLE SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE." Please discuss your plans to remove the word "OPERABLE" at the beginning of the sentence consistent with TSTF-449.

TXU Power Response:

This sentence has been revised and the new page markups and clean pages are included in Attachments 3 and 5 of this letter.

NRC Question:

6. In your Bases proposed for SR 3.4.13.2, the second paragraph states: "For RCS primary to secondary LEAKAGE determination, steady state is defined as stable pressure, temperature (Tavg changing by less than 5 F/hour), power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows." The parenthetical temperature criterion does not appear in the TSTF-449 bases.

Please explain why you added this criteria to the TSTF-449 wording and why such a clarification is only needed for temperature. Alternatively, please discuss your plans to remove this statement which would make your proposal consistent with TSTF-449.

TXU Power Response:

This was the existing SR 3.4.13.1 Bases wording that has been in place at CPSES prior to and after conversion to the STS. Since it was not a marked up 'change' via the TSTF, the same wording was carried over to the TSTF-provided insert for SR 3.4.13.2 (and SR 3.4.13.1) Bases to prevent having a difference between the two SR Bases.

After further review, CPSES believes that this criterion is better controlled at the procedural level instead of within the Bases and in order to be consistent with TSTF-449, this statement has been revised in both SR Bases (3.4.13.1 and 3.4.13.2) and the new page markups and clean pages are included in Attachments 3 and 5 of this letter.

NRC Question:

7. In your proposed TS Bases Section B 3.4.17, the last paragraph on page B 3.4-109 states: "The accident analysis assumes accident induced leakage does not exceed 1gpm per SG, except for specific types of degradation at specific locations where the NRC has approved a greater accident induced leakage." This statement is only applicable to the Unit 1 model D4 SGs. Since there is no distinction made between units, please discuss your plans to modify this paragraph in order to distinguish the difference between the units.

TXU Power Response:

In order to clarify the applicability of that Bases statement, the following change to the bracketed insert provided by TSTF-449 is proposed to point out that it is only the Unit 1 D4 SGs to which the second half of the statement is applicable.

The final wording would be:

> *The accident analysis assumes accident induced leakage does not exceed 1gpm per SG, except for specific types of degradation at specific locations where the NRC has approved a greater accident induced leakage <u>(i.e., Specification 5.5.9.1 "Unit 1 model D4 Steam Generator (SG) Program")</u>.*

This change clarifies the applicability of the different portions of the sentence. Once the D4 SGs are replaced, this clarification will be removed by means of a separate license amendment submitted to remove the ARCs and repair methods which are not required for the remaining Unit 1 and 2 SGs, thereby making the resulting Bases fully consistent with the TSTF.

This statement has been revised in the proposed Bases page markups and clean pages as provided in Attachments 3 and 5 of this letter.

ATTACHMENT 2 to TXX-06092

REPLACEMENT PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Pages ii
1.1-4
3.4-33
3.4-34
3.4-48 (new page)
3.4-49 (new page)
5.0-13
5.0-14
5.0-15
5.0-15a
5.0-16
5.0-16a
5.0-17
5.0-17a
5.0-18
5.0-19
5.0-19a
Insert 5.5.9.2 (page 1 of 3)
Insert 5.5.9.2 (page 2 of 3)
Insert 5.5.9.2 (page 3 of 3)
5.0-36
5.0-36a
Insert 5.6.9

TABLE OF CONTENTS (continued)

(continued)

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

(primary to secondary LEAKAGE)

2. 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

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator ~~(SG)~~ to the Secondary System;

b. Unidentified LEAKAGE

primary to secondary

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except ~~SG~~ LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping or total steps.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

~~d. 1 gpm total primary to secondary LEAKAGE through all steam generators for Unit 2 (SGs); and~~ | 70

d ~~e.~~ 150 gallons per day ~~for Unit 1 and 500 gallons per day for Unit 2~~ primary to secondary LEAKAGE through any one SG. | 70

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS Operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

S

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------------------------------NOTE.------------------------------ Not required to be performed until 12 hours after establishment of steady state operation. -- ~~Perform~~ RCS water inventory balance.	72 hours
SR 3.4.13.2	~~Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.~~	~~In accordance with the Steam Generator Tube Surveillance Program~~

72 hours

1.
2. Not applicable to primary to secondary LEAKAGE.

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

Verify primary to secondary LEAKAGE is ≤150 gallons per day through any one SG.

Verify RCS operational LEAKAGE is within limits by performance of

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 repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired 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
	AND	
	A.2 Plug or repair 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
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

NEW PAGE 3.4.48

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 repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

NEW PAGE 3.4.49

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) ~~Tube Surveillance~~ Program

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

The provisions of SR 3.0.2 are applicable to the SG Surveillance Program test frequencies.

a. 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 5.5-1.

repaired

b. 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 5.5-2 or 5.5-3. Table 5.5-2 applies to all tubes except ~~reapaired~~ tubes (Unit 1 only) which are covered by Table 5.5-3. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.9~~d~~ ~~and~~ the inspected tubes shall be verified acceptable per the acceptance criteria ~~of~~ Specification 5.5.9~~e~~. The tubes selected for each inservice inspection per Table 5.5-2 shall include at least 3% of all the expanded tubes and at least 3% of the remaining number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except: 83 71 83

.1

1. 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;

2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

   b) Tubes in those areas where experience has indicated potential problems, and

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program

5.5 Programs and Manuals .1

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

c) A tube inspection (pursuant to Specification 5.5.9e.1.h) 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. 71

d) Indications left in service as a result of the application of the tube support plate voltage repair criteria shall be inspected by bobbin probe during all future refueling outages. 70

3. The tubes selected as the second and third samples (if required by Table 5.5.~~9~~-2 during each inservice inspection may be subjected to a partial tube inspection provided:

a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

b) The inspections include those portions of the tubes where imperfections were previously found.

4. Implementation of the steam generator tube/tube support plate repair criteria requires a 100% bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg support with known outside diameter stress corrosion cracking (ODSCC) indications. The Determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of the tubes inspected over their full length. 70

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

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

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.

c. Steam Generator F* Tube Inspection (Unit 1 only) - In addition to the minimum sample size as determined by Specification 5.5.9.b., all F* tubes will be inspected within the tubesheet region. The results of the inspections of F* tubes identified in previous inspections will not be a cause for additional inspections per Tables 5.5-1 and 5.5-2. 71

d. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies: 71

1. The first inservice inspection shall be performed after 6 Effective Full Power Months (EFPM) and before 12 EFPM and shall include a special inspection of all expanded tubes in all steam generators. 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, 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;

.1

2. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5-2 at 40-month intervals fall in 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 5.5.9d.1; the interval may then be extended to a maximum of once per 40 months; and 71

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

5.5 Programs and Manuals

~~5.5.9~~ ~~Steam Generator (SG) Tube Surveillance Program~~ ~~(continued)~~

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5-2 during the shutdown subsequent to any of the following conditions:

   a) Primary-to secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification ~~3.4.5.2~~, or 3.4.13

   b) A seismic occurrence greater than the Operating Basis Earthquake, or

   c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   d) A main steam line or feedwater line break.

e. Acceptance Criteria 71

1. As used in this specification:

   a) 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;

   b) Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

   c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

   d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

   e) Defect means an imperfection of such severity that it exceeds the plugging limit or (for Unit 1 only) repair limit. A tube containing a defect is defective; 83

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

5.5 Programs and Manuals

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

f) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging 83
or (for Unit 1 only) repaired by sleeving and is equal to 40% of the wall thickness. The plugging limit for laser welded sleeves is equal to 43% of the nominal wall thickness. The plugging limit for Leak Tight sleeves is equal to 20% of the nominal wall thickness. This definition does not apply to that portion of the Unit 1 tubing that 101
meets the definition of an F* tube. This definition does not apply to 71
tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 5.5.9.1m) for the 70
repair limit applicable to these intersections. All tubes repaired with Leak Limiting sleeves shall be plugged upon detection of 71
degradation in the sleeve and/or pressure boundary portion of the original tube wall in the sleeve/tube assembly (i.e., the sleeve-to- 112
tube joint) regardless of depth. The F* criteria is not applicable to the parent tube located behind the Leak Limiting sleeves installed in the tubesheet transition zone; .1

g) 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 Specification 5.5.9d.3, above; 71

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube repaired by sleeving (for 83
Unit 1 only) the tube inspection shall include the sleeved portion of the tube;

i) 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 shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections; 71

j) F* Distance (Unit 1 only) is the distance of the hardroll expanded 71
portion of a tube which provides a sufficient length of non-degraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.13 inches, plus an allowance for eddy current measurement uncertainty, and is measured down from the top of the tubesheet, or the bottom of the roll transition, whichever is lower in elevation. The F* criteria is not applicable to the parent tube located behind the Leak Limiting sleeves installed in the 112
tubesheet transition zone;

k) F* Tube (Unit 1 only) is that portion of the tubing in the area of the 71
tubesheet region below the F* distance with a) degradation below the F* distance equal to or greater than 40%, b) which has no indication of degradation within the F* distance, and c) that remains inservice;

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

5.5 Programs and Manuals

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

l) Hard Roll Expansion (Unit 1 only) is that portion of a tube which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the tube and the hole in the tubesheet; and 71

m) For Unit 1 only, the Tube Support Plate Plugging Limit is used for the disposition of alloy 600 steam generator tubes for continued service that are experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffle (FDB). At tube support plate intersections (and FDB), the plugging limit is based on maintaining steam generator tube serviceability as described below: 70

1. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to the lower voltage repair limit (1.0 volt), will be allowed to remain in service. 70 .1

2. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with the bobbin voltage greater than the lower voltage repair limit (1.0 volt), will be repaired, except as noted in 5.5.9e.1m)3. below. 71

3. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (1.0 volt) but less than or equal to the upper voltage repair limit*, may remain inservice if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper repair limit** will be plugged or repaired. 70

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued) (continued)

* The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. 70

** V_{URL} will differ at the TSPs and flow distribution baffle.

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

4. Certain intersections as identified in WPT-15949 will be excluded from application of the voltage-based repair criteria as it is determined that these intersections may collapse or deform following a postulated LOCA + SSE event. 70

.1

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9e.1.m)1., 5.5.9e.1.m)2., and 5.5.9e.1.m)3. The midcycle repair limits are determined from the following equations: 71 70

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left[\frac{CL - \Delta t}{CL}\right]$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MLRL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth per cycle
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by the NRC)

.1

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9e.1.m)1., 5.5.9e.1m)2., and 5.5.9e.1.m)3. 71

n. Tube Repair (for Unit 1 only) refers to the process that establishes tube serviceability. Acceptable tube repairs will be performed in accordance with the process described in Westinghouse WCAP-13698, Rev. 3 and Westinghouse Letter WPT-16094 dated March 20, 2000, WCAP-15090, Rev. 1, CEN-630-P, Rev. 2 dated June 1997, and WCAP-15918, Rev. 1, dated January, 2004. 83 112

(continued)

5.5 Programs and Manuals

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5-2 and Table 5.5-3. 101

(continued)

5.5.9.1	Unit 1 model D4 Steam Generator (SG) Program (continued)

5.5 Programs and Manuals

~~5.5.9. Steam Generator (SG) Tube Surveillance Program~~ ~~(continued)~~

TABLE 5.5-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Four
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATIONS

1. The two steam generators that were not inspected during the first inservice inspection shall be inspected during the second and third inspections, one in each inspection period. For the fourth and subsequent inspections, the inservice inspection may be limited to one steam generator on a rotating schedule encompassing 12% of the tubes if the results of the previous inspections of the four steam generators 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.

(continued)

5.5.9.1	Unit 1 model D4 Steam Generator (SG) Program (continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

5.5 Programs and Manuals

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

TABLE 5.5-2

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 repair* defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.	83
			C-2	Plug or repair* defective tubes and inspect additional 4S tubes in this S.G.	C-1	None	
					C-2	Plug or repair* 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 repair* defective tubes and inspect 2S tubes in each other S.G.	All other S.G.s are C-1	None	N.A.	N.A.	83
			Some S.G.s C-2 but no additional S.G. C-3	Perform action for C-2 result of second sample	N.A.	N.A.	103
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair* defective tubes.	N.A.	N.A.	83 103

(continued)

S = 12/n% Where n is the number of steam generators inspected during an inspection 83
* for Unit 1 only

5.5 Programs and Manuals

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

TABLE 5.5-3 83

STEAM GENERATOR REPAIRED TUBE INSPECTION FOR UNIT 1 ONLY

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes (1)	C-1	None	N.A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective tubes and inspect 20% of the repaired tubes in each other S.G.	All other S.G.s are C-1	None
			Same S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample 103
			Additional S.G is C-3	Inspect all repaired tubes in each S.G. and plug defective tubes. 83 103

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

(1) Each repair method is considered a separate population for determination of initial inservice inspection and scope expansion.

INSERT new section 5.5.9.2 on the next page

5.5.9.2 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."

(continued)

5.5.9.2 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.

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

2a. 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 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.]

2b. 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.

(continued)

5.5.9.2 Unit 1 model D76 and Unit 2 model D5 Steam Generator (SG) Program (continued)

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.

5.6 Reporting Requirements (continued)

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 ~~Not used~~ INSERT 5.6.9

5.6.10 ~~Steam Generator Tube Inspection Report~~

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged, repaired or designated as an F* tube in each steam generator shall be reported to the Commission; 83

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a report within 12 months following the completion of the inspection. This report shall include:

1) Number and extent of tubes and (for Unit 1 only) sleeves inspected, 83

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired. 83

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission in a report 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. 103

(continued)

Unit 1 model D4 Steam Generator Tube Inspection Report

5.6 Reporting Requirements (continued)

5.6.10 ~~Steam Generator Tube Inspection Report (continued)~~

d. For implementation of the voltage based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise: 70

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

Unit 1 model D4 Steam Generator Tube Inspection Report (continued)

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 5.5.9.2, 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,

Insert 5.6.9

ATTACHMENT 3 to TXX-06092

REPLACEMENT PROPOSED TECHNICAL SPECIFICATIONS BASES CHANGES

Pages B ii
B 3.4-21
B 3.4-25
B 3.4-31
B 3.4-36
B 3.4-80
B 3.4-81
B 3.4-82
B 3.4-83
B 3.4-84
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B 3.4-107 (new page)
B 3.4-108 (new page)
B 3.4-109 (new page)
B 3.4-110 (new page)
B 3.4-111 (new page)
B 3.4-112 (new page)
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(continued)

BASES (continued)

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required to be in operation at power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops —MODE 3";
LCO 3.4.6, "RCS Loops —MODE 4";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

(continued)

BASES

LCO (continued)

is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron dilution with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program,~~ which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

(continued)

BASES

LCO (continued)

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program,~~ which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODES 1 and 2";
LCO 3.4.5, "RCS Loops — MODE 3";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops— MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one required loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

LCO (continued)

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron dilution with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be $\leq$ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature $\leq$ 350°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A~~n OPERABLE~~ SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE~~ in accordance with the Steam Generator Tube Surveillance Program~~.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA). Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes ~~a 1 gpm primary to secondary LEAKAGE as the initial condition.~~

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

safety analysis assumption

The FSAR (Ref. 3) analysis for SGTR assumes the the secondary fluid is released to the atmosphere via the atmospheric relief valves on the affected steam generator. This valve is assumed to fail to close. The release continues until the reactor operators close the associated block valve. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

the entire

The safety analysis for the SLB accident assumes ~~1 gpm~~ primary to secondary LEAKAGE ~~in one~~ generator as an initial condition. The dose consequences resulting from the SLB accident are within the limits defined in 10 CFR 100 (Ref. 6) as described in the accident analyses (Ref. 3).

is through the affected

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4 and 5).

The RCS operational LEAKAGE satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

(continued)

that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

BASES (continued)

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Seals and gaskets include the canopy seals downstream of ACME threaded connections and Canopy Seal Clamp Assemblies. Therefore, leakage past the canopy seal or CSCAs is not pressure boundary leakage.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and Containment Sump Level and Flow Monitoring System can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

~~d. Primary to Secondary LEAKAGE through All Steam Generators (SGs) (Unit 2 only)~~

~~Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

(continued)

BASES

LCO (continued)

~~e. Primary to Secondary LEAKAGE through Any One SG~~

~~For Unit 2, the 500 gallons per day limit on one SG 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 rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.~~

~~For Unit 1, maintaining an operating leakage of 150 gpd per steam generator (0.1 gpm - at room temperature) (600 gpd total) minimizes the potential for a large leakage event during a main steam line break. Based on the non-destructive examination uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line break is limited to below 27.79 gpm (calculated at room temperature conditions) for Comanche Peak Unit 1 in the faulted loop. Maintaining leakage within the 27.79 gpm limit will ensure that offsite doses will remain within 10 CFR Part 100 guidelines and within control room dose (GDC-19) guidelines. Leakage in the intact loops will be limited to a leak rate of 150 gpd. If the projected end-of-cycle distribution of crack indications results in primary-to-secondary leakage greater than 27.79 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce steam line break leakage to below this limit.~~

~~The leakage limits incorporated in 5.5.9 are more restrictive than the standard operating license limits and are intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shutdown in a timely manner.~~

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

(continued)

d. The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 7). 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 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.

BASES (continued)

APPLICABILITY (continued)

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

or

Unidentified LEAKAGE~~,~~ identified LEAKAGE~~, or primary to secondary LEAKAGE~~ in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

or primary to secondary LEAKAGE is not within limit,

or

If any pressure boundary LEAKAGE exists~~,~~ or if unidentified ~~LEAKAGE,~~ identified LEAKAGE~~, or primary to secondary LEAKAGE~~ cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

~~(continued)~~

(continued)

This surveillance is modified by two Notes.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

1 states

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). ~~Therefore, a~~ Note ~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation near operating pressure. The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature ~~(T_{avg} changing by less than 5°F/hour)~~, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. When non steady state operation precludes surveillance performance, the surveillance should be performed in a reasonable time period commensurate with the surveillance performance length, once steady state operation has been achieved, provided greater than 72 hours have elapsed since the last performance.

(continued)

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

BASES

SUVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions. This surveillance does not tie directly to any of the leakage criteria in the LCO or of the CONDITIONS; therefore failure to meet this surveillance is considered failure to meet the integrity goals of the LCO and LCO 3.0.3 applies.~~

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, the performance criterion is not met and LCO 3.4.17, "Steam Generator Tube Integrity," should be entered. The 150 gallons per day limit is measured at room temperature as described in Reference 8. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 8).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. FSAR, Section 3.6B.
5. NUREG-1061, Volume 3, November 1984.
6. 10 CFR 100.
7. NEI 97-06, "Steam Generator Program Guidelines".
8. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines".

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 SG Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied 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 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

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 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, 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 5.5.9. 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 the Steam Generator Program Guidelines (Ref. 1).

(continued)

BASES (continued)

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).

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 the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

(continued)

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BASES

LCO (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "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 have a significant effect on burst or collapse. In that context, the term "significant" 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 (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

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

(continued)

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BASES

LCO (continued)

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.

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.

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.

<u>A.1 and A.2</u>

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 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

(continued)

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BASES

ACTIONS (continued)

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.

<u>B.1 and B.2</u>

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.

SURVEILLANCE REQUIREMENTS

<u>SR 3.4.17.1</u>

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 content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

(continued)

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BASES (continued)

SUVEILLANCE REQUIREMENTS (continued)

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 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.

SR 3.4.17.2

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 repaircriteria, 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.

(continued)

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BASES (continued)

SUVEILLANCE REQUIREMENTS (continued)

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.

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 4 to TXX-06092

REPLACEMENT RETYPED TECHNICAL SPECIFICATION PAGES

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(continued)

1.1 Definitions (continued)

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. 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

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel req uired for channel OPERABILITY and verifying the OPERABILITY of each req uired master relay. The MASTER RELAY TEST shall include a continuity check of each associated req uired slave relay. The MASTER RELAY TEST may be performed by means of any series of seq uential, overlapping or total steps.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limits	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	-------------------------------NOTES------------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	-------------------------------NOTE------------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is $\leq$150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

<u>AND</u>

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

-----------------------------------NOTE-----------------------------------

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged or repaired 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 or repair 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
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
<u>OR</u>	<u>AND</u>	
SG tube integrity not maintained.	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 repair criteria is plugged or repaired in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program

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

The provisions of SR 3.0.2 are applicable to the SG Surveillance Program test frequencies.

a. 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 5.5-1.

b. 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 5.5-2 or 5.5-3. Table 5.5-2 applies to all tubes except repaired tubes (Unit 1 only) which are covered by Table 5.5-3. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.9.1d., and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.1e. The tubes selected for each inservice inspection per Table 5.5-2 shall include at least 3% of all the expanded tubes and at least 3% of the remaining number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

1. 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;

2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

b) Tubes in those areas where experience has indicated potential problems, and

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

c) A tube inspection (pursuant to Specification 5.5.9.1e.1.h) 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.

d) Indications left in service as a result of the application of the tube support plate voltage repair criteria shall be inspected by bobbin probe during all future refueling outages.

3. The tubes selected as the second and third samples (if required by Table 5.5-2 during each inservice inspection may be subjected to a partial tube inspection provided:

a) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

b) The inspections include those portions of the tubes where imperfections were previously found.

4. Implementation of the steam generator tube/tube support plate repair criteria requires a 100% bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg support with known outside diameter stress corrosion cracking (ODSCC) indications. The Determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20% random sampling of the tubes inspected over their full length.

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

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

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.

c. Steam Generator F* Tube Inspection (Unit 1 only) - In addition to the minimum sample size as determined by Specification 5.5.9.1b., all F* tubes will be inspected within the tubesheet region. The results of the inspections of F* tubes identified in previous inspections will not be a cause for additional inspections per Tables 5.5-1 and 5.5-2.

d. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power Months (EFPM) and before 12 EFPM and shall include a special inspection of all expanded tubes in all steam generators. 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, 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;

2. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5-2 at 40-month intervals fall in 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 5.5.9.1d.1; the interval may then be extended to a maximum of once per 40 months; and

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5-2 during the shutdown subsequent to any of the following conditions:

   a) Primary-to secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13, or

   b) A seismic occurrence greater than the Operating Basis Earthquake, or

   c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   d) A main steam line or feedwater line break.

e. Acceptance Criteria

1. As used in this specification:

   a) 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;

   b) Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

   c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

   d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

   e) Defect means an imperfection of such severity that it exceeds the plugging limit or (for Unit 1 only) repair limit. A tube containing a defect is defective;

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

f) Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or (for Unit 1 only) repaired by sleeving and is equal to 40% of the wall thickness. The plugging limit for laser welded sleeves is equal to 43% of the nominal wall thickness. The plugging limit for Leak Tight sleeves is equal to 20% of the nominal wall thickness. This definition does not apply to that portion of the Unit 1 tubing that meets the definition of an F* tube. This definition does not apply to tube support plate intersections for which the voltage-based plugging criteria are being applied. Refer to 5.5.9.1e.1m) for the repair limit applicable to these intersections. All tubes repaired with Leak Limiting sleeves shall be plugged upon detection of degradation in the sleeve and/or pressure boundary portion of the original tube wall in the sleeve/tube assembly (i.e., the sleeve-to-tube joint) regardless of depth. The F* criteria is not applicable to the parent tube located behind the Leak Limiting sleeves installed in the tubesheet transition zone;

g) 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 Specification 5.5.9.1d.3, above;

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube repaired by sleeving (for Unit 1 only) the tube inspection shall include the sleeved portion of the tube;

i) 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 shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections;

j) F* Distance (Unit 1 only) is the distance of the hardroll expanded portion of a tube which provides a sufficient length of non-degraded tube expansion to resist pullout of the tube from the tubesheet. The F* distance is equal to 1.13 inches, plus an allowance for eddy current measurement uncertainty, and is measured down from the top of the tubesheet, or the bottom of the roll transition, whichever is lower in elevation. The F* criteria is not applicable to the parent tube located behind the Leak Limiting sleeves installed in the tubesheet transition zone;

k) F* Tube (Unit 1 only) is that portion of the tubing in the area of the tubesheet region below the F* distance with a) degradation below the F* distance equal to or greater than 40%, b) which has no indication of degradation within the F* distance, and c) that remains inservice;

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

l) Hard Roll Expansion (Unit 1 only) is that portion of a tube which has been increased in diameter by a rolling process such that no crevice exists between the outside diameter of the tube and the hole in the tubesheet; and

m) For Unit 1 only, the Tube Support Plate Plugging Limit is used for the disposition of alloy 600 steam generator tubes for continued service that are experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffle (FDB). At tube support plate intersections (and FDB), the plugging limit is based on maintaining steam generator tube serviceability as described below:

1. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to the lower voltage repair limit (1.0 volt), will be allowed to remain in service.

2. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with the bobbin voltage greater than the lower voltage repair limit (1.0 volt), will be repaired, except as noted in 5.5.9.1e.1m)3. below.

3. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (1.0 volt) but less than or equal to the upper voltage repair limit*, may remain inservice if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper repair limit** will be plugged or repaired.

(continued)

* The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented.

** V_{URL} will differ at the TSPs and flow distribution baffle.

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

4. Certain intersections as identified in WPT-15949 will be excluded from application of the voltage-based repair criteria as it is determined that these intersections may collapse or deform following a postulated LOCA + SSE event.

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.5.9.1e.1.m)1., 5.5.9.1e.1.m)2., and 5.5.9.1e.1.m)3. The midcycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left[\frac{CL - \Delta t}{CL}\right]$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MLRL} and time into cycle
)t	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth per cycle
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 5.5.9.1e.1.m)1., 5.5.9.1e.1m)2., and 5.5.9.1e.1.m)3.

n. Tube Repair (for Unit 1 only) refers to the process that establishes tube serviceability. Acceptable tube repairs will be performed in accordance with the process described in Westinghouse WCAP-13698, Rev. 3 and Westinghouse Letter WPT-16094 dated March 20, 2000, WCAP-15090, Rev. 1, CEN-630-P, Rev. 2 dated June 1997, and WCAP-15918, Rev. 1, dated January, 2004.

(continued)

5.5 Programs and Manuals

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5-2 and Table 5.5-3.

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

TABLE 5.5-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	Four
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[1]

TABLE NOTATIONS

1. The two steam generators that were not inspected during the first inservice inspection shall be inspected during the second and third inspections, one in each inspection period. For the fourth and subsequent inspections, the inservice inspection may be limited to one steam generator on a rotating schedule encompassing 12% of the tubes if the results of the previous inspections of the four steam generators 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.

(continued)

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

TABLE 5.5-2

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 repair* defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair* defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair* 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 repair* defective tubes and inspect 2S tubes in each other S.G.	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S.G. C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair* defective tubes.	N.A.	N.A.

(continued)

S = 12/n% Where n is the number of steam generators inspected during an inspection
* for Unit 1 only

5.5.9.1 Unit 1 model D4 Steam Generator (SG) Program (continued)

TABLE 5.5-3

STEAM GENERATOR REPAIRED TUBE INSPECTION FOR UNIT 1 ONLY

1^{ST} SAMPLE INSPECTION			2^{ND} SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of repaired tubes (1)	C-1	None	N.A.	N.A.
	C-2	Plug defective repaired tubes and inspect 100% of the repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective tubes and inspect 20% of the repaired tubes in each other S.G.	All other S.G.s are C-1	None
			Same S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of first sample
			Additional S.G is C-3	Inspect all repaired tubes in each S.G. and plug defective tubes.

(continued)

(1) Each repair method is considered a separate population for determination of initial inservice inspection and scope expansion.

5.5 Programs and Manuals (continued)

5.5.9.2 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."

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2 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.

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

2a. 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 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.]

2b. 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.

(continued)

5.5 Programs and Manuals (continued)

5.5.9.2 Unit 1 model D76 and Unit 2 model D5 Steam Generator (SG) Program (continued)

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.

(continued)

5.6 Reporting Requirements (continued)

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 5.5.9.2, 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,

(continued)

5.6 Reporting Requirements (continued)

5.6.10 Unit 1 model D4 Steam Generator Tube Inspection Report

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged, repaired or designated as an F* tube in each steam generator shall be reported to the Commission;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a report within 12 months following the completion of the inspection. This report shall include:

1) Number and extent of tubes and (for Unit 1 only) sleeves inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission in a report 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.

(continued)

5.6 Reporting Requirements (continued)

5.6.10 Unit 1 model D4 Steam Generator Tube Inspection Report (continued)

d. For implementation of the voltage based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

ATTACHMENT 5 to TXX-06092

REPLACEMENT RETYPED TECHNICAL SPECIFICATION BASES PAGES

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BASES (continued)

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required to be in operation at power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops — MODE 3";
LCO 3.4.6, "RCS Loops — MODE 4";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

(continued)

BASES

LCO (continued)

is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron dilution with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

(continued)

BASES

LCO (continued)

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODES 1 and 2";
LCO 3.4.5, "RCS Loops — MODE 3";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops— MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one required loop is inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

LCO (continued)

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron dilution with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be $\leq$ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature $\leq$ 350°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA). Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the secondary fluid is released to the atmosphere via the atmospheric relief valves on the affected steam generator. This valve is assumed to fail to close. The release continues until the reactor operators close the associated block valve. The 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The safety analysis for the SLB accident assumes the entire primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are within the limits defined in 10 CFR 100 (Ref. 6) as described in the accident analyses (Ref. 3).

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per RG 1.45 (Ref. 2) are maintained (Ref. 4 and 5).

The RCS operational LEAKAGE satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Seals and gaskets include the canopy seals downstream of ACME threaded connections and Canopy Seal Clamp Assemblies. Therefore, leakage past the canopy seal or CSCAs is not pressure boundary leakage.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and Containment Sump Level and Flow Monitoring System can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 7). 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 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.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

(continued)

BASES (continued)

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE, cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). This surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation near operating pressure. The 12 hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. When non steady state operation precludes surveillance performance, the surveillance should be performed in a reasonable time period commensurate with the surveillance performance length, once steady state operation has been achieved, provided greater than 72 hours have elapsed since the last performance.

(continued)

BASES (continued)

SUVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, the performance criterion is not met and LCO 3.4.17, "Steam Generator Tube Integrity," should be entered. The 150 gallons per day limit is measured at room temperature as described in Reference 8. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 8).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. FSAR, Section 3.6B.
5. NUREG-1061, Volume 3, November 1984.
6. 10 CFR 100.
7. NEI 97-06, "Steam Generator Program Guidelines".
8. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines".

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 SG Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied 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 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled."

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 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, 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 5.5.9. 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 the Steam Generator Program Guidelines (Ref. 1).

(continued)

BASES (continued)

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).

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 the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

(continued)

BASES

LCO (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "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 have a significant effect on burst or collapse. In that context, the term "significant" 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 (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

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

(continued)

BASES

LCO (continued)	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.
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.
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. <u>A.1 and A.2</u> 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 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

(continued)

BASES

ACTIONS (continued)

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.

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 content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

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BASES (continued)

SUVEILLANCE REQUIREMENTS (continued)

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 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.

SR 3.4.17.2

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 repaircriteria, 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.

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BASES (continued)

SUVEILLANCE REQUIREMENTS (continued)

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.

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."